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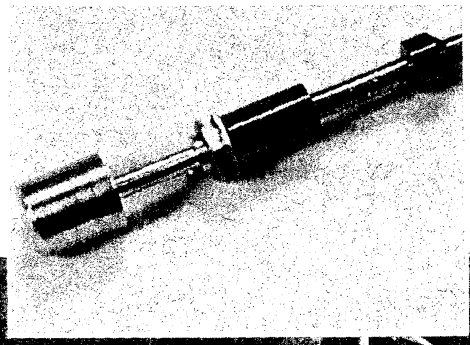
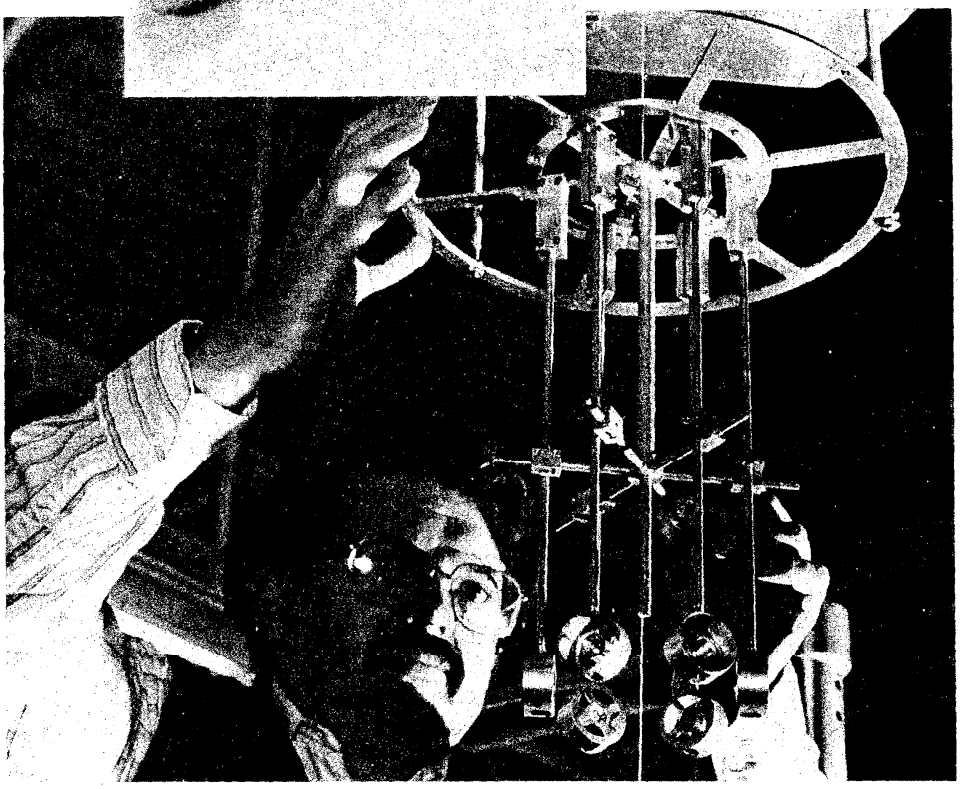


Activation Foil Irradiation with Californium Fission Sources



NBS Special Publication 250-13.

George P. Lamaze
James A. Grundl

U.S. Department of Commerce
National Bureau of Standards

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- Radioactivity
- X-Ray Physics
- Dosimetry

Nuclear Physics Group

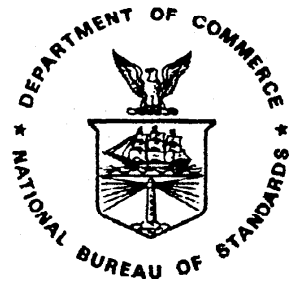
Engages in forefront research in nuclear and elementary particle physics; performs highly accurate measurements and theoretical analyses which probe the structure of nuclear matter; and improves the quantitative understanding of physical processes that underlie measurement science.

NBS MEASUREMENT SERVICES: ACTIVATION FOIL IRRADIATION WITH CALIFORNIUM FISSION SOURCES

George P. Lamaze

James A. Grundl

Center for Radiation Research
National Measurement Laboratory
National Bureau of Standards
Gaithersburg, MD 20899



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PREFACE

The calibration and related measurement services of the National Bureau of Standards are intended to assist the makers and users of precision measuring instruments in achieving the highest possible levels of accuracy, quality, and productivity. NBS offers over 300 different calibration, special test, and measurement assurance services. These services allow customers to directly link their measurement systems to measurement systems and standards maintained by NBS. These services are offered to the public and private organizations alike. They are described in NBS Special Publication (SP) 250, NBS Calibration Services Users Guide.

The Users Guide is being supplemented by a number of special publications (designated as the "SP 250 Series") that provide a detailed description of the important features of specific NBS calibration services. These documents provide a description of the: (1) specifications for the service; (2) design philosophy and theory; (3) NBS measurement system; (4) NBS operational procedures; (5) assessment of measurement uncertainty including random and systematic errors and an error budget; and (6) internal quality control procedures used by NBS. These documents will present more detail than can be given in an NBS calibration report, or than is generally allowed in articles in scientific journals. In the past NBS has published such information in a variety of ways. This series will help make this type of information more readily available to the user.

This document (SP 250-13), NBS Measurements Services: Activation Foil Irradiation with Californium Fission Sources, by G. P. Lamaze and J. A. Grundl, is the thirteenth to be published in this new series of special publications. The scope of the service, the neutron field characteristics, the irradiation procedures, and the uncertainties in the reported neutron fluences or fluence rates are described (see test number 44080C in the SP 250 Users Guide). Inquiries concerning the technical content of this document or the specifications for these services should be directed to the authors or one of the technical contacts cited in SP 250.

The Center for Radiation Research (CRR) is in the process of publishing 21 documents in this SP 250 series, covering all of the calibration services offered by CRR. A complete listing of these documents can be found inside the back cover.

NBS would welcome suggestions on how publications such as these might be made more useful. Suggestions are also welcome concerning the need for new calibration services, special tests, and measurement assurance programs.

Joe D. Simmons
Acting Chief
Measurement Services

Chris E. Kuyatt
Director
Center for Radiation Research

ABSTRACT

This document describes the NBS calibration service 44080C (formerly 8.1Q), which operates in the following way: (i) dosimetry sensors (metal foils, nuclear track detectors, wires, crystals, etc., supplied by the customer or by NBS) are irradiated to a certified neutron fluence in a ^{252}Cf spontaneous fission neutron field (for short lived reaction products, the samples are irradiated to near saturation and the fluence rate specified), and (ii) the irradiated dosimetry sensors are shipped to the customer, followed by a test report stating the fluences or fluence rates and associated uncertainties. The scope and philosophy of the service, the neutron field characteristics, irradiation procedures, and the uncertainties in the reported neutron fluences or fluence rates are discussed. Typical maximum fluences are of the order of 1×10^{13} neutrons/cm^2 and maximum fluence rates of 1×10^7 neutrons/cm^2 sec, with combined (1σ) uncertainties of $\pm$ 1.2%.

