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**Sensitivity of Primary Knock-on Atom
Spectra and Displacement per Atom
Cross-Sections to Different Secondary
Neutron Energy and Angular Distributions
and "In-Group" Weighting Schemes**

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$4.50; Microfiche \$3.00

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Contract No. W-7405-eng-26

Neutron Physics Division

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DIFFERENT SECONDARY NEUTRON ENERGY
AND ANGULAR DISTRIBUTIONS AND
"IN-GROUP" WEIGHTING SCHEMES*

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Date Published - January 1978

* Submitted for journal publication.
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Abstract

The sensitivity of primary knock-on atom (PKA) spectra and displacement per atom (DPA) cross sections to different secondary neutron energy and angular distributions and "in-group" weighting schemes is investigated. It is shown that the sensitivity of the PKA spectra and DPA cross sections for the $(n,n'$ unresolved) and $(n,2n)$ reactions in Fe to different angular distributions and the same secondary neutron spectrum is reasonably large ($\sim 15\%$) whereas the sensitivity of these quantities to grossly different secondary neutron spectra and the same angular distribution is unexpectedly small. It will also be shown that for Al, the sensitivity of damage energy cross sections to different "in-group" weighting schemes is, for the most part, small.

I. INTRODUCTION

Displacements per atom (DPA) and gas production cross sections are used to determine the effective damage levels that can be expected in structural metals when they are exposed to radiation environments. For neutron radiation environments, ENDFB/IV provides directly, in most cases, reasonable data for the gas production cross sections. However, DPA cross sections are obtained from primary knock-on atom (PKA) spectra that must be deduced from ENDFB/IV data. For elastic and inelastic resolved neutron reactions, ENDFB/IV, in most cases, provides the necessary data by which PKA spectra can be obtained with essentially no approximations or assumptions. However, in general, for the remaining reactions, $(n,n'$ unresolved), (n,p) , $(n,2n)$, etc., there are not sufficient data given so that totally definitive PKA spectra can be obtained without resorting to various assumptions and approximations. For example, one of the most common assumptions is that the angular distribution of the secondary neutrons is isotropic in the center-of-mass (C-M) system and that the secondary neutron energy distribution can be obtained from ENDFB/IV or from some sort of evaporation-type spectrum.

Presented in this paper is a set of calculations in which the sensitivity of PKA spectra and DPA cross sections for the $(n,n'$ unresolved) and $(n,2n)$ reactions in Fe to different secondary neutron energy and angular distributions is investigated. It will be shown that the sensitivity of the PKA spectrum and DPA cross sections to different angular distributions and the same secondary neutron spectrum is reasonably large ($\sim 15\%$) whereas for grossly different secondary neutron spectra and the same angular distribution, the sensitivity is unexpectedly small.

Also presented is the sensitivity of the damage energy cross sections to different "in-group" weighting, for example, constant, $1/E$, $1/E\sigma_T$. It is shown that for Al, the sensitivity is extremely small for all weightings except $1/E\sigma_T$ and even this weighting produces only slightly different integral results.

The methods of calculation are presented in Sec. II and the results are presented and discussed in Sec. III.

II. METHODS OF CALCULATION

The neutron and PKA-energy group structures used in the calculations are given in Tables I and II. The neutron group structure up to 14.91 MeV is the same as the GAM-II group structure.¹ The PKA group structure used is based on equal lethargy intervals from 20 eV to 10 MeV.

For elastic, inelastic, $(n,2n)$, $(n,3n)$, (n,np) , etc. reactions involving the emission of mass 1 particles and excluding absorption reactions, a modified version of the XLACS module from the AMPX code system¹ was used to calculate the PKA spectra. For elastic and inelastic resolved reactions, where angular distributions are usually available, the PKA spectra in the form of a neutron group (g_n) to a PKA group (g_R) transfer matrix were obtained from the following expression:

$$\sigma(g_n \rightarrow g_R) = \frac{\frac{1}{4\pi} \sum_{m=1}^M w_m \int_{E_{low}(\mu_m, g_R)}^{E_{high}(\mu_m, g_R)} dE \sigma(E, \mu_m) \phi(E)}{\int_{E \in g_n} dE \phi(E)} \quad (1a)$$

Table I

Neutron Group Structure (eV)

