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Standard Guide for Benchmark Testing of Reactor Dosimetry in Standard and Reference Neutron Fields¹

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1. Scope

1.1 This guide covers facilities and procedures for benchmarking neutron measurements and calculations. Particular sections of the guide discuss: the use of well-characterized benchmark neutron fields to calibrate integral neutron sensors; the use of certified-neutron-fluence standards to calibrate radiometric counting equipment or to determine interlaboratory measurement consistency; development of special benchmark fields to test neutron transport calculations; use of well-known fission spectra to benchmark spectrum-averaged cross sections; and the use of benchmarked data and calculations to determine the uncertainties in derived neutron dosimetry results.

1.2 The values stated in SI units are to be regarded as standard. No other units of measurement are included in this standard.

2. Referenced Documents

2.1 ASTM Standards:²

- E170 Terminology Relating to Radiation Measurements and Dosimetry
- E261 Practice for Determining Neutron Fluence, Fluence Rate, and Spectra by Radioactivation Techniques
- E263 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Iron
- E264 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Nickel
- E265 Test Method for Measuring Reaction Rates and Fast-Neutron Fluences by Radioactivation of Sulfur-32
- E266 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Aluminum
- E343 Test Method for Measuring Reaction Rates by Analysis of Molybdenum-99 Radioactivity From Fission Do-

simeters (Withdrawn 2002)³

- E393 Test Method for Measuring Reaction Rates by Analysis of Barium-140 From Fission Dosimeters
- E482 Guide for Application of Neutron Transport Methods for Reactor Vessel Surveillance, E706 (IID)
- E523 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Copper
- E526 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Titanium
- E704 Test Method for Measuring Reaction Rates by Radioactivation of Uranium-238
- E705 Test Method for Measuring Reaction Rates by Radioactivation of Neptunium-237
- E854 Test Method for Application and Analysis of Solid State Track Recorder (SSTR) Monitors for Reactor Surveillance, E706(IIIB)
- E910 Test Method for Application and Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance, E706 (IIIC)
- E1297 Test Method for Measuring Fast-Neutron Reaction Rates by Radioactivation of Niobium
- E2006 Guide for Benchmark Testing of Light Water Reactor Calculations

3. Significance and Use

3.1 This guide describes approaches for using neutron fields with well known characteristics to perform calibrations of neutron sensors, to intercompare different methods of dosimetry, and to corroborate procedures used to derive neutron field information from measurements of neutron sensor response.

3.2 This guide discusses only selected standard and reference neutron fields which are appropriate for benchmark testing of light-water reactor dosimetry. The Standard Fields considered are neutron source environments that closely approximate the unscattered neutron spectra from ²⁵²Cf spontaneous fission and ²³⁵U thermal neutron induced fission. These standard fields were chosen for their spectral similarity to the

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² For referenced ASTM standards, visit the ASTM website, www.astm.org, or contact ASTM Customer Service at service@astm.org. For *Annual Book of ASTM Standards* volume information, refer to the standard's Document Summary page on the ASTM website.

³ The last approved version of this historical standard is referenced on www.astm.org.

high energy region ($E > 2$ MeV) of reactor spectra. The various categories of benchmark fields are defined in Terminology **E170**.

3.3 There are other well known neutron fields that have been designed to mockup special environments, such as pressure vessel mockups in which it is possible to make dosimetry measurements inside of the steel volume of the “vessel.” When such mockups are suitably characterized they are also referred to as benchmark fields. A variety of these engineering benchmark fields have been developed, or pressed into service, to improve the accuracy of neutron dosimetry measurement techniques. These special benchmark experiments are discussed in Guide **E2006**, and in Refs **(1)**⁴ and **(2)**.

4. Neutron Field Benchmarking

4.1 To accomplish neutron field “benchmarking,” one must perform irradiations in a well-characterized neutron environment, with the required level of accuracy established by a sufficient quantity and quality of results supported by a rigorous uncertainty analysis. What constitutes sufficient results and their required accuracy level frequently depends upon the situation. For example:

4.1.1 Benchmarking to test the capabilities of a new dosimeter;

4.1.2 Benchmarking to ensure long-term stability, or continuity, of procedures that are influenced by changes of personnel and equipment;

4.1.3 Benchmarking measurements that will serve as the basis of intercomparison of results from different laboratories;

4.1.4 Benchmarking to determine the accuracy of newly established benchmark fields; and

4.1.5 Benchmarking to validate certain ASTM standard methods or practices which derive exposure parameters (for example, fluence > 1 MeV or dpa) from dosimetry measurements and calculations.

5. Description of Standard and Reference Fields

5.1 There are a few facilities which can provide certified “free field” fluence irradiations. The following provides a list of such facilities. The emphasis is on facilities that have a long-lived commitment to development, maintenance, research, and international interlaboratory comparison calibrations. As such, discussion is limited to recently existing facilities.