Key words: activation; californium; dosimetry; fast neutrons; fission spectrum; neutron fluence; neutron fluence rate

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I. DESCRIPTION OF SERVICES AND FACILITIES

NBS maintains standard fission neutron fields to supply calibration irradiations and certified fluence and fluence rate standards. The NBS ^{252}Cf Fission Neutron Source (typical maximum fluence of 1.0×10^{13} neutrons/cm^2 in 24 hours) provides one such standard neutron field spectrum. Known fluences and fluence rates are based upon Cf source strengths that are traceable to NBS-I, the U.S. national fast-neutron source standard.

This calibration service (44080C) operates in the following way: 1) dosimetry sensors (metal foils, nuclear track detectors, wires, crystals, etc., supplied by the customer or by NBS) are irradiated to a certified neutron fluence in a ^{252}Cf spontaneous fission neutron field (for short lived reaction products, the samples are irradiated to near saturation and the fluence rate specified), and 2) the irradiated dosimetry sensors are shipped to the customer, followed by a test report stating the fluences and associated uncertainties. For a typical activation sensor, the customer measures the activity of the irradiated sensor with his own sensor system and corrects the activity to obtain the activity at the end of the irradiation (EOI). This activity is then corrected for scattering effects (see section 2.1) to obtain the free-field activity at EOI. For simple decay, this free-field activity of the neutron fluence standard in disintegrations per second is related to the free-field neutron fluence by the activation equation:

$$A = \lambda \cdot C \cdot N \cdot \bar{\sigma} \cdot \Phi \qquad (1)$$

where: A = free-field activity at EOI (Bq),

λ = decay constant of the reaction product (s^{-1}),

C = decay correction factor (dimensionless) given in the test report (this corrects for activity lost during the irradiation),

N = number of atoms of the target isotope,

$\bar{\sigma}$ = fission-spectrum-averaged reaction cross section (cm^2),

and Φ = free-field fission neutron fluence (n/cm^2) given in the test report.

The commonly employed reaction rate probability (experimentally, the saturated specific activity) is given by

$$\frac{A}{\lambda CTN} = \bar{\sigma} \cdot \phi$$

where: $\frac{A}{\lambda CTN}$ = reaction rate probability (disintegrations per second per nucleus),

$\phi = (\Phi/T)$, the effective neutron fluence rate (n/cm^2 s) or flux density, and

T = the length of the irradiation given in the test report.

I.a. GENERIC DESCRIPTION

A standard ^{252}Cf neutron field consists of neutrons from the spontaneous fission of ^{252}Cf with little or no energy degradation. The median energy of the spectrum is 1.68 MeV, with 98% of the neutrons between 0.1 MeV and 8 MeV (see fig. 1). Neutron fluence rates in the range of 10^7 n/(cm^2 s) are obtained in isolated environments near small, intense ^{252}Cf fission sources. Neutron fluences are established in terms of neutron source strength, irradiation time, and source-sensor distance; no microscopic nuclear data or irradiation monitors are required. Certified free-field fluences of up to 10^{13} n/cm^2 may be obtained with uncertainties as low as ± 1.2% (1σ).

Measurements of the ^{252}Cf fission neutron spectrum and the closely related the ^{235}U fission neutron spectrum are extensive and well-documented [1]. These two standard neutron fields are therefore much better known than any other benchmark employed for reactor dosimetry calibration. Moreover, in the energy range above 2 MeV, many neutron fields in and around test and power reactors have fission-spectrum-like components.

I.b. AVAILABLE FISSION NEUTRON EXPOSURES

- fluence rate (5 cm from source) ~ 2×10^7 n/(cm^2 s)
- nominal maximum fluence 1×10^{13} n/cm^2
- accuracy of free-field fluence (1σ) ± 1.2%

I.c. CORRECTIONS FOR NEUTRON SCATTERING AT 5 CM POSITION

- Net perturbation for threshold sensor reaction rates: < (1.0 ± 0.8)% (1σ)
- Net perturbation for ^{235}U (n,f) sensor reaction rate: (1.2 ± 0.4)% (1σ)

I.d. CONFIGURATION AND CHARACTERISTICS [2,3]

The two irradiation locations at NBS are distinguished by the degree to which the neutron source is isolated from environmental neutron return:

Location A: Large room with thick walls and open ceiling. Source 2.2 meters above floor, nearest wall 4.1 meters.

Location B: Corner area of room with thin walls and ceiling. Source 2.8 meters above floor, nearest wall 4.3 meters.

At both locations a lightweight source-sensor assembly is available for irradiation of passive and active neutron sensors. The source-sensor assembly with six activation sensors mounted on three axes is shown in figure 2. Location A is currently being used for calibration of health physics type dosimeters. All activation foil irradiations are now done at location B. All further site specific properties discussed in this report will apply to location B only, although the site specific properties are very similar.