Group	Energy Range	Group	Energy Range
1	0.2000E 08 - 0.1900E 08	54	0.1228E 06 - 0.1111E 06
2	0.1900E 08 - 0.1800E 08	55	0.1111E 06 - 0.8652E 05
3	0.1800E 08 - 0.1700E 08	56	0.8652E 05 - 0.6738E 05
4	0.1700E 08 - 0.1600E 08	57	0.6738E 05 - 0.5248E 05
5	0.1600E 08 - 0.1492E 08	58	0.5248E 05 - 0.4087E 05
6	0.1492E 08 - 0.1350E 08	59	0.4087E 05 - 0.3183E 05
7	0.1350E 08 - 0.1221E 08	60	0.3183E 05 - 0.2479E 05
8	0.1221E 08 - 0.1105E 08	61	0.2479E 05 - 0.1930E 05
9	0.1105E 08 - 0.1000E 08	62	0.1930E 05 - 0.1503E 05
10	0.1000E 08 - 0.9048E 07	63	0.1503E 05 - 0.1171E 05
11	0.9048E 07 - 0.8187E 07	64	0.1171E 05 - 0.9119E 04
12	0.8187E 07 - 0.7408E 07	65	0.9119E 04 - 0.7102E 04
13	0.7408E 07 - 0.6703E 07	66	0.7102E 04 - 0.5531E 04
14	0.6703E 07 - 0.6065E 07	67	0.5531E 04 - 0.4307E 04
15	0.6065E 07 - 0.5488E 07	68	0.4307E 04 - 0.3355E 04
16	0.5488E 07 - 0.4966E 07	69	0.3355E 04 - 0.2613E 04
17	0.4966E 07 - 0.4493E 07	70	0.2613E 04 - 0.2035E 04
18	0.4493E 07 - 0.4066E 07	71	0.2035E 04 - 0.1585E 04
19	0.4066E 07 - 0.3679E 07	72	0.1585E 04 - 0.1234E 04
20	0.3679E 07 - 0.3329E 07	73	0.1234E 04 - 0.9611E 03
21	0.3329E 07 - 0.3012E 07	74	0.9611E 03 - 0.7485E 03
22	0.3012E 07 - 0.2725E 07	75	0.7485E 03 - 0.5830E 03
23	0.2725E 07 - 0.2466E 07	76	0.5830E 03 - 0.4540E 03
24	0.2466E 07 - 0.2231E 07	77	0.4540E 03 - 0.3536E 03
25	0.2231E 07 - 0.2019E 07	78	0.3536E 03 - 0.2754E 03
26	0.2019E 07 - 0.1827E 07	79	0.2754E 03 - 0.2145E 03
27	0.1827E 07 - 0.1653E 07	80	0.2145E 03 - 0.1670E 03
28	0.1653E 07 - 0.1496E 07	81	0.1670E 03 - 0.1301E 03
29	0.1496E 07 - 0.1353E 07	82	0.1301E 03 - 0.1013E 03
30	0.1353E 07 - 0.1225E 07	83	0.1013E 03 - 0.7889E 02
31	0.1225E 07 - 0.1108E 07	84	0.7889E 02 - 0.6144E 02
32	0.1108E 07 - 0.1003E 07	85	0.6144E 02 - 0.4785E 02
33	0.1003E 07 - 0.9072E 06	86	0.4785E 02 - 0.3727E 02
34	0.9072E 06 - 0.8208E 06	87	0.3727E 02 - 0.2902E 02
35	0.8208E 06 - 0.7427E 06	88	0.2902E 02 - 0.2260E 02
36	0.7427E 06 - 0.6721E 06	89	0.2260E 02 - 0.1760E 02
37	0.6721E 06 - 0.6081E 06	90	0.1760E 02 - 0.1371E 02
38	0.6081E 06 - 0.5502E 06	91	0.1371E 02 - 0.1068E 02
39	0.5502E 06 - 0.4979E 06	92	0.1068E 02 - 0.8315E 01
40	0.4979E 06 - 0.4505E 06	93	0.8315E 01 - 0.6476E 01
41	0.4505E 06 - 0.4076E 06	94	0.6476E 01 - 0.5044E 01
42	0.4076E 06 - 0.3688E 06	95	0.5044E 01 - 0.3928E 01
43	0.3688E 06 - 0.3337E 06	96	0.3928E 01 - 0.3059E 01
44	0.3337E 06 - 0.3020E 06	97	0.3059E 01 - 0.2382E 01
45	0.3020E 06 - 0.2732E 06	98	0.2382E 01 - 0.1855E 01
46	0.2732E 06 - 0.2472E 06	99	0.1855E 01 - 0.1445E 01
47	0.2472E 06 - 0.2237E 06	100	0.1445E 01 - 0.1125E 01
48	0.2237E 06 - 0.2024E 06	101	0.1125E 01 - 0.8764E 00
49	0.2024E 06 - 0.1832E 06	102	0.8764E 00 - 0.6826E 00
50	0.1832E 06 - 0.1657E 06	103	0.6826E 00 - 0.5316E 00
51	0.1657E 06 - 0.1500E 06	104	0.5316E 00 - 0.4140E 00
52	0.1500E 06 - 0.1357E 06	105	0.4140E 00 - 0.1000E -03
53	0.1357E 06 - 0.1228E 06		

Table II

Primary Knock-On Atom Energy Group Structure (eV)

Group	Energy Range	Group	Energy Range
1	0.2000E 08 - 0.1900E 08	53	0.2669E 05 - 0.2317E 05
2	0.1900E 08 - 0.1800E 08	54	0.2317E 05 - 0.2012E 05
3	0.1800E 08 - 0.1700E 08	55	0.2012E 05 - 0.1748E 05
4	0.1700E 08 - 0.1600E 08	56	0.1748E 05 - 0.1518E 05
5	0.1600E 08 - 0.1500E 08	57	0.1518E 05 - 0.1318E 05
6	0.1500E 08 - 0.1400E 08	58	0.1318E 05 - 0.1144E 05
7	0.1400E 08 - 0.1300E 08	59	0.1144E 05 - 0.9938E 04
8	0.1300E 08 - 0.1200E 08	60	0.9938E 04 - 0.8631E 04
9	0.1200E 08 - 0.1100E 08	61	0.8631E 04 - 0.7495E 04
10	0.1100E 08 - 0.1000E 08	62	0.7495E 04 - 0.6508E 04
11	0.1000E 08 - 0.8684E 07	63	0.6508E 04 - 0.5652E 04
12	0.8684E 07 - 0.7541E 07	64	0.5652E 04 - 0.4908E 04
13	0.7541E 07 - 0.6549E 07	65	0.4908E 04 - 0.4262E 04
14	0.6549E 07 - 0.5687E 07	66	0.4262E 04 - 0.3701E 04
15	0.5687E 07 - 0.4939E 07	67	0.3701E 04 - 0.3214E 04
16	0.4939E 07 - 0.4289E 07	68	0.3214E 04 - 0.2791E 04
17	0.4289E 07 - 0.3724E 07	69	0.2791E 04 - 0.2424E 04
18	0.3724E 07 - 0.3234E 07	70	0.2424E 04 - 0.2105E 04
19	0.3234E 07 - 0.2809E 07	71	0.2105E 04 - 0.1828E 04
20	0.2809E 07 - 0.2439E 07	72	0.1828E 04 - 0.1587E 04
21	0.2439E 07 - 0.2118E 07	73	0.1587E 04 - 0.1378E 04
22	0.2118E 07 - 0.1839E 07	74	0.1378E 04 - 0.1197E 04
23	0.1839E 07 - 0.1597E 07	75	0.1197E 04 - 0.1040E 04
24	0.1597E 07 - 0.1387E 07	76	0.1040E 04 - 0.9028E 03
25	0.1387E 07 - 0.1205E 07	77	0.9028E 03 - 0.7840E 03
26	0.1205E 07 - 0.1046E 07	78	0.7840E 03 - 0.6808E 03
27	0.1046E 07 - 0.9083E 06	79	0.6808E 03 - 0.5912E 03
28	0.9083E 06 - 0.7888E 06	80	0.5912E 03 - 0.5134E 03
29	0.7888E 06 - 0.6850E 06	81	0.5134E 03 - 0.4458E 03
30	0.6850E 06 - 0.5949E 06	82	0.4458E 03 - 0.3872E 03
31	0.5949E 06 - 0.5166E 06	83	0.3872E 03 - 0.3362E 03
32	0.5166E 06 - 0.4486E 06	84	0.3362E 03 - 0.2920E 03
33	0.4486E 06 - 0.3896E 06	85	0.2920E 03 - 0.2535E 03
34	0.3896E 06 - 0.3383E 06	86	0.2535E 03 - 0.2202E 03
35	0.3383E 06 - 0.2938E 06	87	0.2202E 03 - 0.1912E 03
36	0.2938E 06 - 0.2551E 06	88	0.1912E 03 - 0.1660E 03
37	0.2551E 06 - 0.2215E 06	89	0.1660E 03 - 0.1442E 03
38	0.2215E 06 - 0.1924E 06	90	0.1442E 03 - 0.1252E 03
39	0.1924E 06 - 0.1671E 06	91	0.1252E 03 - 0.1087E 03
40	0.1671E 06 - 0.1451E 06	92	0.1087E 03 - 0.9443E 02
41	0.1451E 06 - 0.1260E 06	93	0.9443E 02 - 0.8200E 02
42	0.1260E 06 - 0.1094E 06	94	0.8200E 02 - 0.7121E 02
43	0.1094E 06 - 0.9501E 05	95	0.7121E 02 - 0.6184E 02
44	0.9501E 05 - 0.8251E 05	96	0.6184E 02 - 0.5370E 02
45	0.8251E 05 - 0.7165E 05	97	0.5370E 02 - 0.4663E 02
46	0.7165E 05 - 0.6222E 05	98	0.4663E 02 - 0.4050E 02
47	0.6222E 05 - 0.5403E 05	99	0.4050E 02 - 0.3517E 02
48	0.5403E 05 - 0.4692E 05	100	0.3517E 02 - 0.3054E 02
49	0.4692E 05 - 0.4075E 05	101	0.3054E 02 - 0.2652E 02
50	0.4075E 05 - 0.3539E 05	102	0.2652E 02 - 0.2303E 02
51	0.3539E 05 - 0.3073E 05	103	0.2303E 02 - 0.2000E 02
52	0.3073E 05 - 0.2669E 05	104	0.2000E 02 - 0.0

