ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60440

DOSIMETRY FOR
RADIATION DAMAGE STUDIES

by

A. D. Rossin

Metallurgy Division

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ABSTRACT

A method is presented for reporting fast-neutron exposure in a meaningful and unambiguous fashion. The steps involve determination of spectrum shape, absolute magnitude, an energy weighting for the neutrons, and a unit for reporting exposure. Various methods for performing the procedure are described, and the reasoning behind the approach is explained.

1. INTRODUCTION

This report presents a procedure for determining and reporting fast-neutron exposure. The following steps are involved:

1. Determination of the shape of the neutron energy spectrum.

2. Absolute measurement of the neutron flux rate by activation detectors to fix the magnitude of the spectrum.

3. Choice of an appropriate energy weighting for the neutrons (i.e., relative effectiveness in producing damage as a function of energy).

4. Determination of time-integrated reactor power during the period of the irradiation, multiplying this by the flux rate, and reporting the exposure thus obtained in meaningful and unambiguous units.

Each step must be accomplished to arrive at a useful result. There are methods for accomplishing each step; however, the techniques available are all in the process of development and can certainly be improved. Each method depends on quantitative nuclear data, which are subject to experimental uncertainties. Current knowledge contains numerous gaps where assumptions must serve in place of factual data, but the use of assumptions does not change the logic of the over-all approach.

In some instances the opinions of different investigators differ on the choice of assumptions. This is only natural, considering the state of
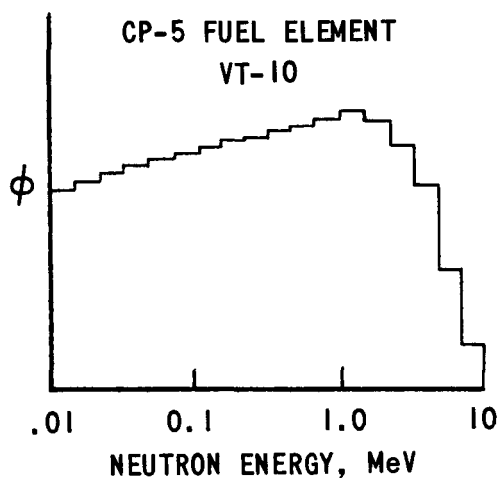
present knowledge, but it makes it essential for a report to state clearly what assumptions are employed. If new experimental data are obtained that fill some of these gaps, they can be applied to improve the numerical accuracy of the result. It is expected that these gaps will be filled, so the procedure is written to permit direct substitution of numbers without changing the method itself.

Procedures for performing the four steps are outlined. For each step, the investigator should choose the alternative which best suits his purpose until experience or mutual agreement dictates one particular choice. The logic can be applied in a parallel manner for any alternate method.

The availability of large computer programs and experience with fast-reactor physics analysis have made multigroup treatment of fast-neutron problems quite feasible. Multigroup methods are straightforward and easy to explain and are described fully in the procedure. Appendix A presents a complete procedure using multigroup methods in "cook-book" fashion.

2. SHAPE OF THE NEUTRON ENERGY SPECTRUM

The distribution of the neutron population will be defined as a function of neutron energy. For typical radiation damage studies, the energy range from 10 MeV to 10 keV should give sufficient coverage. This spectral shape is to be determined at the location of interest, that is, where dosimeter foils are mounted, where irradiation specimens are exposed, or where a component must serve in a neutron environment.

The spectrum can be shown as a histogram that gives the number of neutrons having energies within each energy interval. The group fluxes from the histogram may be presented in the form of a table. In the example shown at the left, the spectrum is broken up into small energy bands having equal width on a log plot. These numbers give a spectral shape. They have not yet been normalized to a particular reactor power level. A smooth curve is sometimes used in place of the histogram, and some techniques give an analytical expression that fits this smooth curve. A pair of exponential terms with adjustable coefficients can work satisfactorily.

CP-5 FUEL ELEMENT
VT-10

ϕ

.01   0.1   1.0   10
NEUTRON ENERGY, MeV

Group	E_L, MeV	ϕ_i, n/cm²-sec
1	7.788	0.062
2	6.065	0.103
3	4.724	0.348
4	3.679	0.627
.	.	.
.	.	.
.	.	.
20	0.067	0.226

Spectral shape may be determined by any one of the following methods:

a. <u>Assumption</u>: There may be good justification for believing that the spectrum in question fits a well-known form, i.e., the fission spectrum, fission spectrum with a 1/E low-energy component, fission spectrum attenuated through a known thickness of a given material (based on calculations found in the literature), etc. A specific shape may be assumed, provided of course, that a reasonable basis exists for the choice.