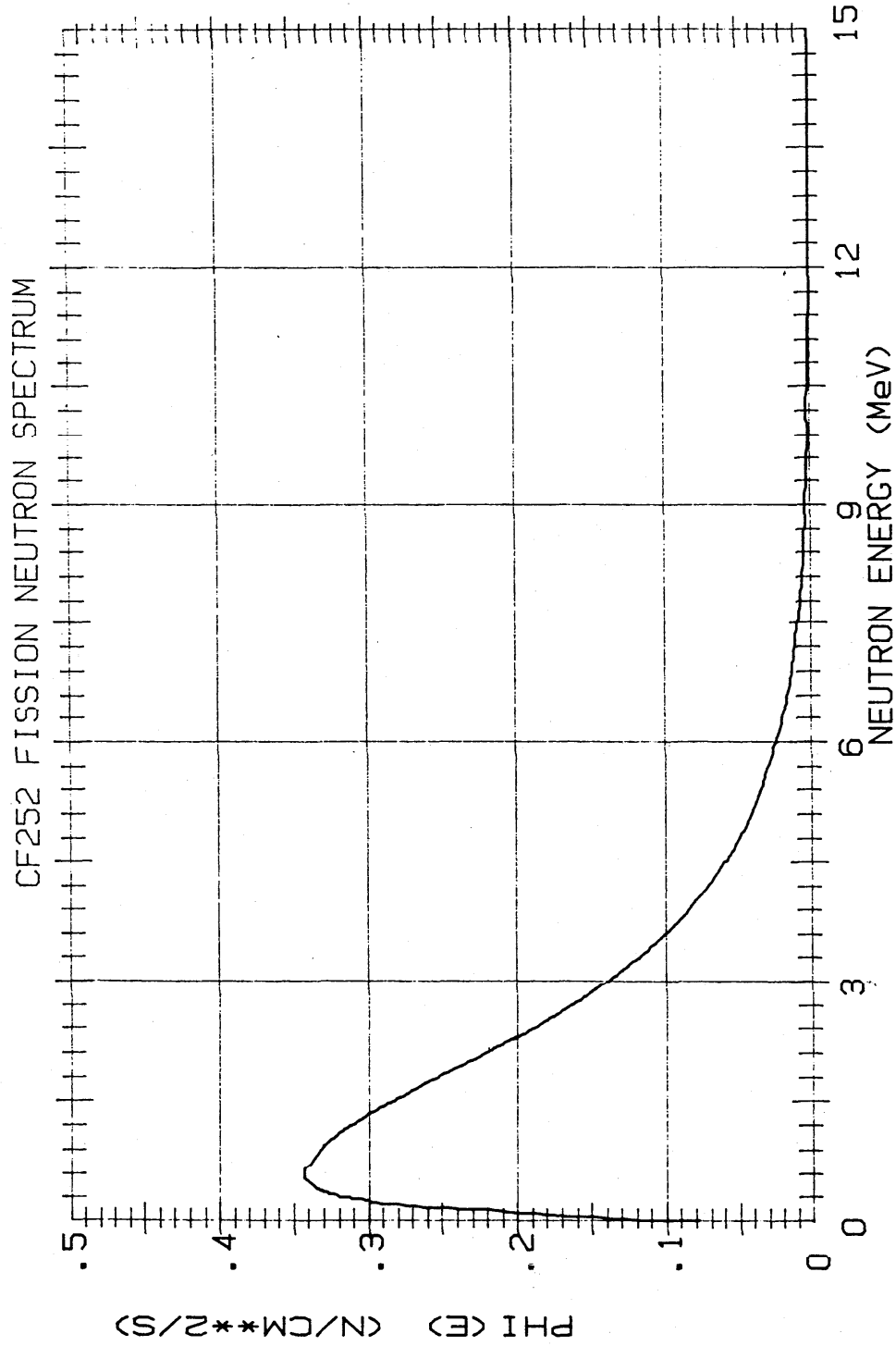

Figure 1. The NBS evaluation of the ^{252}Cf fission neutron spectrum.

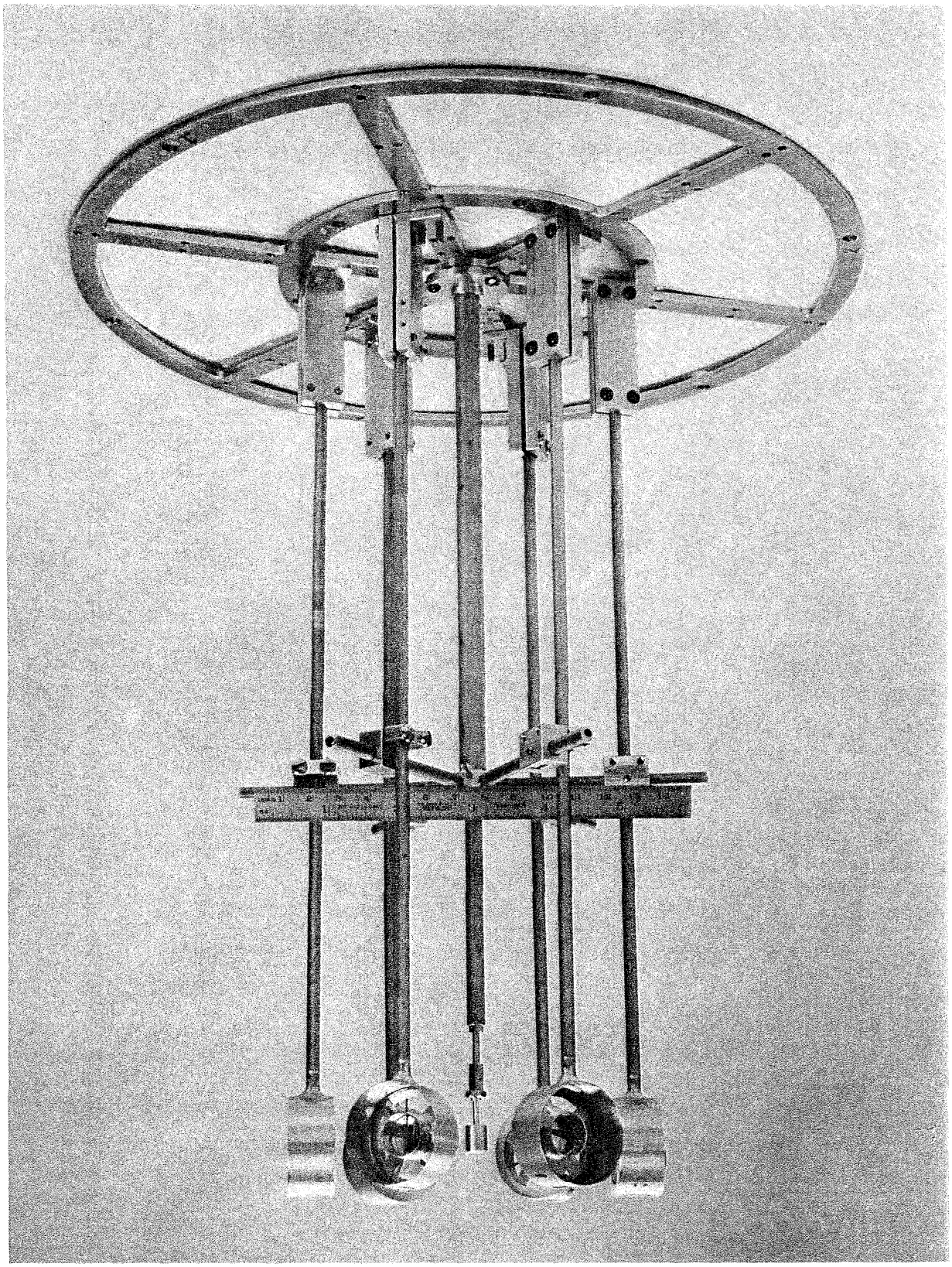

Fig. 2. Typical arrangement of six activation sensors around the ^{252}Cf source.

The californium source capsule, shown in figure 3, is made up of a disk-shaped ^{252}Cf deposit in an aluminum pellet encapsulated in a thin-walled stainless steel cylinder. The position of the californium deposit relative to the capsule surface is known to a maximum displacement ± 0.5 mm based on constraints of fabrication and x-ray photographs. Current capsule designs feature a short attachment stem in place of the first few cm of the source guide tube; the ^{252}Cf deposit enclosure is unchanged.

The traceability path of NBS neutron field strengths begins with NBS-I, the U.S. national fast neutron source standard [4]. This source consists of a beryllium sphere, 4.0 cm in diameter, at the center of which is placed one gram of radium in the form of $RaBr_2$. The radium is enclosed in platinum-iridium capsules of 0.2 mm wall thickness. This source has been calibrated by several independent methods [5-7] and the source strength is now established as $1.243 \times 10^6 \pm 0.85\%$ (1σ). The NBS ^{252}Cf spontaneous fission neutron source strengths are obtained by measuring their ratio to NBS-I in a manganese sulfate bath [8]. The uncertainty of the source strength obtained in this fashion is ± 1.1% (1σ).