where

$\sigma(E, \mu_m)$ = the neutron energy and angular dependence
 $(\mu_m = \cos \theta_m)$ of the cross-section data obtained
 from ENDFB/IV,

$\phi(E)$ = the flux weighting factor taken to be one of the
 following: 1/E-Maxwellian (standard), constant-
 Maxwellian (1), fission-1/E-Maxwellian (2),
 1/E σ_T -Maxwellian (3),

w_m = the angular quadrature weights, and

$\sigma(g_n \rightarrow g_R)$ = the probability (or cross section) for a neutron in
 neutron group g_n to produce a PKA in group g_R .

This expression contains no complicated physics except for the determination of the limits on the dE integral, and these, of course, are determined through momentum and energy conservation. The calculations were carried out in the following way: for an angle μ_m , the PKA energy range of T_{high} and T_{low} was determined from the expressions

$$\begin{aligned} T_h &= n_1 n_2 E_j + (n_1/n_2) E_{m,j} - 2n_1 (E_j E_{m,j})^{1/2} \mu_m \\ T_l &= n_1 n_2 E_{j+1} + (n_1/n_2) E_{m,j+1} - 2n_1 (E_{j+1} E_{m,j+1})^{1/2} \mu_m \end{aligned} \quad (1b)$$

where

$$n_1 = 1.009/(1.009 + A),$$

$$n_2 = A/(1.009 + A),$$

A = atomic number of the target,

E_j = upper neutron energy for neutron group j ,

E_{j+1} = lower neutron energy for neutron group j ,

$E_{m,j} = n_2(n_2 E_j + Q_i)$ = recoil neutron energy in the C-M system,

Q_i = excitation energy for the i th level (for elastic scattering, $Q = 0$).

If T_h and T_l fall within one PKA group, the averaged neutron cross section is assigned to that group, and the calculation proceeds to the next angle. However, as is usually the case, T_h and T_l will span several PKA groups. Therefore, starting with the energy of the first PKA group T_k above the PKA energy T_l , the neutron energy E_k (between E_j and E_{j+1}) is determined from Eq. (1b). The average of the neutron cross sections between neutron energies of E_l and E_k can then be determined and assigned to PKA group k . The above process is continued until the PKA energy T_h is reached. The calculation then proceeds to the next angle. The integration of the angular variable was replaced by a quadrature sum. For the results given, a modified 96th order Lobatto angular quadrature¹ was used. The effect of gamma rays emitted after nonelastic and inelastic collisions on the PKA spectra is small and was not included in the calculations.

For the remaining reactions given above, the PKA spectra were obtained by using a one-neutron emission model² and the secondary-neutron spectra (assumed to be applicable at all angles in the C-M system) given in ENDFB/IV. Therefore,

$$\sigma(g_n \rightarrow g_R) = \frac{\sigma(g_n)}{4\pi} \sum_{m=1}^M w_m \sum_{g_{n'} \text{ C-M}} P_1(g_{n'} \text{ C-M} \rightarrow g_R) P_2(\mu_m)$$

where

$P_1(g_n, C-M \rightarrow g_R)$ = the normalized probability for the C-M emission of a neutron with energy in group $g_n, C-M$ that can be related to a lab PKA energy in group g_R by using $T = n_1 n_2 E + (n_1/n_2) E' - 2n_1 (EE')^{1/2} \mu$ where E is the energy of the incident neutron and E' and μ are the energy and angle cosine in the C-M system of the emitted neutron,

$P_2(\mu_m)$ = the normalized probability for the C-M emission of a neutron at the angle $\mu_m (= \cos \theta_m)$, [$P_2(\mu_m) = 1$ for isotropic emission],

$\sigma(g_n)$ = the cross section for the reaction for incident neutrons in group g_n and the remaining variables are as described before.

The unnormalized angular distributions considered here are isotropic; 75% forward ($\mu_m > 0$), 25% backward ($\mu_m < 0$); and $(1 + \cos \theta)$. Note that no coupling of the distributions has been considered; i.e., it is assumed that $d^2\sigma/d\Omega dE = (d\sigma/dE)(d\sigma/d\Omega)$.

For neutron absorption reactions, such as (n,p) (n,α) , etc., the methods of Doran,³ Jenkins,² and Parkin and Golland⁴ were used.

For (n,γ) reactions, a Monte Carlo program was written. The secondary gamma-ray energies, along with their emission probabilities, were obtained from the nuclear data sheets. In contrast to previous calculations³, the incident neutron energy is used in the kinematic equations for total energy and momentum balance with the gamma rays being emitted isotropically in the rest system. Even though the capture-level

probabilities used in the calculations are for thermal-neutron energies, it was assumed that they apply for all neutron energies. It was also assumed that the gamma rays are emitted fast enough so that no interaction with other nuclei takes place until the residual nucleus is fully deexcited.