b. <u>Multigroup Calculation</u>: The fast-neutron energy range is broken into a number of intervals. All neutrons having energies within the limits of an interval are treated as one group. Cross-section sets are formed by averaging microscopic data over each energy interval to give an effective value for neutrons in each group. The computer program determines the migration and slowing down of neutrons throughout the system by diffusion theory, transport theory, Monte Carlo methods, etc. Examples of codes are shown in Sec. 10.2 of ANL-5800.[1] The solution gives the number of neutrons in each energy group at any point in the reactor system. The numbers in the table on the previous page are examples.

c. <u>Analysis of Activation Data</u>: By exposing a set of foils to the neutron flux in question, a set of activation rates for various nuclides can be measured. If the cross sections for these fast neutron reactions are known, a set of integral equations can be constructed (in theory) and solved to find the neutron spectrum. In practice, this technique has produced limited results. This is because experimental uncertainties, lack of knowledge of cross sections, complex shapes of cross sections, and an inadequate spread of threshold energies for activation of various foils combine to make the solution of the equations undependable (if they can indeed give a solution at all).

Some recent innovations may increase the usefulness of this method. They depend first on abandoning the familiar concepts of threshold energy and average cross section. Few cross sections look anything like a step function. By definition, the choice of a threshold energy value directly determines the "average" or "effective" cross-section value. Consequently, there is little numerical agreement in the literature, since choices which look reasonable for one system seem to fail when tested in other spectra.

Methods based on the use of foil data depend on assumptions about the spectrum shape. W. McIlroy* tabulated multigroup cross-section data for various detectors. Choosing a wide variety of neutron spectra and presenting them in multigroup form, he calculated ratios of activation rates between various pairs of detector nuclides. Comparing data taken from a set of experimental foil irradiations with the library of ratios he calculated,

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*Illinois Institute of Technology Research Institute, Chicago, Ill.

a "most likely spectrum type" can be chosen. Then a computer code can use the chosen spectrum as input data, and apply the foil data to it, attempting to improve the accuracy of the spectral shape.

J. P. Genthon* approximated the fast-neutron spectrum by a pair of exponential terms with adjustable coefficients. The activation cross sections were also approximated by a pair of exponentials, and foil results provided the best possible coefficients to represent the spectrum. Again, the choice of an approximate spectral shape to begin with kept one bad piece of foil data from giving an unrealistic spectral shape as a solution.

Other experience with these methods is described in the Proceedings of the IAEA Symposium on Neutron Dosimetry.[2] See, in particular, papers by Bresesti et al., Dierckx, and Moteff.

In practice, the difference between these methods and direct calculation (method b above) is in the choice of which data will finally determine the spectrum shape. Foils can be used with multigroup calculations to verify the calculated shape as well as to determine the absolute magnitude of the flux. The method is described in Section 3 below.

3. MAGNITUDE OF THE FLUX

Take the spectrum shape determined by one of the methods of Section 2, and put numbers on the vertical axis to correspond to a specific power level of the reactor. The sketch below (left) is the spectrum shape or the unnormalized spectrum. The sketch below (right) is quantitative. It gives the flux rate at a particular location when the reactor is operating at 1 MW.
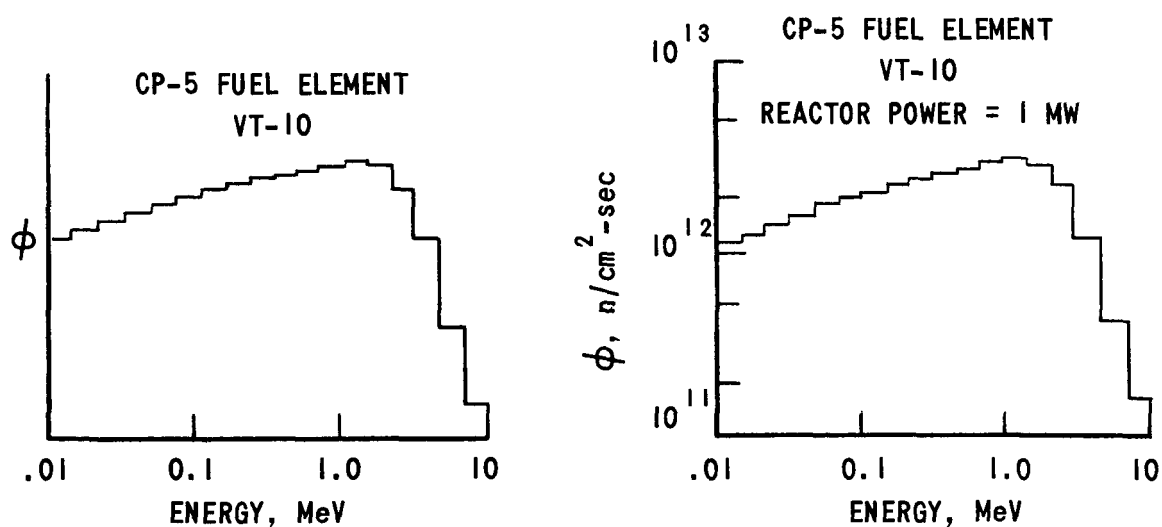

*C.E.A., Saclay, France.