TABLE 1. ^{252}Cf FISSION NEUTRON FIELD PARAMETERS AND UNCERTAINTY COMPONENTS

Free-field fission neutron fluence rate (5-cm distance; source 6×10^9 n/s)	2×10^7 n/(cm^2 s)
Source decay rate	2.2% / month
Typical maximum fluence	10^{13} n/cm^2
Source capsule and support scattering correction (inelastic plus net elastic inscatter)(see Table 2)	< 1.3%
*Gamma-ray exposure at 5 cm (2.8 yr after separation)	~ 170 R/h
Uncertainties (1σ) in the free-field fission neutron fluence due to:	
Source strength	± 1.1%
Source capsule and support scattering	± 0.4%
Distance measurements (typical for compensated-beam geometry)	± 0.2%
Quadrature sum:	± 1.2%

*This estimate does not include x-rays which the light-weight encapsulation may not absorb.

Neutron field parameters and uncertainty estimates for a nominal 5 cm source-to-sensor distance are given in Table 1. The source strength uncertainty of ± 1.1% dominates the composite uncertainty of ± 1.2% for the

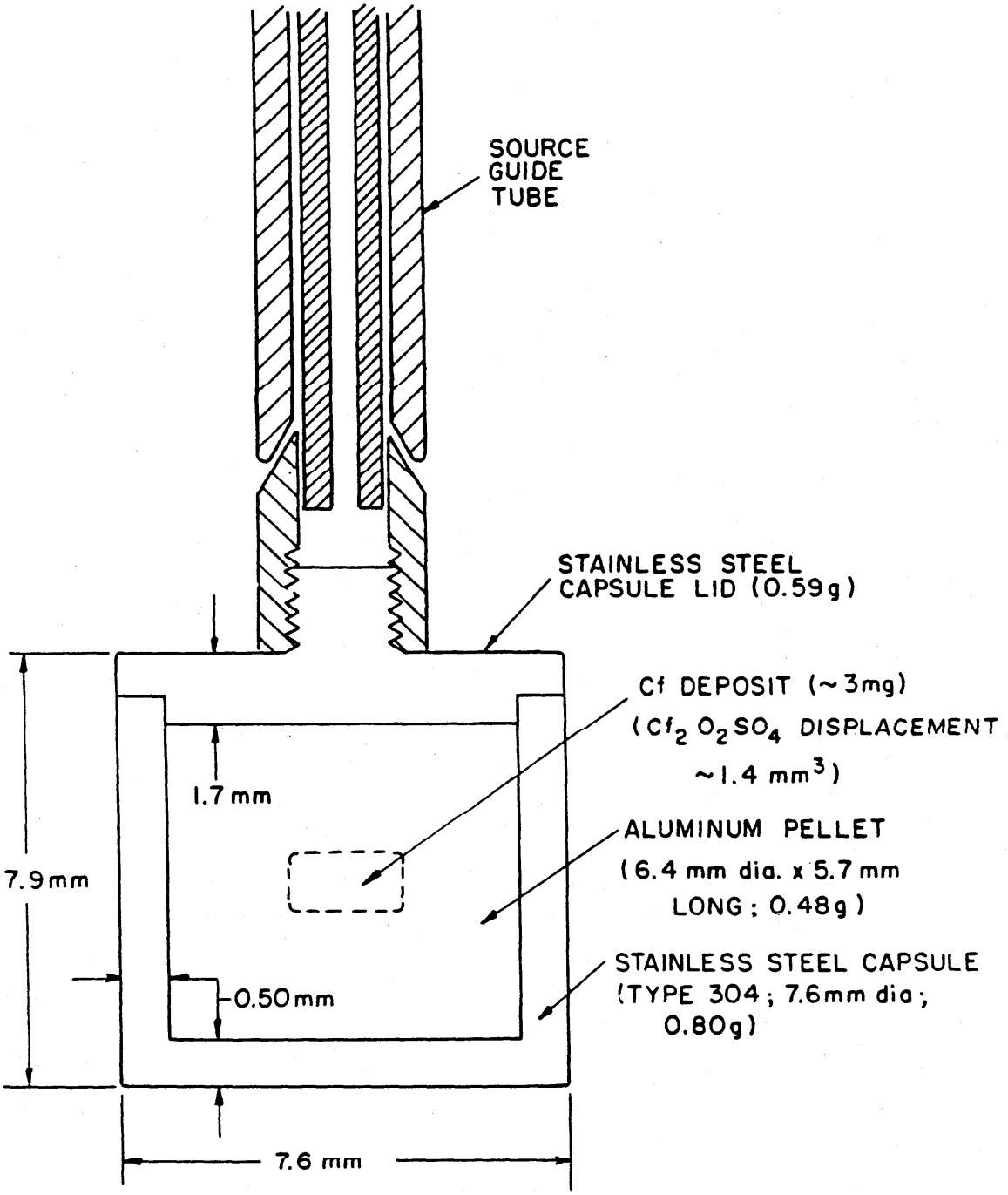

Fig. 3 Scattering materials in the low-mass ^{252}Cf fission neutron source capsule.

free-field fission neutron fluence. The irradiation geometry shown in figure 2 is termed compensated-beam geometry and refers to the experimental practice of placing sensors of similar sensitivity in pairs on opposite sides of the source, and nearly equidistant from it. The geometric mean of the responses of the two sensors in this case can be expressed in terms of a mean neutron fluence which is a function of sensor separation with very little dependence on source position.

It can be shown that the percent uncertainty due to source displacement on axis with the sensors is $(\Delta r^2/r^2-\Delta r^2) \times 100$.

For a 0.5-mm displacement where r = 50 mm, the uncertainty is 0.01%. For displacement perpendicular to the displacement axis, the maximum percent decrease in the fluence rate is $(2\cdot[\sqrt{r^2+\Delta x}-r]/r) \times 100$. For a perpendicular displacement of 0.5 mm where r = 50 mm, the fluence rate is decreased by 0.1%. The uncertainty in source position therefore does not contribute to the overall uncertainty. The principal percent uncertainty due to positioning can be shown to be equal to $(2\Delta d/d) \times 100$ where d is the measured distance between samples and Δd is the uncertainty in measurement. For the Boice C786 coordinate measurement machine, the measuremnt accuracy for linear distances is ± .005 mm (1σ). For our sample holders, which are not perfectly rigid, the reproducibility of measurements is ± .04 mm (1σ). For the previously discussed case of a sample 50 mm from the source (100 mm sample separation), the uncertainty from positioning uncertainty is ± .08% (1σ).