5.2 ²⁵²Cf Fission Spectrum—Standard Neutron Field:

5.2.1 The standard fission-spectrum fluence from a suitably encapsulated ²⁵²Cf source is characterized by its source strength, the distance from the source, and the irradiation time. In the U.S., neutron source emission rate calibrations are all referenced to source calibrations at the National Institute of Standards and Technology (NIST) accomplished by the MnSO₄ technique **(3)**. Corrections for neutron absorption, scattering, and other than point-geometry conditions may, by careful experimental design, be held to less than 3 %. Associ-

ated uncertainties for the NIST ²⁵²Cf irradiation facility are discussed in Ref **(4)**. The principal uncertainties, which only total about 2.5 %, come from the source strength determination, scattering corrections, and distance measurements. Extensive details of standard field characteristics and values of measured and calculated spectrum-averaged cross sections are all given in a compendium, see Ref **(5)**.

5.2.2 The NIST ²⁵²Cf sources have a very nearly unperturbed spontaneous fission spectrum, because of the light-weight encapsulations, fabricated at the Oak Ridge National Laboratory (ORNL), see Ref **(6)**.

5.2.3 For a comprehensive view of the calibration and use of a special (32 mg) ²⁵²Cf source employed to measure the spectrum-averaged cross section of the ⁹³Nb(n,n') reaction, see Ref **(7)**.

5.3 ²³⁵U Fission Spectrum—Standard Neutron Field:

5.3.1 Because ²³⁵U fission is the principal source of neutrons in present nuclear reactors, the ²³⁵U fission spectrum is a fundamental neutron field for benchmark referencing or dosimetry accomplished in reactor environments. This remains true even for low-enrichment cores which have up to 30 % burnup.

5.3.2 There are currently two ²³⁵U standard fission spectrum facilities, one in the thermal column of the NIST Research Reactor **(8)** and one at CEN/SCK, Mol, Belgium **(9)**.

5.3.3 A standard ²³⁵U neutron field is obtained by driving (fissioning) ²³⁵U in a field of thermal neutrons. Therefore, the fluence rate depends upon the power level of the driving reactor, which is frequently not well known or particularly stable. Time dependent fluence rate, or total fluence, monitoring is necessary in the ²³⁵U field. Certified fluence irradiations are monitored with the ⁵⁸Ni(n,p)⁵⁸Co activation reaction. The fluence-monitor calibration must be benchmarked.

5.3.4 For ²³⁵U, as for ²⁵²Cf irradiations, small (nominally < 3 %) scattering and absorption corrections are necessary. In addition, for ²³⁵U, gradient corrections of the measured fluence which do not simply depend upon distance are necessary. The scattering and gradient corrections are determined by Monte Carlo calculations. Field characteristics of the NIST ²³⁵U Fission Spectrum Facility and associated measured and calculated cross sections are given in Ref **(5)**.

5.4 There are several additional facilities that can provide free field fluence irradiations that qualify as reference fields. The following is a list of some of the facilities that have characterized reference fields:

5.4.1 Annular Core Research Reactor (ACRR) Central Cavity – Reference Neutron Field **(10,11)**,

5.4.2 ACRR Lead-Boron Cavity Insert – Reference Neutron Field **(11)**,

5.4.3 YAYOI fast neutron field – Reference Neutron Field **(12,13)**,

5.4.4 SIGMA-SIGMA neutron field – Reference Neutron Field **(12,13)**.

6. Applications of Benchmark Fields

6.1 *Notation—Reaction Rate, Fluence Rate, and Fluence—* The notation employed in this section will follow that in **E261** (Standard Practice for Determining Neutron Fluence Rate, and

⁴ The boldface numbers given in parentheses refer to a list of references at the end of the text.

Spectra by Radioactivation Techniques) except as noted. The reaction rate, R , for some neutron-nuclear reaction {reactions/(dosimeter target nucleus)(second)} is given by:

$$R = \int_0^{\infty} \sigma(E) \varphi(E) dE \quad (1)$$

or:

$$R = \bar{\sigma} \varphi \quad (2)$$

where:

- $\sigma(E)$ = the dosimeter reaction cross section at energy E (typically of the order of 10^{-24} cm²),
- $\varphi(E)$ = the differential neutron fluence rate, that is the fluence per unit time and unit energy for neutrons with energies between E and $E + dE$ (neutrons cm⁻² s⁻¹ MeV⁻¹),
- φ = the total fluence rate (neutrons cm⁻² s⁻¹), the integral of $\varphi(E)$ over all E , and
- $\bar{\sigma}$ = the spectral-averaged value of $\sigma(E)$, R/φ .

NOTE 1—Neutron fluence and fluence rate are defined formally in Terminology E170 under the listing “particle fluence.” Fluence is just the time integral of the fluence rate over the time interval of interest. The fluence rate is also called the flux or flux density in many papers and books on neutron transport theory.