The Lindhard *et al.*,⁵ theory as used by Robinson⁶ is employed to determine that fraction of the energy of the PKA that will produce damage, i.e., further nuclear displacements. The damage energy cross section can be evaluated from the expression

$$\sigma_{i,\text{damage}}(E) = \int_{T_d}^{T_{\max}} \frac{d\sigma_i(E,T)}{dT} T_{\text{damage}} dT, \quad (2)$$

where

$\sigma_{i,\text{damage}}(E)$ = the damage energy cross section as a function of neutron energy E for the i th reaction type,

$\frac{d\sigma_i(E,T)}{dT}$ = the PKA spectrum for the i th reaction type as a function of PKA energy T and of neutron energy E ,

T_{damage} = that fraction of the energy T that will produce further nuclear displacements, and

T_d = the effective threshold energy.

T_{damage} can be calculated as follows:

$$T_{\text{damage}} = T[1 + k g(\epsilon)]^{-1}, \quad (3)$$

$$g(\epsilon) = \epsilon + 0.40244 \epsilon^{3/4} + 3.4008 \epsilon^{1/6}$$

$$k = \frac{0.0793 Z_1^{2/3} Z_2^{1/2} (A_1 + A_2)^{3/2}}{(Z_1^{2/3} + Z_2^{2/3})^{3/4} A_1^{3/2} A_2^{1/2}} \quad (4)$$

$$A = \frac{\epsilon = AT}{27.2 Z_1 Z_2 (Z_1^{2/3} + Z_2^{2/3})^{1/2} (A_1 + A_2)} \quad (eV^{-1})$$

where

A_1, Z_1 = the atomic weight and number of the PKA, respectively, and

A_2, Z_2 = like quantities for the matrix atoms.

For alloys

$$A_2 = \frac{\sum_{i=1}^N n_i A_i}{\sum_{i=1}^N n_i} \quad (5)$$

$$Z_2 = \frac{\sum_{i=1}^N n_i Z_i}{\sum_{i=1}^N n_i}$$

where

n_i = the number density of the i th element, and

A_i, Z_i = the atomic weight and number of the i th element in the alloy, respectively.

The displacement cross sections are evaluated in a similar way by replacing T_{damage} with N_d , defined as

$$\begin{array}{ll}
N_d = 0 & T < T_d \\
N_d = 1 & T_d \leq T_{\text{damage}} < 2T_d \\
N_d = \frac{\beta T_{\text{damage}}}{2 T_d} & 2 T_d \geq T_{\text{damage}}
\end{array}$$

where

$$\beta = 0.8$$

The values used for T_d are Al, 25 eV; and Fe, 40 eV.

III. RESULTS AND DISCUSSION

Some of the differential secondary neutron energy spectra for Fe, (n,2n), and (n,n' unresolved reactions), used in the calculations are shown in Fig. 1. The histograms represent ENDFB/IV data. The (n,2n) ENDFB/IV data can be closely approximated by an evaporation-type spectrum. In contrast to this, the (n,n' unresolved) ENDFB/IV data cannot.

The sensitivity of the Fe(n,2n) and (n,n' unresolved) Lab PKA spectra resulting from incident neutrons in Group 1 to different C-M secondary neutron energy and angular distributions are shown in Figs. 2 and 3, respectively. The symbols o, x, and · represent midpoint histogram values. As can be seen, a rather diverse set of distributions can be obtained depending upon the initial conditions.

The sensitivity of the Fe(n,2n) and (n,n' unresolved) DPA cross sections as a function of incident neutron energy to different secondary neutron angular distributions and energy spectra are given in Tables 3 and 4. Rather large (~ 15%) variations are evident. Considering that rather diverse secondary neutron energy spectra, but the same angular distributions are used to obtain the results in columns 2 and 5

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Fig. 1. Differential Secondary Neutron Energy Spectra for Fe.

Fig. 2. Sensitivity of the Fe (n,2n) Lab PKA spectrum resulting from incident neutrons in group 1 to different center-of-mass secondary neutron energy and angular distributions. The symbols o, x, and $\bullet$ represent the midpoint histogram values. The center-of-mass secondary neutron energy spectrum, unless otherwise specified, has been taken from ENDFB/IV.

Fig. 3. Sensitivity of the Fe(n,n' unresolved) Lab PKA spectrum resulting from incident neutrons in group 1 to different center-of-mass secondary neutron energy and angular distributions. The symbols o, x, and • represent the midpoint histogram values. The center-of-mass secondary neutron energy spectrum, unless otherwise specified, has been taken from ENDFB/IV.

Table III
Sensitivity of the Fe(n,2n) DPA Cross Sections
to Different Secondary Neutron Angular
Distributions and Energy Spectra^a

Neutron Group Number	DPA Cross Sections (barns)			
	$\frac{d\sigma}{d\Omega} = 1^b$	$\frac{d\sigma}{d\Omega} = 2$	$\frac{d\sigma}{d\Omega} = 3$	$\frac{d\sigma}{d\Omega} = 1$
	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = B$
1	1095	968	925	1118
2	1084	959	917	1108
3	1060	938	897	1084
4	1011	895	856	1032
5	898	798	764	915
6	631	566	544	642
7	230	210	204	233
8	13.6	12.9	12.7	13.8

^aCalculated using the one-neutron emission model approximation.

^b1 = constant; 2 = 75% forward, 25% backward;

3 = $1 + \cos \theta$.

A = ENDFB/IV; B = $Ee^{-E/T}$ where $T(\text{eV}) = 3.22 \times 10^3 \sqrt{E'/A}$, A is the atomic number of the target, E' is the incident neutron energy in eV, and E is the energy of the emitted neutron.

All energy and angular distributions are for the center of mass system.