Scattering Corrections. Neutron fluence perturbations attributable to room return and scattering in the source capsule and support structures are given in Table 2. Corresponding sensor perturbations are listed for a threshold and a low-energy response neutron sensor. The room-return component is estimated from calculations matched to response-versus-distance measurements in which the source is moved while the sensors are kept at their normal 5 cm position [9]. For sensors with low-energy response, a large cadmium basket is placed around the source-sensor assembly to absorb room-return thermal neutrons. The fission neutron return from this basket is less than 0.1%.

Three types of calculations are employed for estimating free-field neutron fluences: (1) discrete ordinates calculation of a spherical cavity in concrete to obtain the albedo from boundaries; (2) geometry and simple energy degradation factors calculated for single-scatter events in individual pieces to obtain corrections for scattering in source capsule and support structures; and (3) calculations based on published analytical formulations involving a simple scattering kernel to obtain estimates of air scattering. Air scatter contributions are less than 0.1% for source-sensor distances of 5 cm and < 0.5% for distances up to 15 cm.

The degree of difficulty in correcting a free-field neutron fluence for scattering in a sensor depends upon the mass, arrangement, and material of the sensor. The NBS has codes available that provide correction factors for isotropic scattering of neutrons in lightly constructed sensors with cylindrical symmetry. More extensive Monte-Carlo calculations have been carried out for special sensors, specifically the NBS double fission chamber.

TABLE 2. TYPICAL NEUTRON FLUENCE AND SENSOR RESPONSE PERTURBATIONS DUE TO NEUTRON SCATTERING (5-cm DISTANCE)

Neutron fluence perturbations: (above 0.4 eV)	
room return	0.1%
source capsule scattering	0.8%
support structure scattering	0.4%
air scatter	< 0.1%
Net perturbation of sensor response due to neutron scattering:[†]	
$^{235}U(n,f)$ sensor with cadmium cover:	
room return	(0.1 ± 0.01)%
source capsule	(0.8 ± 0.4)%
support structures	(0.3 ± 0.1)%
$^{238}U(n,f)$ threshold sensor:	
room return	(0.0 ± 0.01)%
source capsule*	(0.0 ± 0.5)%
support structures	(0.3 ± 0.3)%

[†]All uncertainties are 1σ.

*Source capsule perturbations depend markedly upon sensor threshold.

Multiple scattering in more massive sensors are difficult to estimate and often require auxiliary experiments. Generally, isolated ^{252}Cf fission neutron fields are most appropriate for accurate, uncluttered exposures with lightweight sensors.

II. NEUTRON FIELD CHARACTERIZATION

The energy spectrum of ^{252}Cf spontaneous fission neutrons is similar to that of ^{235}U and other fissionable materials. As such it provides a spectrum of neutrons characteristic of the driving source for most of nuclear energy. Because fission neutron spectra are similar and have been evaluated with the same methods, the ^{252}Cf spectrum is specified here together with the ^{235}U spectrum. A concise description of fission neutron spectra may be given in terms of a broad energy range,

	lower bound $E_p(p=0.99)$	median $E_p(p=0.5)$	upper bound $E_p(p=0.01)$
^{252}Cf	0.09 MeV	1.68 MeV	7.8 MeV
^{235}U	0.08	1.57	7.2

where p = fraction of the neutron spectra above E_p; and in a coarse seven-group display, $\phi(E)\Delta E$, as follows:

E limits:	0	0.25	0.8	1.5	2.3	3.7	8	12 MeV
$\phi(E)dE$ (^{252}Cf)	0.047	0.184	0.220	0.194	0.200	0.146	0.009	
$\phi(E)dE$ (^{235}U)	0.054	0.197	0.229	0.195	0.192	0.127	0.006	

A more detailed description of the two spectra is given in Section II.b.

Prompt fission gamma-rays and gammas from fission products contribute about equally to the total gamma-ray emission throughout the useful lifetime of the source. As indicated in Table 1, the neutron-to-gamma ratio is about 1.1×10^5(n/cm^2 s)per R/h.

II.a. NEUTRON FLUENCE (see also Section I.d)

The neutron emission rate for the singly-encapsulated sources employed at NBS are $\sim 6 \times 10^9$ n/s when fabricated. The sources are periodically calibrated in a $MnSO_4$ bath and the source strength is calculated for the day of the irradiation (the decay rate is $\sim$ 2.2% per month). For a neutron fluence rate of $\sim 2 \times 10^7$ n/(cm^2 s) at the 5-cm distance, the fluence for a one week exposure is $\sim 1 \times 10^{13}$ n/cm^2. Spatial gradients for uncollided fission neutrons are just those of a point source in free space: $(\Delta\phi/\phi) = (2\Delta r/r)$ Free-field fluence accuracy and sensor response perturbations for typical irradiation arrangements are discussed in Section I.d.

In the case of compensated-beam geometry where nearly identical sensors are placed on opposite sides of the source and nearly equidistant from it, a mean neutron fluence rate may be defined,

$$\langle\Phi\rangle = \frac{S}{4\pi(r/2)^2}$$

S = source strength (n/s)

r = distance between sensors (cm),

which is proportional to the geometric mean of the sensor responses per atom:

$$\sqrt{\frac{D_1}{N_1}\frac{D_2}{N_2}} \approx a\ \sigma\ \langle\Phi\rangle$$

where D/N = sensor response per atom

σ = reaction cross section

a = efficiency factor (same for both sensors).

[The exact form of the above expression departs from this approximation by < 0.2% for D_1/N_1 within 20% of D_2/N_2.]

A small anisotropy results from the cylindrical shape of the fission source capsule. The ratio of the fluence rate averaged over all directions to the fluence rate along a direction perpendicular to the capsule axis is 0.992 when the sensors do not subtend an angle of more than about 20° at the source.

II.b. NEUTRON SPECTRUM

II.b.1. Measurement [10], [11]. Fission neutron spectra are most accurately determined by laboratory measurement. (See, however, Refs. [12], [13], and earlier work cited therein.) From 1952 to the present, well-documented measurements of ^{252}Cf and ^{235}U fission neutron spectra have employed many different techniques of neutron spectrometry [14]. Fission spectra are thus the most widely studied fast-neutron energy distribution in existence. The large body of documented fission spectrum data was evaluated at NBS in 1975 [11]. The evaluation included an estimate of the spectrum uncertainties in multigroup format based on the spread of the various data sets. The ^{252}Cf and ^{235}U fission spectrum shapes recommended in the evaluation are based on the sixteen selected measurements listed in Table X-3 of Ref. [1].