Foils do not measure neutron flux. The only information obtainable directly from a foil is the activation rate at a particular power level. The arithmetic involved concerns the weight of the sample, corrections for decay with time, determination of reactor power and operating time, counting efficiency, etc.

The equation for the activation rate of a foil is

$$A \; (\text{activations}/\text{second}/\text{atom}) \; = \; \int_0^\infty \sigma_{act} \, (E) \; \phi(E) \; dE.$$

If the cross section of a nuclide for a particular fast-neutron reaction is known, and the neutron spectrum is known, this integral may be evaluated.
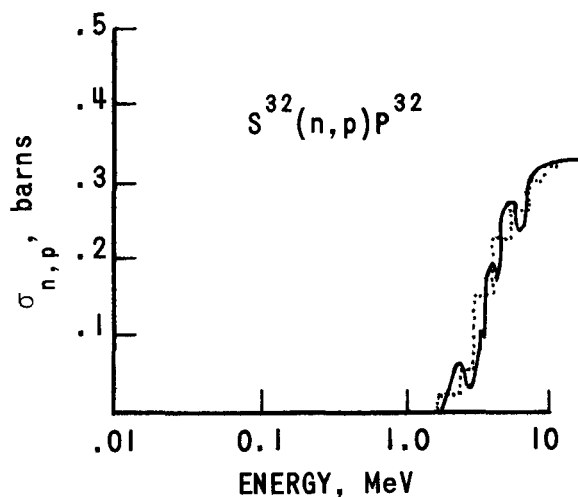
The multigroup notation is easily applied, approximating the actual cross section (solid line) by group values (dotted line) in this sketch. The table below is used to compute the activation rate for the reaction S^{32} (n,p) P^{32}. The integral is approximated by the summation

$$A \; = \; \sum_i \sigma_i(S) \; \phi_i.$$

But this activation rate was obtained from an unnormalized spectrum. A normalization factor F must be defined so that $\phi_{true} = F\phi_{not \; normalized}$. To do this, the calculated A is compared with a measurement. For example, in a sulfur foil,

$$A_{calc} \; (P^{32}) \; = \; 0.2555 \; \text{act}/\text{sec (not normalized)},$$

$$A_{meas} \; (P^{32}) \; = \; 3.50 \; x \; 10^{10} \; \text{act}/\text{sec}/10^{24} \, S^{32} \; \text{atoms at 1 MW of reactor power,}$$

and

$$F \; (\text{based on } P^{32} \; \text{activity}) \; = \; A_{meas}/A_{calc} \; = \; 1.37 \; x \; 10^{11}.$$

Group	E_L	Cross Section σ_i, barns	Calculated Flux Shape, ϕ_i	$\sigma_i \phi_i$
1	7 788	0 350	0 0206	0 0072
2	6 065	0 335	0 0570	0 0191
3	4 724	0 260	0 1606	0 0418
4	3 679	0 245	0 2545	0 0624
5	2 865	0 170	0 3898	0 0663
6	2 231	0 075	0 5358	0 0402
7	1 738	0 030	0 6161	0 0185
8	1 353	0	0 6734	0

$$A \; = \; \sum_1 \sigma_i \phi_i \; = \; \overline{0 \; 2555}$$

Now the multigroup fluxes in the table can be multiplied by 1.37×10^{11} to give the results in the table below, and flux units can be put on the graph.

Group	Calculated Flux Shape	Neutron Flux, n/cm^2-sec
1	0.0206	2.82×10^9
2	0.0570	7.81×10^9
3	0.1606	2.20×10^{10}
4	0.2545	3.49×10^{10}
.	.	.
.	.	.
.	.	.
20	0.4749	6.51×10^{10}
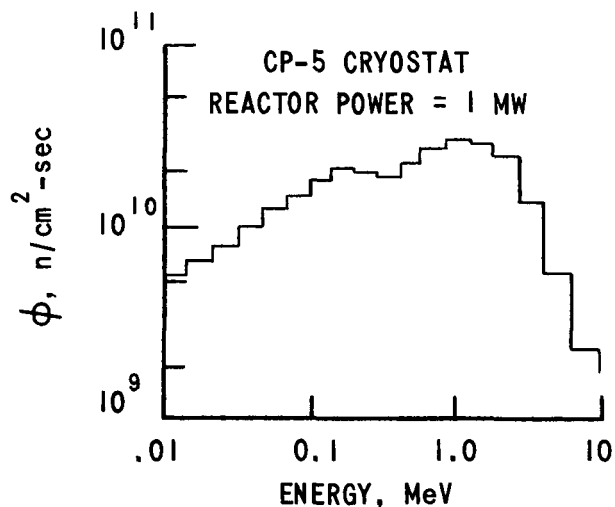
When data from additional foils are available, each foil can give a value for the normalization factor F. Thus an average can be obtained, and should one foil result be badly in error, it will show up when compared with the others. One can also study the ratios of activation rates of various pairs of detecting foils, both measured ratios and those calculated using the multigroup spectra and cross sections.

