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Standard Practice for Calculation of Absorbed Dose From Neutron Irradiation by Application of Threshold-Foil Measurement Data¹

This standard is issued under the fixed designation E 763; 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 Radiation effects test results can be correlated to the absorbed dose in the material being tested. This practice outlines a procedure for calculating the absorbed dose in a material given the following information: the material composition, the neutron fluence, and threshold-foil data related to the neutron spectrum. This practice can also be used to calculate that fraction of the total absorbed dose in silicon caused by displacements (see Practice E 722). This quantity can be used to correlate the relative permanent damage in silicon semiconductor devices from one test facility to another. The practice is applicable to fast-pulsed-reactors of the Godiva or TRIGA type, where a substantial fraction of the neutron fluence exceeds epithermal energies (that is, $E > 0.01$ MeV). A material exposed to the mixed neutron and gamma-ray radiation from such a reactor absorbs energy from both components. This practice provides a procedure for calculating the neutron absorbed dose, and a companion standard, Practice E 666, describes a technique for calculating the absorbed dose resulting from the gamma-ray component. It is limited to materials for which tabulated kerma factors are available.

1.2 To the greatest extent practical, the international system of units (SI) is used throughout this practice (see Practice E 380). The values stated in SI units are to be regarded as the standard.

1.3 *This standard does not purport to address all of the safety problems, if any, associated with its use. It is the responsibility of the user of this standard to establish appropriate safety and health practices and determine the applicability of regulatory limitations prior to use.*

2. Referenced Documents

2.1 ASTM Standards:

- E 170 Terminology Relating to Radiation Measurements and Dosimetry²
- E 181 General Methods for Detector Calibration and Analysis of Radionuclides²
- E 261 Test Method for Determining Neutron Fluence Rate, Fluence, and Spectra by Radioactivation Techniques²
- E 265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32²

E 343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity from Fission Dosimeters²

E 380 Practice for Use of the International System of Units (SI) (the Modernized Metric System)³

E 393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 from Fission Dosimeters²

E 666 Practice for Calculating Absorbed Dose from Gamma or X Radiation²

E 704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238²

E 705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237²

E 720 Guide for Selection of a Set of Neutron-Activation Foils for Determining Neutron Spectra Used in Radiation-Hardness Testing of Electronics²

E 721 Method for Determining Neutron Energy Spectra with Neutron Activation Foils for Radiation-Hardness Testing of Electronics²

E 722 Practice for Characterizing Neutron Energy Fluence Spectra in Terms of an Equivalent Monoenergetic Neutron Fluence for Radiation-Hardness Testing of Electronics²

E 844 Guide for Sensor Set Design and Irradiation for Reactor Surveillance²

E 854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance²

E 944 Guide for Application of Neutron Spectrum Adjustment Methods in Reactor Surveillance, (11A)²

2.2 International Commission on Radiation Units and Measurements (ICRU) Reports:

ICRU Report 13—Neutron Fluence, Neutron Spectra and Kerma⁴

ICRU Report 33—Radiation Quantities and Units⁴

ICRU Report 26—Neutron Dosimetry for Biology and Medicine⁴

3. Terminology

3.1 Descriptions of Terms Specific to This Standard:

3.1.1 *absorbed dose, D*—the quantity expressed in grays (Gy), given by the equation:

$$D = d\bar{\epsilon}/dm \quad (1)$$

where:

³ Annual Book of ASTM Standards, Vol 14.02.

⁴ Available from International Commission on Radiation Units and Measurements, 7910 Woodmont Ave., Washington, DC 20014.

¹ This practice is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications and is the direct responsibility of Subcommittee E10.07 on Radiation Dosimetry for Radiation Effects on Materials and Devices.

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² Annual Book of ASTM Standards, Vol 12.02.



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$\bar{\epsilon}$ = mean energy imparted by ionizing radiation to the matter in a volume element, and
 m = mass of the matter in that volume element.

3.1.2 *charged particle equilibrium (CPE)*—a condition that exists in a material under irradiation if the energies, number, and direction of charged particles induced by the radiation are constant throughout the volume of interest. Within such a volume, the sum of the energies of all charged particles entering it is equal to the corresponding sum for all particles leaving it.

NOTE 1—For detailed discussion of these and other related terms, see Terminology E 170 and ICRU Report 33.

3.1.3 *directly ionizing particles*—charged particles (electrons, protons, alpha particles, etc.) having sufficient kinetic energy to produce ionization by collision.

3.1.4 *energy-deposition event*—the deposition of energy in the mass of interest by correlated ionizing particles.