II.b.2. Evaluated Spectrum. The NBS fission spectrum evaluation has been chosen for presentation in this document. The ^{252}Cf spectrum is defined up to 20 MeV by means of a reference Maxwellian, M(E), modified by four piecewise-continuous linear segments below 6 MeV plus one exponential segment above 6 MeV. The reference Maxwellian for the ^{252}Cf fission spectrum is

$$M(E) = 0.667 \sqrt{E} \cdot \exp(-1.5E/2.13) \quad , \quad E \text{ in MeV},$$

with adjustment functions $\mu(E)$ as follows:

Energy Interval (MeV)	$\mu_{Cf}(E)$
0.0 - 0.25	1 + 1.20E - 0.237
0.25 - 0.8	1 - 0.14E + 0.098
0.8 - 1.5	1 + 0.024E - 0.0332
1.5 - 6.0	1 - 0.0006E + 0.0037
6.0 -20	1.0 exp[-0.03(E-6.0)]

The evaluated spectrum then is given by $\chi(E) = \mu(E) \cdot M(E)$. A 45-group tabulation, normalized to 1, of the evaluated ^{252}Cf fission spectrum is given in Table 3 (and plotted in Figure 1) along with the corresponding ^{235}U evaluated spectrum for comparison purposes. For other energy group structures this tabulation may be interpolated as recommended in the footnote of Table 3.

TABLE 3. EVALUATED FISSION NEUTRON SPECTRA FOR ^{252}Cf AND ^{235}U IN 45-GROUP FORMAT

	^{252}Cf		^{235}U	
Lower Energy Boundary (MeV)	Group Fluence $\phi \cdot \Delta E$	Cumulative[a] Fluence to Lower Boundary	Group Fluence $\phi \cdot \Delta E$	Cumulative[a] Fluence to Lower Boundary
0.0	0.0039	1.0000	0.0048	1.0000
0.05	0.0074	0.9961	0.0088	0.9952
0.1	0.0219	0.9887	0.0249	0.9864
0.2	0.0140	0.9668	0.0154	0.9615
0.25	0.0152	0.9528	0.0166	0.9461
0.3	0.0323	0.9377	0.0350	0.9295
0.4	0.0337	0.9054	0.0363	0.8945
0.5	0.0343	0.8717	0.0367	0.8582
0.6	0.0343	0.8374	0.0365	0.8215
0.7	0.0338	0.8031	0.0358	0.7851
0.8	0.0664	0.7693	0.0697	0.7493
1.0	0.0641	0.7029	0.0668	0.6796
1.2	0.0608	0.6388	0.0629	0.6128
1.4	0.0290	0.5780	0.0298	0.5499
1.5	0.0279	0.5490	0.0286	0.5201
1.6	0.0527	0.5211	0.0536	0.4915
1.8	0.0484	0.4684	0.0487	0.4379
2.0	0.0442	0.4200	0.0441	0.3892
2.2	0.0206	0.3758	0.0204	0.3451
2.3	0.0196	0.3552	0.0193	0.3247
2.4	0.0364	0.3356	0.0356	0.3054
2.6	0.0328	0.2992	0.0319	0.2698
2.8	0.0296	0.2664	0.0284	0.2379
3.0	0.0503	0.2368	0.0477	0.2095
3.4	0.0310	0.1865	0.0290	0.1618
3.7	0.0413	0.1555	0.0378	0.1328
4.2	0.0253	0.1142	0.0227	0.09503
4.6	0.0200	0.0889	0.0176	0.07233
5.0	0.0190	0.0689	0.0164	0.05473
5.5	0.0140	0.0499	0.0118	0.03833
6.0	0.0102	0.0359	0.00831	0.02653
6.5	0.00734	0.0257	0.00573	0.01822
7.0	0.00527	0.01839	0.00394	0.01249
7.5	0.00378	0.01312	0.00271	0.00855
8.0	0.00270	0.00934	0.00186	0.00584
8.5	0.00193	0.00664	0.00127	0.00398
9.0	0.00137	0.00471	0.00087	0.00271
9.5	0.00098	0.00334	0.00059	0.00184
10.0	0.00118	0.00236	0.00068	0.00125
11	0.00059	0.00118	0.00031	0.00057
12	0.00030	0.00059	0.00014	0.00026
13	0.00015	0.00029	0.00007	0.00012
14	0.00011	0.00014	0.00004	0.00005
16	0.00003	0.00003	0.00001	0.00001
18	0.00000	0.00000	0.00000	0.00000

For interpolation up to 10 MeV, use the shape function $E^{\frac{1}{2}} \cdot \exp(-aE)$; $a = 0.70$ for ^{252}Cf; $a = 0.76$ for ^{235}U.

[a] Spectrum fraction above lower energy bound.

Spectrum uncertainties in the evaluation are based on standard deviations from the mean of subsets of measured data from the final adjusted Maxwellian. This estimate, carried out in a seven group structure, has been made for a 1σ uncertainty. Results are given in Table 4.

TABLE 4. UNCERTAINTY ESTIMATES FOR ^{252}Cf AND ^{235}U EVALUATED FISSION NEUTRON SPECTRA

Energy Boundaries	^{252}Cf (Spontaneous Fission)		^{235}U (Thermal-Neutron-Induced Fission)	
	Group Fluence $\phi \cdot \Delta E$	Uncertainty 1σ (%)	Group Fluence $\phi \cdot \Delta E$	Uncertainty 1σ (%)
0.0				
	0.047	±13	0.054	±16
0.25				
	0.184	± 1.1	0.197	± 4.1
0.8				
	0.220	± 1.8	0.229	± 3.0
1.5				
	0.194	± 1.0	0.195	± 3.1
2.3				
	0.200	± 2.0	0.192	± 2.0
3.7				
	0.146	± 2.1	0.127	± 4.8
8				
	0.0087	± 8.5	0.0056	± 5.3
12				
	(0.00058)*		(0.00026)*	
20				

*Above 12 MeV, there is insufficient data to obtain accurate fluence estimates.

III. DETAILED OPERATING PROCEDURES

Assembly and Irradiation of Sensors

1) If the sensors are unmarked, they should be marked with a permanent identifier at this time, preferably by scribing. Sensors should be cleaned with acetone and weighed after drying. The weighing accuracy should be 0.2% or better. Masses should be recorded on the worksheet and on the plastic bag containing the sensor. Although the mass is not necessary to specify the fluence, it is often useful for the calculation of other parameters, e.g., scattering.

2) The sensors should be mounted in the lightweight holders ("spiders") and subsequently in the mounts. Sensors are always mounted in pairs, on the same axis and on opposite sides of the source. The positions of the sensors should be recorded on the worksheet and in the notebook.

3) Have the shops measure the separation distance of the sensor pairs on the coordinate measuring machine. This should be a direct measurement of face-to-face separation when possible. Record all parameters, so that a final center-to-center separation can be obtained. A notebook sketch of the measurement procedure is advisable. The current shop contact is Gene Morgan at extension 2448.

4) Arrange time at the Time-of-Flight-Facility (TOFF) with Linac Operations, CW Operations, and the Neutron T-O-F group. Notify Health Physics.

5) Mount the source and sensor holder ring at the TOFF and connect the desired source to the beaded chain. For some irradiations, the cadmium shield will be necessary and should be mounted at this time. Lock the road gate with padlock and put up any required ropes and signs (requirements vary with source strength and length of irradiation). Be sure that there are no persons at the TOFF end stations or in the blockhouse. Turn on the red outside warning lights.