If a set of exponentials or other analytical expressions is used for the flux, the normalization can be performed in two ways. The cross section can be approximated analytically so that the integration

$$\int_0^\infty \sigma(E) \, \phi(E) \, dE$$

can be performed, or, the spectrum can be broken into a multigroup format. In either event, the remainder of the procedure to obtain the normalization factor remains the same.

The more finely divided the energy group structure, the better the approximation to a real integral. Use of only a few broad groups will yield adequate results in some spectra but may fail badly in others. The concept of a single threshold energy and its associated effective cross section is the limiting case of two energy groups; hence, the values only have meaning for the spectrum used in their definition. This fact follows from the definition
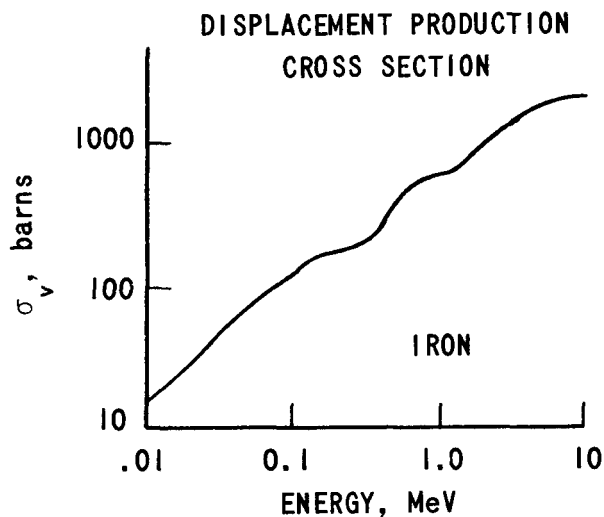
$$\sigma_{eff} = \frac{\int_{E_t}^\infty \sigma(E) \, \phi(E) \, dE}{\int_0^\infty \phi(E) \, dE},$$

and any change in spectrum $\phi(E)$ or threshold energy E_t will give a different value for σ_{eff}. Thus, to use effective cross sections, one must first know the spectrum to be determined. The multigroup format successfully avoids this limitation.

4. RELATIVE IMPORTANCE OF NEUTRONS

The previous two steps have produced a complete picture of the fast neutron flux. These neutrons are spread over an energy range of greater than a factor of 100. Neutrons of different energies may have different relative effectiveness in producing radiation damage. The detailed mechanisms involved in radiation damage are not completely understood. However, theories have been proposed for several parts of the process. Perhaps these can be applied to arrive at the relative importance of neutrons of various energies.

The energy-dependent model used by Rossin[3] and Hyder[4] indicates that the importance of a neutron is proportional to its energy and to the probability that it has an elastic scattering collision. The theory, as presented by Snyder and Neufeld[5] or Kinchin and Pease,[6] comes from basic energy and momentum considerations applied to the scattering event. Most of the energy transferred by the neutron to a struck lattice atom ultimately goes into displacing lattice atoms. This model therefore implies that the amount of damage is related to the number of displacements produced. This model of neutron importance for fast neutrons in iron[3] is sketched at the left. The dimensions of the damage cross section are the same as for microscopic neutron cross sections,

DISPLACEMENT PRODUCTION CROSS SECTION

IRON

ordinate: σ_v, barns — values 10, 100, 1000

abscissa: ENERGY, MeV — values .01, 0.1, 1.0, 10

$\dfrac{\text{events}}{\text{neutron}}$ (cm^2 x 10^{-24}), or "barns."

If (as it is classically stated) it takes about 25 eV to displace an iron atom, the cross section could be put in units of the number of displacements produced if all energy available due to the collision goes to produce displacements. Using this assumption, the sketch suggests that a 2-MeV neutron causes over 2000 displaced atoms/cc. There are many theories on how many displacements are produced or survive more than a few microseconds after the neutron collision. Since the purpose of the model is to determine neutron exposure, only the relative effectiveness of the neutrons need be used, and the numbers in the ordinate do not need to be normalized to anything.

