



Designation: E 525 – 90 (Reapproved 1996)

Standard Practice for Reporting Dosimetry Results on Nuclear Graphite¹

This standard is issued under the fixed designation E 525; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 This practice covers procedures for determining and reporting the neutron fluence rate and fluence for the correlation of radiation-induced changes in nuclear graphites.

1.2 The purpose of this practice is to achieve better correlation and interpretation of new data in the field of radiation effects testing of specimens of graphites to be used for moderator or reflector components of fission reactors.

1.3 Excluded from this practice are graphite test specimens containing fissionable materials and specimens containing materials having high neutron cross sections.

2. Referenced Documents

2.1 ASTM Standards:

C 625 Practice for Reporting Irradiation Results on Graphite²

E 261 Practice for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques³

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, (IIA)³

E 1018 Guide for Application of ASTM Evaluated Cross Section Data File (ENDF/A), E 706(II B)³

3. Significance and Use

3.1 Practice C 625 covers information recommended for inclusion in reports giving graphite irradiation results, with the exception of neutron dosimetry results.

3.2 It is well documented (1,2,3)⁴ that mechanical properties, physical properties, and physical dimensions of graphites are altered by exposure to high-energy neutron radiation and that the amount, rate, and energy spectrum of the radiation influences the magnitude and relationship of the changes.

3.3 It is also well documented (4,5,6) that graphite irradiation results obtained at different dose rates or different energy spectra, or both, can be adequately correlated through the use of appropriate atomic displacement functions. The function that is most widely used for the correlation and interpretation

of graphite irradiation results is that developed by Thompson and Wright (7).

4. Exposure Unit for Graphite Irradiations

4.1 The “Equivalent Fission Fluence for Damage in Graphite,” Φ_G , is defined as follows:

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

where:

$\phi(E, t)$ = neutron fluence rate spectrum at time t ,

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

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

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

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

4.2 The two principal methods for obtaining neutron fluence rate spectra are: (1) calculations using reactor physics codes (8, 9), and (2) computer-iterative techniques (see Guide E 944) using activation data from many different neutron detector materials, simultaneously irradiated in the reactor facility. It is recommended that spectra obtained by either method be checked by the activation of monitors having significantly different response functions.

4.2.1 The response function, $R(E)$, of an activation monitor is defined as follows:

$$R(E) = \sigma_a(E) \phi(E) / \int_0^\infty \sigma_a(E) \phi(E) dE \quad (2)$$

where:

$\sigma_a(E)$ = activation cross section as a function of neutron energy, and

$\phi(E)$ = fluence spectrum at the measurement location.

4.2.2 Other considerations are discussed in Practice E 261 and other methods covering the use of individual detectors.

4.3 It is recommended that the fission spectrum, $\chi(E)$, the scattering cross section $\sigma_s(E)$ used in evaluating Eq 1, and all cross sections used in the evaluation of the neutron fluence spectrum, be processed from the most recent edition of the Evaluated Nuclear Data File (10, Guide E 1018).

4.4 An acceptable form of the atomic displacement weighting function $p(E)$ for evaluation of Eq 1 is the Thompson and Wright function. Group averaged values for this function are listed in Table 1 in the GAM-II energy group structure. Other

¹ This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.05 on Nuclear Radiation Metrology.

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² Annual Book of ASTM Standards, Vol 15.01.

³ Annual Book of ASTM Standards, Vol 12.02.

⁴ The boldface numbers in parentheses refer to the references at the end of this practice.

TABLE 1 Group Averaged Values^A of the Thompson and Wright Atomic Displacement Weighting Function for Damage in Graphite