6) Check time marking instrument with telephone time (844-2525) or Naval Observatory Time (653-0258). Raise the source while noting the time. Note that the source up warning light has come on. After securing the source, record the time in notebook and source activities log. Visually inspect the source with telescope when possible.

7) On leaving, close and padlock locking gate 11; put up rope and sign in hallway; close but do not lock door at top of stairwell; make sure that Cf source up sign is showing; leave sign with your name and phone numbers showing; leave appropriate signs on external door and in Linac control room.

8) At the end of the irradiation (EOI) , again check time. Lower the source while noting the time. Be sure that the source up warning light goes out! Record the EOI time in notebook and source activities log. Later be sure that these times are recorded on the worksheets. Secure source, unlock road gate, remove appropriate ropes and signs. Remove sensor holder.

9) Remove sensors from holders and package individually in labeled plastic containers. Store in shielded card file in Building 235, Room B123.

IV. FLUENCE CALCULATIONS AND UNCERTAINTIES

IV.a. FLUENCE AND FLUENCE RATE CALCULATIONS

1) Obtain the source strength, S, from output of the FORTRAN program CFYLD (See Appendix A). This program can be run for a specific day or can list in tabular form the source strength for consecutive days. The source strength of the most recent manganous sulfate bath measurements of the source is entered and the program uses the ^{252}Cf half life to calculate the source strength on the days of interest.

2) The fluence rate, ϕ, averaged over the surface of the sensor is obtained from the equation:

$$\phi = \frac{S}{4\pi R^2} \left\{\frac{R^2}{\rho^2} \ln[1+\rho^2/R^2]\right\}$$

where R is the average source to sensor distance and ρ is the radius of the sensor, typically 6.35 mm.

This equation has been programmed on a TI SR-52 desk calculator.

3) Run the program CFGFAC (See appendix B) to obtain the corrections for decay within irradiation of both the Cf source and the activation products of the sensors. This program will correct for interruptions during the irradiation.

4) Since the scattering corrections differ for each reaction, scattering corrections have been done on an ad hoc basis. Table 5 lists these corrections for a few of the more common sensors.

5) Fill out test report (See appendix C). All reduced data should be kept in the fluence standard file in room A156 of Building 235.

IV.b. UNCERTAINTIES

Table 1 gives uncertainty estimates for free-field fluence irradiations at the Cf facility. The uncertainty in the distance measurement can be considered a random uncertainty and part of the source strength uncertainty is also random. The ^{252}Cf source strength is determined by a ratio measurement to NBS-I and therefore the uncertainty of the NBS-I source strength is systematic and the scattering corrections are also systematic, while the uncertainty of the ratio to NBS-I is random.

TABLE 5. NET SCATTERING CORRECTIONS FOR SOME COMMON SENSORS

Sensor	μ_{sc}	$\delta\mu_{sc}(\%)(1\sigma)$
$^{115}In(n,n')$	+ 0.007	0.5
$^{58}Ni(n,p)$	- 0.001	0.5
$^{54}Fe(n,p)$	- 0.003	0.5
$^{103}Rh(n,n')$	+ 0.013	0.5

V. BENCHMARK REFERENCING

A benchmark neutron field is a well-characterized neutron field which will provide an accurate neutron fluence for validation or calibration of experimental techniques and methods. For purposes of fluence transfer to a study field, the benchmark field should have a neutron energy spectrum that is similar to the dosimetry environment to be monitored. Exposing identical sensors in both the benchmark field and the study field and hence determining reaction rate ratios, eliminates a number of systematic errors (detector efficiency, gamma attenuation, absolute cross section values, etc.) in the determination of the unknown fluence.

Exposure of a dosimetry sensor to a known neutron fluence, χ_{Cf}, in a ^{252}Cf fission spectrum establishes a calibration factor,

$$G = \frac{[\text{observed sensor response}]}{[\chi_{Cf}\ \text{neutron fluence}]} \qquad (1)$$

which may be used in conjunction with the calculated fission-spectrum-averaged cross section, σ_χ, for the sensor reaction to obtain a consistent benchmark calibration. The cross section, σ_χ, should be calculated with (1) the fission spectrum shape associated with the a priori spectrum (e.g., the source spectrum if the a priori spectrum is from a neutron transport calculation); and (2) the same energy dependent reaction cross section, R(E), used to calculate the spectrum averaged cross sections of the study field.

[If the absolute detection efficiency for the dosimetry measurement method has been established (e.g., a good resolution fission chamber with known deposit mass, or a calibrated gamma counting system and known reaction parameters, branching ratios, isotopic abundances, fission yields etc.), then the "observed sensor response" in Eq. (1) becomes the reaction probability, and G is an observed spectrum-averaged cross section for the fission spectrum.] The calibration factor may then be applied to the investigation of a study spectrum.

By irradiating samples identical to those irradiated in the ^{252}Cf field, the fluence of the study field, Φ_{study}, can be obtained as a simple ratio of sensor responses, D, and spectrum averaged cross sections, $\bar{\sigma}$, so that:

$$\frac{D_{study}}{D_\chi} = \frac{\bar{\sigma}_{study}\ \Phi_{study}}{\bar{\sigma}_\chi\ \Phi_\chi} \qquad (2)$$

In practice, (D_{study}/D_χ) is simply the time history corrected counting rate ratio of the sample irradiated in the unknown field to that irradiated in the Cf field. Uncertainties due to isotopic abundances, nuclear parameters, detector efficiencies, etc., disappear. The cross section ratios must be calculated, but the absolute value becomes unimportant. The value of Φ_χ is that given in the test report and one then solves equation (2) for Φ_{study}.

VI. SAFETY

The neutron sources used for these irradiation services range in source strength from 1 to 5 × 10^9 n/s. While these sources are not lethal, mishandling can result in radiation exposures well over the legal limit for radiation exposures. The sources should be handled remotely. If close handling is necessary (e.g., when remounting the source on its holder), the source should be placed in the water tank. New irradiation setups should be first practiced with the dummy sources expressly designed for this purpose. Personnel dosimeters must be worn at all times and a "chirper" is recommended when close handling is necessary. Unattended irradiations require that the area be locked to prevent accidental entry. Ropes with warning signs should be placed at the 2.5 mR radiation level in accordance with NRC regulations. Health Physics will assist in determining the rope locations if necessary.

The irradiated samples will be radioactive, but at a very low level (usually a few nanocuries). There is therefore no hazard from the samples, but all radioactive shipping regulations must still be followed.