Variations on the Model

At the high-energy end of the spectrum, the energy-dependent model exaggerates neutron effectiveness. This is because primary knock-ons having extremely high energies lose some energy by ionization of neighboring atoms rather than by elastic collisions. Ionization does not give rise to displacements. Therefore the damage model tends to level off above some energy (not universally agreed upon). Since the cross section for isotropic scattering is also decreasing because of forward-peaked scattering at high energies, the damage model shows a further dropoff above 2 MeV.
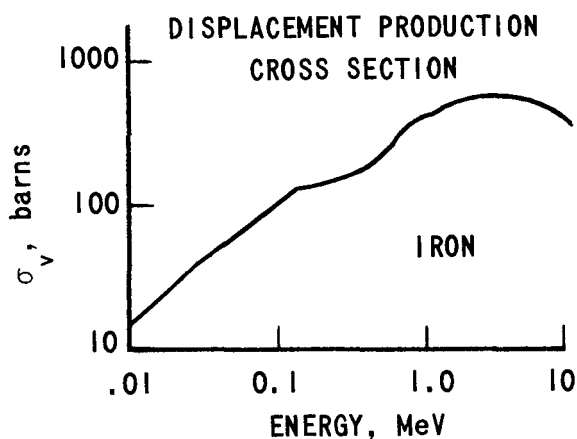
DISPLACEMENT PRODUCTION CROSS SECTION

IRON

σ_v, barns / ENERGY, MeV
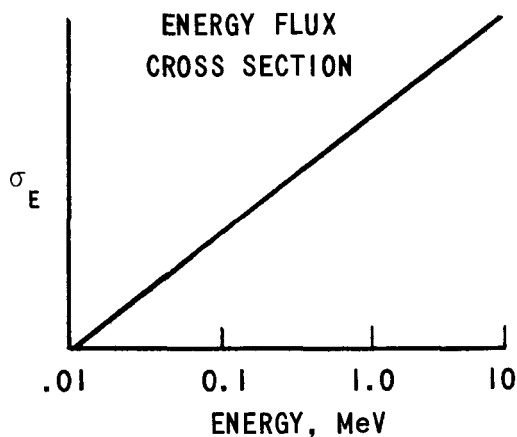
Other models have been used that are based on neutron energy alone. Here the effectiveness is assumed to be directly proportional to neutron energy ignoring any influence of scattering cross section. This model gives the "energy flux."

ENERGY FLUX CROSS SECTION

σ_E / ENERGY, MeV

Many investigators have used a step-function damage model without referring to it in these terms. The most familiar model is shown below (right), where damaging flux is expressed in terms of neutrons greater than 1 MeV (solid line). Thus any neutron greater than 1 MeV is taken to have one unit of effectiveness, and those less than 1 MeV are not counted. Any other arbitrary step choice is possible, as suggested by the dotted step at 0.3 MeV.

These models have all been expressed in arbitrary units, since the aim is to account for the relative effectiveness of different energy neutrons. For consistency, the models have been normalized below so that

>1MeV MODEL

> 0.3 MeV MODEL →

σ / ENERGY, MeV

when integrated over the fission spectrum, the result is unity. That is,
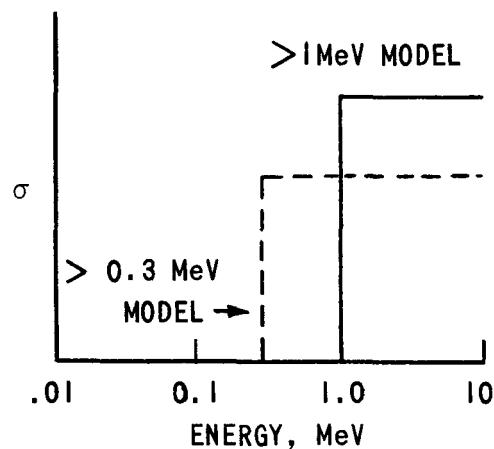
$$\int_0^\infty \sigma_D(E)\, \phi_f(E)\, dE \Big/ \int_0^\infty \phi_f(E)\, dE = 1.$$

This can be shown graphically. At the left (below), the damage cross section is a dotted line, the normalized fission spectrum is shown by dashes, and their product is a solid line. The cross sections (dotted line) have been adjusted to make the shaded area under the solid curve equal unity. The multigroup values of various damage cross sections in the table below have been normalized in this manner. Appendix B presents details of the computation of RDU values for iron and copper.
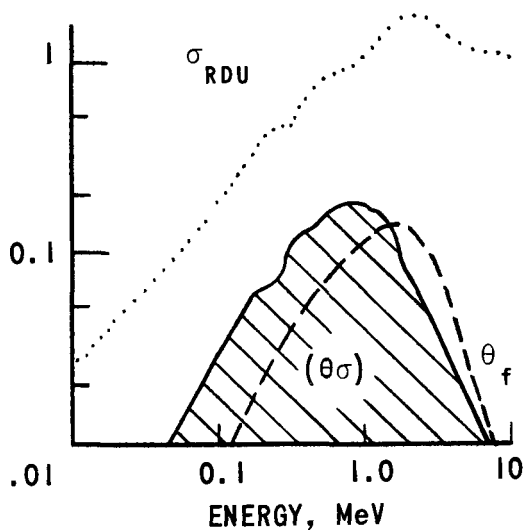
The unit shaded area could be used as a unit of radiation damage exposure. This quantity of fast neutrons may be called one "radiation damage unit," or RDU. Thus 1000 neutrons having a perfect fission-spectrum energy distribution will deliver 1000 RDU, by definition.