Group Number	Upper Energy, MeV	Lower Lethargy ^B	Displacement Function ^C	Group Number	Upper Energy, MeV	Lower Lethargy	Displacement Function	Group Number	Upper Energy, MeV	Lower Lethargy	Displacement Function
1	14.918	0.10	548.4	26	1.2246	2.60	317.6	51	86.517	5.40	1.2
2	13.499	0.20	540.6	27	1.1080	2.70	306.9	52	67.379	5.65	49.7
3	12.214	0.30	532.7	28	1.0026	2.80	296.1	53	52.475	5.90	40.2
4	11.052	0.40	524.7	29	0.90718	2.90	285.2	54	40.868	6.15	32.3
5	10.000	0.50	516.5	30	0.82085	3.00	274.2	55	31.828	6.40	25.9
6	9.0484	0.60	508.2	31	0.74274	3.10	262.9	56	24.788	6.65	20.7
7	8.1873	0.70	499.8	32	0.67206	3.20	251.5	57	19.305	6.90	16.5
8	7.4082	0.80	491.5	33	0.60810	3.30	240.5	58	15.034	7.15	13.1
9	6.7032	0.90	482.9	34	0.55023	3.40	229.2	59	11.709	7.40	10.3
10	6.0653	1.00	474.1	35	0.49787	3.50	218.1	60	9.1188	7.65	8.0
11	5.4881	1.10	465.3	36	0.45049	3.60	207.3	61	7.1017	7.90	6.2
12	4.9659	1.20	456.3	37	0.40762	3.70	196.7	62	5.5308	8.15	4.8
13	4.4933	1.30	447.2	38	0.36883	3.80	186.3	63	4.3074	8.40	3.7
14	4.0657	1.40	437.6	39	0.33373	3.90	176.1	64	3.3546	8.65	2.9
15	3.6788	1.50	428.2	40	0.30197	4.00	166.2	65	2.6126	8.90	2.3
16	3.3287	1.60	418.7	41	0.27324	4.10	156.5	66	2.0347	9.15	1.8
17	3.0119	1.70	409.1	42	0.24724	4.20	147.0	67	1.5846	9.40	1.4
18	2.7253	1.80	399.4	43	0.22371	4.30	137.8	68	1.2341	9.65	1.1
19	2.4660	1.90	389.6	44	0.20242	4.40	128.8	69	0.96112	9.90	0.8
20	2.2313	2.00	379.7	45	0.18316	4.50	120.0	70	0.74852	10.15	0.6
21	2.0190	2.10	369.6	46	0.16573	4.60	111.5	71	0.58295	10.40	0.5
22	1.8268	2.20	359.4	47	0.14966	4.70	103.2	72	0.45400	10.65	0.4
23	1.6530	2.30	349.1	48	0.13569	4.80	95.2	73	0.35358	10.90	0.3
24	1.4957	2.40	338.7	49	0.12277	4.90	87.4	74	0.27536	11.15	0.2
25	1.3534	2.50	328.2	50	0.11109	5.15	74.8	75	0.21445	11.40 ^D	0.1

^A Standard GAM-II neutron energy group structure.

^B $E_o = 14.918$ MeV, lower energy for group number i is the upper energy of group number $(i + 1)$.

^C In units of displaced carbon atoms per neutron scattering event.

^D Lower energy of group Number 75 is 167.02 eV.

forms of $p(E)$ derived from the Evaluated Nuclear Data File are also acceptable if properly documented.

5. Report

5.1 Report the following information:

5.1.1 Reactor, test position, and dates of irradiation,

5.1.2 Information regarding the duration of exposure, position of neutron monitors relative to the samples, temperature of detector materials, and reactor power variations during the irradiation test,

5.1.3 The saturation activities of the monitors and all pertinent information used to obtain the saturation activities,

5.1.4 Tabulated values for the neutron spectrum in the form (group averaged or other) actually used, and all basic information used to obtain the spectrum,

5.1.5 Tabulated values for the scattering cross section of carbon and the atomic displacement function, or the product of the two functions, and

5.1.6 The normalization factor:

$$\int_0^\infty \sigma_s(E)p(E)\chi(E)dE / \int_0^\infty \chi(E)dE$$

and tabulated values for the fission neutron spectrum, $\chi(E)$, used in the calculations.

6. Precision and Bias

6.1 The neutron scattering cross section of carbon has been accurately determined (10) over most of the energy range encompassed by the neutron spectra in fission reactors; therefore, the precision with which Φ_G can be calculated will be primarily dependent on the uncertainties in $\phi(E, t)$ and $p(E)$.

6.2 The precision of neutron fluence rate and spectrum determination is dependent on the complexity of the irradiation

environment and on the extent of measured data available.

6.2.1 In the use of reactor physics codes to calculate neutron spectra, considerable professional judgment is required to choose a suitable code and to properly model the idealized space and materials configuration used to represent the actual reactor test position.

6.2.1.1 If the choices are appropriate and the calculated spectra are normalized on the basis of radioactivation results, the precision of the neutron fluence rate spectra in the energy range of interest can be expected to be (9,11) within about $\pm 10\%$ (one standard deviation) where a single set of materials cross sections are used in all calculations and perhaps $\pm 15\%$ where different sets of materials cross sections are used.

6.2.2 The precision of neutron fluence spectra derived by computer-iterative techniques using radioactivation results from an appropriate set of detector materials can also be expected (Guide E 944) to be within about $\pm 10\%$ (one standard deviation).

6.3 The neutron spectrum and fluence rate in an irradiation testing facility can vary as a complex function of time due to changes in power generation rates, materials densities, and isotopic content of fuel near the test facility. Uncertainties arising from fluence rate and spectrum variations need to be evaluated on a case-by-case basis.

6.4 Simplifications made in derivation of the Thompson and Wright form for $p(E)$ result in an underestimation of the energy lost by displaced atoms in electronic interactions; but the effect on precision is believed to be small for the range of energies of importance in fission reactor spectra.