3.1.5 *energy imparted*, ϵ —the stochastic quantity imparted by ionizing radiation to the matter in a volume equal to:

$$\epsilon = \Sigma\epsilon_{in} - \Sigma\epsilon_{ex} + \Sigma Q \quad (2)$$

where:

$\Sigma\epsilon_{in}$ = sum of the energies (excluding rest energies) of all those directly and indirectly ionizing particles which have entered the volume,

$\Sigma\epsilon_{ex}$ = sum of the energies (excluding rest energies) of all those directly and indirectly ionizing particles which have left the volume, and

ΣQ = sum of all the energies released, minus the sum of all the energies expended, in any transformation of nuclei and elementary particles that have occurred within the volume.

3.1.6 *fluence*, Φ —of particles, the quantity given by the equation:

$$\Phi = dN/da \quad (3)$$

where:

N = number of particles that enter a sphere of cross-sectional area, a .

3.1.7 *indirectly ionizing particles*—uncharged particles (neutrons, photons, etc.) that can liberate directly ionizing particles or can initiate nuclear transformations.

3.1.8 *ionizing radiation*—any radiation consisting of directly or indirectly ionizing particles or a mixture of both.

3.1.9 The *kerma*, K , expressed in grays (Gy) is given by

$$K = dE_{tr}/dm \quad (4)$$

where:

E_{tr} = sum of the initial kinetic energies of all the charged particles liberated by indirectly ionizing particles in a volume element of the specified material, and

m = mass of the matter in that volume element.

3.1.10 *mass energy absorption coefficient*, μ_{en}/ρ , $m^2 \cdot kg^{-1}$ —of a material for indirectly ionizing particles of specified energy, the product of the mass energy transfer coefficient for that energy and $(1 - f)$ where f is the fraction of the energy of secondary charged particles that is lost to bremsstrahlung in the material.

$$\mu_{en}/\rho = (\mu_{tr}/\rho)(1 - f) \quad (5)$$

3.1.11 *mass energy transfer coefficient*, μ_{tr}/ρ —of a mate-

rial for indirectly ionizing particles of specified energy, $m^2 \cdot kg^{-1}$, is given by

$$\mu_{tr}/\rho = (1/\rho E)(dE_{tr}/dl) \quad (6)$$

where:

E_{tr}/E = fraction of incident particle energy (excluding rest energies) transferred to kinetic energy of charged particles by interactions in traversing a distance l in a medium of density ρ .

3.1.12 *mean energy imparted*—the expectation value of the energy imparted.

3.1.13 *specific energy or specific energy imparted*, z —the stochastic quantity, expressed in grays (Gy), given by the equation:

$$z = \epsilon/m \quad (7)$$

where:

ϵ = energy imparted by ionizing radiation to the matter in a volume element, and

m = mass of the matter in that volume element.

4. Significance And Use

4.1 Absorbed dose is one parameter often used for correlating the results of radiation-effects testing. When materials are exposed to a mixed neutron-gamma field, the absorbed dose deposited by neutrons can vary from a small fraction of the absorbed dose from gamma rays in high-Z materials, to many times the absorbed dose from gamma rays in low-Z materials. The neutron contribution to the absorbed dose is especially important for materials containing hydrogen (for example, tissue, hydrocarbon insulators, etc.).

4.2 Since energy deposition by neutrons in a material is an energy-dependent process, experience has shown that many radiation effects correlate better with absorbed-dose determinations than with neutron-fluence measurements. However, it is important to point out that a simple relationship between a particular effect being studied and absorbed dose may depend on other factors and, thus, should not be arbitrarily assumed.

4.3 Application of this practice provides an estimate of the absorbed dose in a material resulting from exposure to neutrons. The quantity actually calculated is the kerma. This quantity can be used to estimate absorbed dose with an accuracy that depends on the degree to which charged-particle equilibrium (CPE) exists in the test sample and, to a lesser extent, on knowledge of the detailed shape of the neutron spectrum.

4.4 Greater accuracy can be obtained by using techniques that include detailed neutron energy spectral determinations using multiple fission and activation foils, followed by an absorbed-dose calculation using a radiation-transport and energy-deposition program. However, these more expensive and time-consuming techniques are beyond the scope of this practice.

NOTE 3—Details of the application of these techniques can be found in Guides E 720 and E 844, Method E 721, and Practices E 722 and E 944.

5. Relationship Between Kerma and Absorbed Dose

5.1 For a given monoenergetic radiation the relation between energy fluence and kerma may be written as

$$K = \psi(\mu_{tr}/\rho) \quad (8)$$