SHADED AREA = 1 RDU

MULTIGROUP CROSS SECTIONS

Group	E_L, MeV	Activation				Damage		
		$S^{32}(n,p)P^{32}$, barns	$Ni^{58}(n,p)Co^{58}$, barns	$Fe^{54}(n,p)Mn^{54}$, barns	U^{238} fiss, barns	RDU (iron)	RDU (Cu)	RDU >1 MeV
1	7 788	0 350	0 660	0 650	1 040	1 223	1 357	1 45
2	6 065	0 335	0 610	0 600	0 840	1 213	1 542	1 45
3	4 724	0 260	0 540	0 530	0 600	1 121	1 409	1 45
4	3 679	0 245	0 410	0 375	0 567	1 643	1 313	1 45
5	2 865	0 170	0 240	0 150	0 572	1 601	1 182	1 45
6	2 231	0 075	0 140	0 075	0 583	1 416	1 156	1 45
7	1 738	0 030	0 060	0 030	0 528	1 164	1 127	1 45
8	1 353	0	0 015	0 005	0 283	0 991	1 088	1 45
9	1 054		0 005	0	0 0568	0 736	0 998	1 45
10	0 821		0		0 0114	0 609	0 889	0 34
11	0 639				0 0027	0 749	0 777	0
12	0 498				0 0006	0 475	0 685	
13	0 388				0	0 647	0 580	
14	0 302					0 358	0 504	
15	0 235					0 291	0 400	
16	0 183					0 240	0 311	
17	0 143					0 234	0 257	
18	0 111					0 156	0 209	
19	0 086					0 170	0 154	
20	0 067					0 120	0 147	

To find the exposure rate, the neutron spectrum, normalized to a particular reactor power level, is multiplied by the RDU cross section in the same manner in which the activation rate of a foil is calculated. The RDU cross section is expressed in multigroup format or as an analytical

function of neutron energy. The result is an RDU rate. In a pure fission spectrum, all the damage models discussed above will give the identical RDU rate. From the definition of the RDU, this RDU rate will be identical to the number of fission neutrons. In any other spectrum, the different models will give different RDU rates. As an example, the damage models used in the table above are applied to four actual spectra and the results are shown below. The spectra were arbitrarily normalized to the $S^{32}(n,p)P^{32}$ activation rate; that is a sulfur foil would give the same activation in each.

		Spectrum		
Model	Fission	CP-5 Fuel	CP-5 Dummy	EBR-I
$S^{32}(n,p)P^{32}$ act/sec/10^{24}	1.00	1.00	1.00	1.00
ϕ total if fiss. spect.	(15.35)	(15.35)	(15.35)	(15.35)
RDU (Iron)	15.35	22.93	24.29	26.09
$\phi > 1$ MeV	(10.68)	(12.78)	(11.57)	(13.50)
RDU (>1 MeV)	15.35	18.52	16.77	19.58
RDU (>0.3 MeV)	15.35	25.13	25.06	33.02

It was shown above that knowledge of the spectral shape is needed to permit activation data to give a true picture of the neutron flux. The relative importance of the neutrons also has an influence. Compare two methods: The RDU rates determined above for the EBR-I and the CP-5 fuel element for identical sulfur activation rates give the ratio 1.14:1. If fission spectra had been arbitrarily assumed in each case, the result would have been off by 14% This is not necessarily the worst case, nor do these deviations for particular sets of assumptions follow a simple pattern. In addition to the author's work, numerous similar examples are in the literature, such as, Dahl and Yoshikawa,[7] Shure,[8] Pawlicki,[9] Claiborne,[10] and Wright.[11]

5. INTEGRATED EXPOSURE

Once the instantaneous RDU rate has been determined for a given reactor power, the integrated exposure can be obtained directly from the operating history of the reactor. It is useful to determine the dose rate in RDU/sec for a nominal power level, such as 1 MW. Then the number of MW-sec the reactor has operated, multiplied by RDU/sec at 1 MW, gives the integrated exposure.

A word of caution should be given about the simple concept of MW-sec of reactor operation. The experimenter must find out exactly how that number is obtained. Some reactor operators do not include start-up or

shutdown transients in their reports, and sometimes conflicting values from different sensors are reported. Although transients may be negligible in long-term irradiations, they can be significant in short runs for calibration or dosimetry purposes.

Thus the integrated exposure can be reported in RDU for any reactor irradiation. For design purposes, the approach is the same except that foil results upon which the spectrum shape may be normalized will not be available. The neutron-flux intensity must be calculated from the fission rate in the reactor core. The resulting uncertainty becomes greater with increases in distance from the fuel region.