Table IV
Sensitivity of the Fe(n,n' unresolved) DPA Cross Sections
to Different Neutron Angular
Distributions and Energy Spectra

Neutron Group Number	DPA Cross Sections (barns)			
	$\frac{d\sigma}{d\Omega} = 1^a$	$\frac{d\sigma}{d\Omega} = 2$	$\frac{d\sigma}{d\Omega} = 3$	$\frac{d\sigma}{d\Omega} = 1$
	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = A$	$\frac{d\sigma}{dE} = B$
1	460	376	346	394
2	472	385	356	410
3	504	412	381	444
4	565	463	428	506
5	689	566	524	631
6	955	791	736	906
7	1309	1099	1028	1289
8	1485	1259	1183	1492
9	1395	1181	1109	1410
10	1261	1066	1000	1277
11	1112	939	881	1129
12	936	792	744	956
13	730	622	585	751
14	512	442	419	533
15	308	271	259	323
16	144	130	126	152
17	27.5	25.4	24.8	28.3

^a1 = constant; 2 = 75% forward, 25% backward; 3 = $1 + \cos \theta$.

A = ENDFB/IV; B = $Ee^{-E/T}$, where $T(\text{eV}) = 3.22 \times 10^3 \sqrt{E'}/A$,

A is the atomic number of the target,

E' is the incident neutron energy in eV, and

E is the energy of the emitted neutron.

All energy and angular distributions are for the center of mass system.

in Table 4, the results are remarkably similar. The reason for this is due to the nonlinear dependence of the damage energy, that is, as the PKA energy increases the fraction of deposited energy that contributes to damage decreases.

Given in Tables 5-8 are the percent changes in the total and various partial groups averaged damage energy cross sections for Al due to different in-group weighting schemes. The type of weighting has been defined in Sec. II. Overall, except for weighting 3, there is little effect. The damage energy cross sections given in the tables can, to a good approximation, be converted to DPA cross sections by multiplying by 0.8 and dividing by two times the effective displacement threshold. In Table 9 the changes are investigated by averaging over several types of "typical" neutron environments.

Even though Al and Fe have been used to demonstrate the sensitivity of various neutronic parameters, the effects seen here can be expected for other elements.

Table V

Changes in the Total Group Averaged Damage Energy Cross Sections
for Al Due to Different In-Group Weighting Schemes

Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting		
		1	2	3
		% Change		
1	0.18063E 06	-0.0	0.0	0.0
2	0.18095E 06	-0.0	0.0	-0.0
3	0.18107E 06	-0.0	0.0	-0.0
4	0.18050E 06	-0.0	-0.0	0.0
5	0.17817E 06	-0.0	-0.1	0.0
6	0.17364E 06	-0.0	-0.3	-0.0
7	0.16842E 06	0.0	-0.2	-0.0
8	0.16363E 06	0.0	-0.2	-0.0
9	0.15436E 06	0.0	-0.3	-0.0
10	0.14827E 06	0.0	-0.3	-0.0
11	0.14410E 06	-0.0	0.0	-0.0
12	0.14494E 06	-0.0	0.0	-0.1
13	0.14244E 06	-0.0	0.2	-0.1
14	0.13938E 06	0.0	-0.1	-0.2
15	0.13784E 06	0.1	-0.3	-0.2
16	0.13598E 06	-0.0	0.1	-0.2
17	0.13040E 06	0.0	-0.1	-0.3
18	0.12795E 06	-0.0	0.1	-0.4
19	0.12986E 06	0.0	-0.0	-0.2
20	0.12934E 06	0.0	-0.0	-0.6
21	0.12399E 06	0.0	-0.0	-0.5
22	0.12314E 06	0.0	-0.0	-1.1
23	0.13008E 06	-0.1	0.1	-1.7
24	0.10796E 06	-0.3	0.1	-2.0
25	0.12519E 06	-0.0	0.0	-0.8
26	0.11324E 06	0.3	-0.0	-2.9
27	0.10767E 06	-0.1	-0.0	-2.1
28	0.10196E 06	-0.1	-0.0	-1.5
29	0.98868E 05	0.1	0.0	-2.0
30	0.87297E 05	-0.2	-0.1	-1.2
31	0.10511E 06	-0.0	-0.0	-2.3
32	0.75860E 05	-0.0	-0.0	-2.8
33	0.70694E 05	0.0	0.0	-1.9
34	0.92101E 05	-0.3	-0.2	-2.4
35	0.10047E 06	0.3	0.3	-6.3
36	0.64877E 05	0.1	0.1	-1.2
37	0.78636E 05	0.3	0.3	-6.4
38	0.70326E 05	-0.2	-0.2	-4.8
39	0.70414E 05	0.3	0.3	-4.2
40	0.65868E 05	1.4	-0.1	-3.2
41	0.75467E 05	0.0	0.0	-1.0
42	0.48232E 05	-0.5	-0.6	-21.6
43	0.43452E 05	0.8	1.0	-10.9
44	0.51097E 05	-0.3	-0.4	-2.9
45	0.56882E 05	0.3	0.4	-11.7
46	0.22030E 05	-0.6	-0.8	-9.2
47	0.33601E 05	-0.5	-0.6	-2.8
48	0.50636E 05	-0.3	-0.4	-1.4
49	0.31340E 05	0.5	0.7	-3.2
50	0.33181E 05	-0.3	-0.5	-1.5
51	0.67507E 05	-0.2	-0.2	-4.9
52	0.59486E 05	0.8	1.1	-10.3
53	0.12553E 05	0.5	0.8	-9.9

Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting		
		1	2	3
		% Change		
54	0.25899E 05	1.2	1.6	-22.2
55	0.38631E 05	-4.2	-6.0	-33.4
56	0.22911E 05	6.1	8.9	-48.2
57	0.54539E 04	-0.4	0.0	-0.6
58	0.40580E 04	-1.4	0.0	-5.0
59	0.31659E 05	-0.9	0.0	-27.4
60	0.20263E 04	4.8	0.0	-29.6
61	0.82759E 03	-0.3	0.0	-0.4
62	0.85181E 03	0.1	0.0	0.1
63	0.85774E 03	0.2	0.0	-1.9
64	0.72466E 03	0.5	0.0	-1.1
65	0.59340E 03	0.4	0.0	0.1
66	0.94501E 03	-2.8	0.0	-42.9
67	0.37541E 03	0.7	0.0	-0.3
68	0.29105E 03	0.5	0.0	0.0
69	0.23106E 03	0.5	0.0	-0.0
70	0.18135E 03	0.5	0.0	0.1
71	0.14195E 03	0.5	0.0	-0.2
72	0.11078E 03	0.5	0.0	0.1
73	0.86032E 02	0.5	0.0	-0.1
74	0.66598E 02	0.5	0.0	-1.3
75	0.51120E 02	0.5	0.0	-0.3
76	0.38585E 02	0.5	0.0	0.1
77	0.28175E 02	0.7	0.0	0.3
78	0.19330E 02	0.9	0.0	0.5
79	0.11529E 02	1.3	0.0	1.0
80	0.48263E 01	2.1	0.0	1.7
81	0.29460E 01	-0.2	0.0	-0.1
82	0.31989E 01	-0.2	0.0	-0.1
83	0.35706E 01	-0.3	0.0	0.9
84	0.40173E 01	-0.3	0.0	0.2
85	0.45675E 01	-0.3	0.0	0.0
86	0.51193E 01	-0.3	0.0	-0.0
87	0.57842E 01	-0.3	0.0	-0.1
88	0.65191E 01	-0.3	0.0	-0.1
89	0.73234E 01	-0.3	0.0	-0.2
90	0.82545E 01	-0.3	0.0	-0.2
91	0.94475E 01	-0.3	0.0	-0.2
92	0.10593E 02	-0.3	0.0	1.0
93	0.12001E 02	-0.3	-0.0	0.3
94	0.13595E 02	-0.3	0.0	0.1
95	0.15401E 02	-0.3	0.0	-0.1
96	0.17446E 02	-0.3	0.0	-0.1
97	0.19763E 02	-0.3	0.0	-0.1
98	0.22389E 02	-0.3	0.0	-0.2
99	0.25362E 02	-0.3	0.0	-0.2
100	0.26731E 02	-0.3	0.0	-0.2
101	0.32552E 02	-0.3	-0.0	0.7
102	0.36892E 02	-0.3	0.0	0.4
103	0.41816E 02	-0.3	0.0	0.1
104	0.47398E 02	-0.3	0.0	-0.1
105	0.16341E 03	-8.3	0.0	-5.9

Table VI

Changes in the Elastic Group Averaged Damage Energy Cross Sections
for Al Due to Different In-Group Weighting Schemes

Neutron Group	Standard Damage Energy $\sigma(\text{b-eV})$	Type of Weighting		
		1	2	3
		% Change		
1	0.51100E 05	0.0	-0.3	-0.0
2	0.48319E 05	0.0	-0.4	-0.0
3	0.45079E 05	0.0	-0.4	0.0
4	0.42467E 05	0.0	-0.3	-0.0
5	0.40105E 05	0.0	-0.2	-0.0
6	0.39434E 05	0.0	-0.2	-0.0
7	0.38151E 05	0.0	-0.1	0.0
8	0.38164E 05	-0.0	0.2	-0.0
9	0.37468E 05	-0.0	0.2	-0.0
10	0.40564E 05	0.0	-0.2	-0.0
11	0.43866E 05	-0.2	0.8	-0.2
12	0.50129E 05	-0.1	0.4	-0.2
13	0.56503E 05	-0.3	1.1	-0.5
14	0.63887E 05	-0.0	0.2	-0.4
15	0.66599E 05	0.1	-0.4	-0.3
16	0.70816E 05	-0.1	0.3	-0.5
17	0.72068E 05	-0.0	0.0	-0.5
18	0.74564E 05	-0.1	0.3	-0.7
19	0.81919E 05	-0.0	0.1	-0.3
20	0.87299E 05	-0.0	0.0	-0.9
21	0.88449E 05	-0.0	0.1	-0.7
22	0.97407E 05	-0.1	0.0	-1.5
23	0.10955E 06	-0.2	0.1	-2.1
24	0.91550E 05	-0.4	0.1	-2.5
25	0.11059E 06	-0.0	0.0	-1.0
26	0.99635E 05	0.4	-0.0	-3.3
27	0.95599E 05	-0.1	-0.0	-2.4
28	0.92416E 05	-0.2	-0.0	-1.7
29	0.91541E 05	0.1	0.0	-2.2
30	0.81575E 05	-0.2	-0.1	-1.3
31	0.10135E 06	-0.0	-0.0	-2.4
32	0.74968E 05	-0.1	-0.0	-2.9
33	0.70654E 05	0.0	0.0	-1.9
34	0.92096E 05	-0.3	-0.2	-2.4
35	0.10047E 06	0.3	0.3	-6.3
36	0.64872E 05	0.1	0.1	-1.2
37	0.78631E 05	0.3	0.3	-6.4
38	0.70321E 05	-0.2	-0.2	-4.8
39	0.70409E 05	0.3	0.3	-4.2
40	0.65862E 05	1.4	-0.1	-3.2
41	0.75478E 05	0.0	0.0	-1.0
42	0.48223E 05	-0.5	-0.6	-21.6
43	0.43444E 05	0.8	1.0	-10.9
44	0.51041E 05	-0.3	-0.4	-2.9
45	0.56676E 05	0.3	0.4	-11.7
46	0.22025E 05	-0.6	-0.8	-9.2
47	0.33596E 05	-0.5	-0.6	-2.8
48	0.50631E 05	-0.3	-0.4	-1.4
49	0.31334E 05	0.5	0.7	-3.2
50	0.33174E 05	-0.3	-0.5	-1.5
51	0.67496E 05	-0.2	-0.2	-4.9
52	0.59474E 05	0.8	1.1	-10.3
53	0.12544E 05	0.5	0.8	-4.9

Neutron Group	Standard Damage Energy $\sigma(\text{b-eV})$	Type of Weighting		
		1	2	3
		% Change		
54	0.25887E 05	1.2	1.6	-22.2
55	0.38621E 05	-4.2	-6.0	-33.4
56	0.22906E 05	6.1	8.9	-48.2
57	0.54517E 04	-0.3	0.0	-0.6
58	0.80524E 04	-1.4	0.0	-5.0
59	0.31628E 05	-0.9	0.0	-27.4
60	0.20200E 04	4.8	0.0	-29.7
61	0.62329E 03	-0.3	0.0	-0.4
62	0.84769E 03	0.1	0.0	0.1
63	0.85386E 03	0.2	0.0	-1.9
64	0.72058E 03	0.5	0.0	-1.1
65	0.58856E 03	0.4	0.0	0.1
66	0.90517E 03	-2.7	0.0	-41.8
67	0.36957E 03	0.7	0.0	-0.3
68	0.28657E 03	0.5	0.0	0.1
69	0.22748E 03	0.5	0.0	-0.0
70	0.17843E 03	0.5	0.0	0.1
71	0.13952E 03	0.5	0.0	-0.2
72	0.10870E 03	0.5	0.0	0.1
73	0.84291E 02	0.5	0.0	-0.1
74	0.64836E 02	0.5	0.0	-1.4
75	0.49234E 02	0.5	0.0	-0.3
76	0.36537E 02	0.6	0.0	0.1
77	0.25989E 02	0.7	0.0	0.3
78	0.16981E 02	1.0	0.0	0.6
79	0.89790E 01	1.8	0.0	1.3
80	0.20905E 01	5.0	0.0	4.0
81	0.14813E 02	12.7	0.0	10.6