6.1.1 The reaction rate is found experimentally using an active instrument such as a fission chamber (see Ref (14)) or a passive dosimeter such as a solid state track recorder (see Test Method E854), a helium accumulation fluence monitor (see Test Method E910), or a radioactivation dosimeter (see Practice E261). For the radioactivation method, there are also separate standards for many particularly important dosimetry nuclides, for example, see Test Methods E263, E264, E265, E266, E343, E393, E523, E526, E704, E705, and E1297.

6.2 *Fluence Rate Transfer*: Note that if one determines $\varphi = R/\bar{\sigma}$ from Eq 2, then the uncertainty in φ will be a propagation of the uncertainties in both R and $\bar{\sigma}$. The uncertainty in $\bar{\sigma}$ is frequently large, leading to a less accurate determination of φ than desired. However, if one can make an additional irradiation of the same type of dosimeter in a standard neutron field with known fluence rate, then one may apply Eq 2 to both irradiations and write

$$\varphi_A = \varphi_B (R_A/R_B) (\bar{\sigma}_B/\bar{\sigma}_A) \quad (3)$$

where “A” denotes the field of interest and “B” denotes the standard neutron field benchmark. In Eq 3 the ratios of spectral average cross section, will have a small uncertainty if the spectral shapes $\varphi_A(E)$ and $\varphi_B(E)$ are fairly similar. There may also be important cancellation of poorly known factors in the ratio R_A/R_B , which will contribute to the better accuracy of Eq 3. Whether φ is better determined by Eq 3 or Eq 2 must be evaluated on a case by case basis. Often the fluence rate from Eq 3 is substantially more accurate and provides a very useful validation of other dosimetry. The use of a benchmark neutron field irradiation and Eq 3 is called fluence rate transfer.

6.2.1 *Certified Fluence or Fluence Rate Irradiations*—The primary benefit from carefully-made irradiations in a standard neutron field is that of knowing the neutron fluence rate. Consider the case of a lightly encapsulated ²⁵²Cf sintered-oxide bead, which has an emission rate known to about $\pm 1.5\%$ by calibration in a manganese bath (MnSO₄ solution).

Further, consider a dosimeter pair irradiated in compensated beam geometry (with each member of the pair equidistant from, and on opposite sides of, the ²⁵²Cf source). For such an irradiation in a large room (where very little room return occurs), the fluence rate – with a ²⁵²Cf fission spectrum – is known to within $\pm 3\%$ from the source strength, and the average distance of the dosimeter pair from the center of the source. Questions concerning in- and out-scattering by source encapsulation, source and foil holders, and foil thicknesses may be accurately investigated by Monte Carlo calculations. There is no other neutron-irradiation situation that can approach this level of accuracy in determination of the fluence or fluence rate.

6.2.2 *Fluence Transfer Calibrations of Reference Fields*—The benefit of irradiating with a source of known emission rate is lost when one must consider reactor cores or, even, thermal-neutron fissioned ²³⁵U sources. When the latter are carefully constructed to provide for an unmoderated ²³⁵U spectrum, this mentioned disadvantage can be circumvented by a process called fluence transfer. As explained briefly in 6.2, this process is basically as follows. A gamma-counter (spectrometer) geometry is chosen to enable proper counting of the activities of a particular isotopic reaction for example, ⁵⁸Ni(n,p)⁵⁸Co, after irradiation in either a ²⁵²Cf or ²³⁵U field. Then the ²⁵²Cf irradiation is accomplished and the nickel foil counted. From this, a ratio of the dosimeter response divided by the ²⁵²Cf certified fluence is determined. Subsequently, an identical nickel is irradiated in the ²³⁵U spectrum and that foil is counted with the same counter geometry. Within the knowledge of the ratio of the spectrum average cross sections in the two spectra, knowledge of the counter response to the recent irradiation yields the average ²³⁵U fluence. Note, the average fluence is measured. The thermal fluence rate at the ²³⁵U sources may not have been constant over the time of the irradiation but that time is assumed to be short relative to the 70 day half-life of the ⁵⁸Co, which monitors the fast neutron fluence through-out the irradiation. The method of calibration is termed fluence rate transfer because it is fluence rate which is determined, and there is no need to determine the absolute radioactivity of the dosimeters. Relative response of the same counter geometry is the only requirement.

6.2.3 *Reactor Irradiations*—In principle, the same fluence-transfer procedures can be applied to more complex irradiations. However, there are certain other situations which must be considered and weighed to determine if fluence transfer or reaction rate determination is the better method. Also remember that error estimation can be examined by using both methods.

6.2.3.1 If radioactivation dosimeters are employed for long term irradiations in a power reactor, the fluence at a dosimeter location can be determined by the method explained in 9.7, Long Term Irradiations, of Practice E261, taking into account the relative power level changes over the course of the irradiation. There may be practical problems, however. In particular, if the measured activity does not have a sufficiently long half-life, it can not provide a correct measure of the fluence. Said another way, if the dosimeter exposure time is more than about 3.5 times the half-life of the radioactive