Using these concepts, comparisons of irradiation data from different reactors will be free from errors resulting from differences in spectra. Irradiation effects are influenced strongly by temperature, differences in material composition and history, and even by dose rate at elevated temperatures. For comparison, these variables must be carefully analyzed. To understand their influence, accurate exposure measurements throughout an entire experimental program are essential.

APPENDIX A

STEP-BY-STEP PROCEDURE FOR CALCULATING RADIATION DAMAGE EXPOSURE USING MULTIGROUP METHODS

1. Spectrum Shape

a. Select the multigroup code and cross section set that will give adequate detail in the energy range between 100 keV (or below) and 10 MeV. (Twenty energy groups of 0.25 lethargy unit width, from 67 keV to 10 MeV, are used in the example.)

b. Choose an idealized geometry that represents the location of interest. Supply, as input, the dimensions and compositions of the regions and the distribution of fission neutron sources.

c. Tabulate the calculated spectrum at the point of interest as in Column (2) of Table A-1.

Table A-1

SAMPLE CALCULATION TABLE
EBR-I, MIDPLANE, C3, 1 MW

(1)	(2)	(3)	(4)	(5)
Group	Calculated Spectrum	σ_{act} (barns) $Fe^{54}(n,p)Mn^{54}$	Flux at 1 MW (2)·(F)	σ_{RDU}
1	0.053	0.650	0.051×10^{13}	1.223
2	0.147	0.600	0.141	1.213
3	0.341	0.530	0.328	1.121
4	0.606	0.375	0.582	1.643
5	0.894	0.150	0.859	1.601
6	1.107	0.075	1.064	1.416
7	1.203	0.030	1.156	1.164
8	1.419	0.005	1.364	0.991
9	1.420	0	1.365	0.736
10	1.389	0	1.335	0.609
11	1.580	0	1.518	0.749
12	1.673	0	1.608	0.475
13	1.409	0	1.354	0.647
14	1.317	0	1.266	0.358
15	1.079	0	1.037	0.291
16	0.912	0	0.876	0.240
17	0.632	0	0.607	0.234
18	0.531	0	0.510	0.156
19	0.291	0	0.280	0.170
20	0.277	0	0.266	0.120

$$A^* = \sum_{20} (2)(3) = 0.7909; \quad A_{ex} = 7.60 \times 10^{12} \text{ act/sec}/10^{24} \text{ atoms/MW}$$

$$F = \frac{A_{ex}}{A^*} = \frac{7.60 \times 10^{12}}{0.7909} = 9.61 \times 10^{12}; \quad RDU = \sum_{20} (4)(5) = 1.30 \times 10^{14} \text{ RDU/sec/MW}$$

2. Absolute Magnitude of the Flux: Experimental

      a. For each foil material available, tabulate its activation cross section, as in Column (3) of Table A-1 for the reaction $Fe^{54}(n,p)Mn^{54}$.

      b. Multiply the flux by the cross section, Column (2) x Column (3), and add, thus obtaining the unnormalized activation rate, A*.

      c. From irradiated foils obtain the activation rate, A_{ex}, in units of activations per second per 10^{24} per Megawatt of reactor power.

      d. Compute the ratio of each experimentally determined activation rate to its calculated counterpart. Analyze these ratios to determine the normalization factor F for the calculated flux. (If they agree, or if only one foil is available, F is uniquely determined. If discrepancies exist, F is the average of all reasonable ratios.)

      e. Multiply the unnormalized flux values by F to give the actual flux spectrum. Tabulate as in Column (4).

3. Absolute Magnitude of the Flux: Design

      a. Since foil data cannot exist, obtain the normalization from the design power density of the reactor. Normalize the neutron source distribution to that expected at 1 MW.

      b. Tabulate the flux spectrum as in e above, and normalize to required reactor power level.

      c. Calculate predicted activation rates for various detectors as above, if desired.

4. Damage Rate

      a. Select the appropriate damage model. Identify the choice and reasons for it [Column (5) of Table A-1]. The example used is the energy-dependent model (σ_{RDU}) for iron.

      b. Multiply the multigroup fluxes, Column (4), by the RDU multigroup values, Column (5), and add. This gives the damaging exposure rate in RDU per second per Megawatt of reactor power.

      c. To find the instantaneous damage rate, multiply the above by the actual reactor power level in Megawatts.

5. Integrated Exposure

a.   From the reactor operating history (strip chart, integrated power indicator, log book, etc.) determine the number of megawatt-seconds of operation during which the sample was in the reactor.

b.   Multiply this integrated power by the RDU rate to find total RDU exposure that the specimen has received.