Table VII

Changes in Various Partial Group Averaged Damage Energy Cross Sections
for Al Due to Different In-Group Weighting Schemes

Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting		
		1	2	3
% Change				
INELASTIC. RESOLVED				
1	0.67916E 05	-0.0	0.4	0.0
2	0.72246E 05	-0.1	0.4	0.0
3	0.78028E 05	-0.1	0.4	0.0
4	0.84401E 05	-0.1	0.4	0.0
5	0.92011E 05	-0.1	0.5	0.0
6	0.10052E 06	-0.1	0.5	-0.0
7	0.10217E 06	0.0	-0.2	-0.0
8	0.99481E 05	0.0	-0.2	-0.0
9	0.93609E 05	0.1	-0.5	-0.0
10	0.87583E 05	0.0	-0.3	-0.0
11	0.84205E 05	0.0	-0.2	0.0
12	0.83067E 05	-0.0	-0.0	-0.0
13	0.78191E 05	0.1	-0.4	0.1
14	0.71208E 05	0.1	-0.2	0.0
15	0.67692E 05	0.0	-0.2	-0.1
16	0.62793E 05	0.1	-0.2	0.1
17	0.56844E 05	0.1	-0.2	0.0
18	0.52527E 05	0.1	-0.1	0.1
19	0.47288E 05	0.1	-0.2	0.0
20	0.42039E 05	0.1	-0.1	0.0
21	0.35539E 05	0.2	-0.2	-0.0
22	0.25728E 05	0.3	-0.2	0.2
23	0.20531E 05	0.2	-0.1	0.4
24	0.16405E 05	0.2	-0.1	0.8
25	0.14601E 05	0.1	0.0	0.0
26	0.13596E 05	0.1	-0.0	-0.3
27	0.12066E 05	0.1	-0.0	0.2
28	0.95383E 04	0.3	0.0	0.6
29	0.73226E 04	0.2	0.1	-0.1
30	0.57189E 04	0.2	0.1	0.7
31	0.37623E 04	0.5	0.3	0.4
32	0.88904E 03	2.4	1.7	2.5
33	0.37024E 02	2.0	1.5	2.2
34	0.55153E 00	4.0	3.2	26.1

(n,2n)

1	0.45555E 05	0.0	-0.4	-0.0
2	0.41742E 05	0.0	-0.5	-0.0
3	0.36748E 05	0.1	-0.8	-0.0
4	0.29820E 05	0.1	-1.4	-0.0
5	0.19783E 05	0.4	-3.5	-0.1
6	0.54067E 04	1.2	-20.1	-0.2

(n,T)

1	0.18895E 04	0.1	-0.9	-0.0
2	0.15105E 04	0.1	-1.6	-0.0
3	0.10380E 04	0.2	-2.5	-0.0
4	0.58874E 03	0.4	-3.8	-0.1
5	0.22393E 03	0.8	-8.0	-0.1
6	0.16953E 02	3.4	-29.2	-0.2

Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting		
		1	2	3
		(n,p)		
		% Change		
1	0.47969E 04	-0.1	1.2	0.0
2	0.57282E 04	-0.1	0.9	-0.1
3	0.66085E 04	-0.1	0.8	-0.1
4	0.74115E 04	-0.1	0.7	0.0
5	0.82279E 04	-0.1	0.7	0.1
6	0.91625E 04	-0.2	1.0	-0.0
7	0.10004E 05	-0.1	0.7	-0.0
8	0.10702E 05	-0.1	0.5	-0.1
9	0.11104E 05	-0.0	0.2	0.0
10	0.10735E 05	0.0	-0.3	-0.0
11	0.94950E 04	0.1	-0.5	0.0
12	0.79701E 04	0.1	-0.6	0.1
13	0.63995E 04	0.1	-0.7	0.2
14	0.48726E 04	0.2	-0.8	0.1
15	0.35448E 04	0.2	-0.8	-0.2
16	0.23622E 04	0.4	-1.0	0.4
17	0.14838E 04	-0.1	0.2	-0.0
18	0.85081E 03	0.8	-1.4	1.0
19	0.65036E 03	-0.3	0.3	-0.2

(n,D)

1	0.33266E 04	-0.0	0.9	0.0
2	0.36939E 04	-0.1	0.6	-0.1
3	0.39970E 04	-0.0	0.4	-0.0
4	0.41436E 04	-0.0	0.2	0.0
5	0.40819E 04	0.0	-0.2	-0.0
6	0.34506E 04	0.3	-2.1	0.0
7	0.22956E 04	0.4	-3.3	-0.0
8	0.12184E 04	0.6	-4.6	0.1
9	0.45135E 03	1.1	-6.8	-0.0
10	0.63446E 02	2.3	-13.2	-0.6

(n, α)

1	0.59107E 04	-0.2	1.5	0.0
2	0.75907E 04	-0.1	1.4	0.0
3	0.94682E 04	-0.1	1.2	0.0
4	0.11576E 05	-0.1	1.1	0.0
5	0.13663E 05	-0.1	1.0	0.1
6	0.15482E 05	-0.1	0.8	-0.0
7	0.15747E 05	0.0	-0.3	-0.0
8	0.14021E 05	0.1	-0.8	0.0
9	0.11691E 05	0.2	-1.0	0.0
10	0.92956E 04	0.2	-1.2	-0.1
11	0.65154E 04	0.4	-1.8	0.2
12	0.37520E 04	0.6	-2.3	0.3
13	0.17337E 04	0.8	-3.0	1.1

Table VIII

Changes in the (n, γ) Group Averaged Damage Energy Cross Sections
for Al Due to Different In-Group Weighting Schemes

Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting			Neutron Group	Standard Damage Energy σ (b-eV)	Type of Weighting		
		1	2	3			1	2	3
		% Change					% Change		
1	0.13245E 03	0.1	-0.6	-0.0	54	0.12663E 02	1.3	1.8	-18.9
2	0.11796E 03	0.1	-0.6	-0.0	55	0.93439E 01	-1.6	-2.4	-14.6
3	0.10352E 03	0.1	-0.7	-0.0	56	0.52868E 01	3.7	5.4	-30.7
4	0.89597E 02	0.1	-0.8	-0.0	57	0.22823E 01	-0.9	0.0	-1.4
5	0.75248E 02	0.1	-1.0	-0.0	58	0.56515E 01	-2.0	0.0	-6.8
6	0.62780E 02	0.2	-1.1	0.0	59	0.31388E 02	-2.3	0.0	-37.8
7	0.52105E 02	0.2	-1.0	-0.0	60	0.63119E 01	1.3	0.0	-7.6
8	0.42883E 02	0.2	-1.0	0.0	61	0.42975E 01	0.1	0.0	0.2
9	0.34975E 02	0.2	-0.9	0.0	62	0.39198E 01	0.0	0.0	0.0
10	0.28108E 02	0.2	-0.8	0.0	63	0.38803E 01	-0.2	0.0	-0.1
11	0.22298E 02	0.2	-0.8	0.1	64	0.40842E 01	-0.3	0.0	-0.0
12	0.17288E 02	0.2	-0.7	0.1	65	0.48380E 01	-0.7	0.0	-0.1
13	0.13092E 02	0.2	-0.8	0.3	66	0.39844E 02	-5.1	0.0	-69.1
14	0.95809E 01	0.3	-0.8	0.1	67	0.58476E 01	0.4	0.0	-0.3
15	0.66568E 01	0.3	-0.9	-0.2	68	0.44808E 01	0.3	0.0	-0.1
16	0.42113E 01	0.4	-1.0	0.4	69	0.35759E 01	0.4	0.0	-0.1
17	0.35094E 01	-0.2	0.3	-0.0	70	0.29265E 01	0.4	0.0	-0.0
18	0.40286E 01	-0.1	0.2	-0.1	71	0.24309E 01	0.3	0.0	-0.2
19	0.44189E 01	-0.1	0.2	-0.0	72	0.20805E 01	0.3	0.0	-0.0
20	0.47799E 01	-0.0	0.2	0.1	73	0.17412E 01	0.2	0.0	-0.1
21	0.50652E 01	-0.1	0.1	0.0	74	0.17619E 01	-0.2	0.0	0.4
22	0.52899E 01	-0.1	0.1	-0.1	75	0.18861E 01	-0.2	0.0	0.0
23	0.55837E 01	-0.1	0.0	-0.3	76	0.20476E 01	-0.2	0.0	-0.0
24	0.57290E 01	-0.1	-0.0	-0.3	77	0.21855E 01	-0.2	0.0	-0.1
25	0.57405E 01	-0.1	-0.0	-0.0	78	0.23486E 01	-0.2	0.0	-0.1
26	0.55055E 01	0.0	0.0	-0.1	79	0.25504E 01	-0.2	0.0	-0.1
27	0.49847E 01	0.0	-0.0	0.0	80	0.27358E 01	-0.2	0.0	-0.1
28	0.44988E 01	0.0	-0.0	0.0	81	0.29446E 01	-0.2	0.0	-0.1
29	0.40615E 01	0.0	-0.0	-0.0	82	0.31989E 01	-0.2	0.0	-0.1
30	0.37127E 01	0.0	-0.1	0.0	83	0.35706E 01	-0.3	0.0	0.9
31	0.33463E 01	0.0	-0.0	0.0	84	0.40173E 01	-0.3	0.0	0.2
32	0.30060E 01	0.0	-0.0	0.0	85	0.45675E 01	-0.3	0.0	0.0
33	0.34910E 01	-0.4	-0.3	-0.2	86	0.51193E 01	-0.3	0.0	-0.0
34	0.45417E 01	-0.3	-0.2	-1.0	87	0.57842E 01	-0.3	0.0	-0.1
35	0.51334E 01	-0.1	-0.1	0.3	88	0.65191E 01	-0.3	0.0	-0.1
36	0.51663E 01	-0.1	-0.1	0.0	89	0.73234E 01	-0.3	0.0	-0.2
37	0.52170E 01	-0.1	-0.1	0.2	90	0.82545E 01	-0.3	0.0	-0.2
38	0.50948E 01	-0.1	-0.1	-0.3	91	0.93475E 01	-0.3	0.0	-0.2
39	0.49291E 01	-0.1	-0.1	0.0	92	0.10593E 02	-0.3	0.0	1.0
40	0.65399E 01	-0.5	-0.5	-1.0	93	0.12001E 02	-0.3	-0.0	0.3
41	0.89040E 01	-0.3	-0.4	0.0	94	0.13595E 02	-0.3	0.0	0.1
42	0.95762E 01	0.1	0.1	0.2	95	0.15401E 02	-0.3	0.0	-0.1
43	0.78300E 01	0.1	0.1	-0.4	96	0.17446E 02	-0.3	0.0	-0.1
44	0.64453E 01	0.1	0.1	0.5	97	0.19763E 02	-0.3	0.0	-0.1
45	0.56786E 01	-0.1	-0.1	0.4	98	0.22389E 02	-0.3	0.0	-0.2
46	0.55923E 01	-0.1	-0.2	-0.4	99	0.25362E 02	-0.3	0.0	-0.2
47	0.53660E 01	0.0	-0.0	-0.2	100	0.28731E 02	-0.3	0.0	0.2
48	0.52256E 01	0.0	-0.0	-0.1	101	0.32552E 02	-0.3	-0.0	0.7
49	0.54096E 01	-0.3	-0.4	1.0	102	0.36892E 02	-0.3	0.0	0.4
50	0.66881E 01	-0.3	-0.4	-1.3	103	0.41816E 02	-0.3	0.0	0.1
51	0.11033E 02	-0.5	-0.7	-2.4	104	0.47398E 02	-0.3	0.0	-0.1
52	0.12257E 02	0.1	0.1	-1.2	105	0.16341E 03	-8.3	0.0	-5.9
53	0.91559E 01	-0.0	-0.1	-3.6					

Table IX

Changes in Deposited Damage Energy in Al Due to the
Different In-Group Weighting Schemes

Type of Neutron Environment	Type of Weighting		
	1	2	3
	% change relative to deposited damage energy obtained using standard cross section set		
First Wall CTR	0.0	0.0	-2.6
HFIR PTP (z = 0.)	0.0	0.0	-3.5
HFIR PTP (z = 20. cm)	0.0	0.0	-3.8
Fast Reactor	0.0	0.0	-6.5

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