6. Simplified Procedure for Irradiations of Steel Specimens

a.   Consider a number of steel specimens distributed in a reactor system, and calculate the spectrum as a function of position.

b.   Calculate the relative $Fe^{54}(n,p)Mn^{54}$ activation rate and the RDU rate.

c.   Construct curves showing the ratio of RDU/Fe^{54}_{act} rates as a function of position in the core.

d.   Count the Mn^{54} activity in each sample. This makes each specimen its own dosimeter foil.

e.   Make the arithmetic corrections for buildup and decay to give the activation rate of Fe^{54} in each.

f.   From the plot of ratio versus position, obtain directly for each specimen the RDU exposure rate.

g.   Multiply exposure rates by integrated reactor power to obtain the total integrated exposure.

APPENDIX B

DETERMINATION OF RDU VALUES FOR THE ENERGY-DEPENDENT MODEL FOR IRON AND COPPER

The energy-dependent damage cross section σ_{RDU} is constructed using the formula

$$\sigma_{RDU}(E) = c\sigma_{iso}(E)E. \tag{B-1}$$

Here, σ_{RDU} is normalized so that

$$\frac{\int_0^\infty \sigma_{RDU}(E)\,\phi_f(E)\,dE}{\int_0^\infty \phi_f(E)\,dE} = 1 \tag{B-2}$$

where

E is the incident neutron energy,

c is the constant of normalization, and

σ_{iso} is the cross section for isotropic scattering of the fast neutron by the lattice atom.

Note that

$$\sigma_{iso} = (\sigma_{tot} - \sigma_c - \sigma_{in})(1 - \bar{\mu}), \tag{B-3}$$

where

σ_{tot} is the total cross section,

σ_{in} is the inelastic scattering cross section,

σ_c is the capture cross section, and

$\bar{\mu}$ is the transport correction for anisotropic scattering; $\bar{\mu}$ is large if forward scattering is appreciable.

Equation (B-3) is developed in the author's first paper on this subject.[3] The multigroup format was used, but the normalization and RDU concept are new. The 20 energy group structure used here gives more detail in the high-energy range, and the quarter-lethargy unit group widths are convenient because they are used with several popular cross section libraries. The multigroup structure is tabulated in the report.

Tables B-1 and B-2 present the RDU cross sections for iron and copper. Cross sections are taken from BNL-325 and other cross section sets. Values for $\bar{\mu}$ for iron were taken from Yiftah,[12] and for copper from Wollenberger.[13] The fission spectrum is Cranberg's, as given in ANL-5800 (Rev.).[1]

Table B-1

σ_{RDU} FOR IRON

Group	E_{ave}, MeV	σ_{iso}, barns	ϕ_f	σ_{RDU}, barns	Group	E_{ave}, MeV	σ_{iso}, barns	ϕ_f	σ_{RDU}, barns
1	8.825	0.336	0.0064	1.223	11	0.723	2.512	0.0646	0.749
2	6.873	0.428	0.0168	1.213	12	0.564	2.044	0.0501	0.475
3	5.352	0.508	0.0464	1.121	13	0.438	3.580	0.0371	0.647
4	4.167	0.956	0.0681	1.643	14	0.342	2.536	0.0276	0.358
5	3.245	1.196	0.0966	1.601	15	0.267	2.644	0.0197	0.291
6	2.528	1.358	0.1140	1.416	16	0.208	2.787	0.0143	0.240
7	1.970	1.432	0.1190	1.164	17	0.161	3.523	0.0100	0.234
8	1.534	1.568	0.1120	0.991	18	0.125	3.014	0.0071	0.156
9	1.193	1.496	0.0983	0.736	19	0.098	4.222	0.0049	0.170
10	0.930	1.588	0.082	0.609	20	0.076	3.884	0.0035	0.120

Table B-2

σ_{RDU} FOR COPPER

Group	$\sigma_t - \sigma_c - \sigma_{in}$, barns	$\bar{\mu}$	$K \phi_f$	σ_{RDU}, barns	Group	$\sigma_t - \sigma_c - \sigma_{in}$, barns	$\bar{\mu}$	$K \phi_f$	σ_{RDU}, barns
1	2.15	0.78	0.0267	1.357	11	3.80	0.13	0.1544	0.777
2	2.30	0.70	0.0797	1.542	12	4.2	0.11	0.1056	0.685
3	2.20	0.63	0.2011	1.409	13	4.5	0.095	0.0662	0.580
4	1.90	0.49	0.2750	1.313	14	4.9	0.075	0.0428	0.504
5	1.80	0.38	0.3510	1.182	15	4.9	0.06	0.0242	0.400
6	1.90	0.26	0.4052	1.156	16	4.8	0.04	0.0137	0.311
7	2.20	0.20	0.4126	1.127	17	5.0	0.02	0.0079	0.257
8	2.66	0.18	0.3748	1.088	18	5.2	0.01	0.0046	0.209
9	3.10	0.17	0.3018	0.998	19	4.9	0.01	0.0023	0.154
10	3.46	0.15	0.2245	0.889	20	6.0	0.01	0.0016	0.147

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