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APPENDIX A

```
C     CFYLD--CALIFORNIUM YIELD 6/26/82.
      DIMENSION KCOM(66)
      CHARACTER*1 REP
      CHARACTER*4 A
C      THIS PROGRAM USES 2.645 YR HALFLIFE FOR 252CF AND NU=3.766.
C      FOR 250CF IT USES 13.2 YRS AND 3.53. FOR 254CF IT USES 61.9 DAYS
C      AND 3.93. NU RATIOS ARE FROM N. SC. AND ENGIN. 43,54(1971).
C      RATIO OF ALPHA PARTICLE TO SPONTANEOUS FISSION RATE IS FROM
C      J. INORG. NUCL CHEM 27,33(1965)., IE., 31.3 AND 1260.
C      LAMBDA ALPHA/(LAMBDA SP FISSION) = 1/(1260+1) AND 1/(31.3+1)
C      SP FISSION/ALPHA FOR 254 IS 0.997
C
      DATA CF0HF/13.2/, CF2HF/2.645/, CF4HF/61.9/, CF0NU/3.53/,
     &CF2NU/3.766/, CF4NU/3.93/
      A='XXXX'
    1 FORMAT(F14.0)
    2 FORMAT(6X,A4,I2,',',I4,1P4E12.4)
    3 FORMAT(I5)
    4 FORMAT(/7X,
     &'GIVE THE DAY,MONTH, AND YEAR OF ISOTOPIC ANALYSIS ')
    5 FORMAT(7X,'IN F FORMAT THE ATOM PERCENT COMPOSITION OF CF250,
     & 252, AND 254 =')
    6 FORMAT(7X,'GIVE THE DAY, MONTH, AND YEAR OF CALIBRATION ')
    7 FORMAT(/7X'   DATE  '3X'250/TOTAL'3X'254/TOTAL'3X
     &'DECAY'7X'N/SEC'/)
    8 FORMAT(7X'GIVE THE DAY, MONTH, AND YEAR FOR THE END OF THE '
     &/7X'TABULATION '/)
    9 FORMAT(7X'IN F OR G FORMAT WHAT IS THE CALIBRATION
     &ON THAT DATE ?')
   10 FORMAT(7X'GIVE THE DAY, MONTH, AND YEAR FOR THE START OF THE '
     &/7X'TABULATION IN I FORMAT'/)
   11 FORMAT(/7X,
     &'THE HALF LIVES USED IN YEARS FOR CF250 AND CF252,',/
     &7X,'          AND IN DAYS FOR CF254 = '/)
   12 FORMAT(7X'TYPE THE TITLE INFORMATION '/)
   13 FORMAT(66A1)
   14 FORMAT(7X,66A1)
   15 FORMAT(7X,3F14.6)
   16 FORMAT(/7X,'THE VALUES USED FOR NUBAR = '/)
   18 FORMAT(/7X'GIVE A CARRIAGE RETURN AFTER EACH DATA ENTRY')
   19 FORMAT(7X'THE ATOM PERCENT COMPOSITION OF CF250,
     & 252 AND 254 =')
   20 FORMAT(7X'WHENEVER YOU REQUIRE NO FURTHER OUTPUT, ANSWER THE NEXT'
     &/7X'REQUEST FOR DAY, MONTH, AND YEAR WITH THREE CARRIAGE RETURNS.'
     &'  '/)
   21 FORMAT(/7X'THE OUTPUT WILL BE ON FOR002.DAT.')
   22 FORMAT(A1)
   23 FORMAT(7X'GIVE THE DAY, MONTH, AND YEAR '/)
   24 FORMAT(7X'DO YOU WISH TO INPUT A SPECIFIC DATE (Y/N) ? '/)
      OPEN(2,STATUS='NEW')
      WRITE(7,18)
      WRITE(7,4)
      READ(5,3)ID,M,IYR0
      CALL IDAYS(ID,M,IYR0,IDAY0)
      WRITE(7,5)
```

```
      READ(5,1)XO50,XO52,XO54
C     NORMALIZE THE NEUTRON EMMITTERS TO 1.
      SN=XO50+XO52+XO54
      X50=XO50/SN
      X52=XO52/SN
      X54=XO54/SN
      WRITE(7,6)
      READ(5,3)ID,M,IYR1
      CALL IDAYS(ID,M,IYR1,IDAY1)
      CALL LPDAYS(IYR1,IYR0,ITL)
      WRITE(7,9)
      READ(5,1)CA
      IT=(IYR1-IYR0)*365+(IDAY1-IDAY0)+ITL
C *    DEBUG OPTIONS FOLLOW
C      WRITE(7,400)IDAY0,IYR0,IDAY1,IYR1,ITL,IT
C  400 FORMAT(7X,'IDAY0,IYR0,IDAY1,IYR1,ITL,IT',/7X,8I6)
      T=FLOAT(IT)
      A500=0.693147*CF0NU*X50/(CF0HF*365.25*1261.)
      A520=0.693147*CF2NU*X52/(CF2HF*365.25*32.3)
      A540=0.693147*CF4NU*0.997*X54/CF4HF
      AAT=A520+A500+A540
      AT0=1.
      F00=A500/AAT
      F40=A540/AAT
      A50=A500*EXP(-0.693147*T/(CF0HF*365.25))
      A52=A520*EXP(-0.693147*T/(CF2HF*365.25))
      A54=A540*EXP(-0.693147*T/CF4HF)
      ATT=A50+A52+A54
      AT1=ATT/AAT
      F01=A50/ATT
      F41=A54/ATT
      AT=AT1
      CN0=CA/AT1
      WRITE(7,12)
      READ(5,13)KCOM
      WRITE(2,14)KCOM
      WRITE(2,19)
      WRITE(2,15)XO50,XO52,XO54
      WRITE(2,11)
      WRITE(2,15) CF0HF, CF2HF, CF4HF
      WRITE(2,16)
      WRITE(2,15) CF0NU, CF2NU, CF4NU
      WRITE(2,7)
      CALL MODAY(IYR0,IDAY0,A,ID)
      WRITE(2,2)A,ID,IYR0,F00,F40,AT0,CN0
      CN1=CN0*AT1
      CALL MODAY(IYR1,IDAY1,A,ID)
      WRITE(2,2)A,ID,IYR1,F01,F41,AT1,CN1
      WRITE(7,20)
   60 WRITE(7,24)
      ACCEPT 22,REP
      IF ( (REP.EQ.'Y').OR.(REP.EQ.'y') ) GO TO 70
      WRITE(7,10)
      READ(5,3)ID,M,IYR2
      IF(ID.EQ.0)GO TO 200
      CALL IDAYS(ID,M,IYR2,IDAY2)
      WRITE(7,8)
```

```
      READ(5,3)ID,M,IYR3
      CALL IDAYS(ID,M,IYR3,IDAY3)
      CALL LPDAYS(IYR3,IYR2,ITL)
      N=(IYR3-IYR2)*365+IDAY3-IDAY2+ITL+1
      GO TO 80
   70 N=1
      WRITE(7,23)
      READ(5,3)ID,M,IYR2
      IF(ID.EQ.0)GO TO 200
      CALL IDAYS(ID,M,IYR2,IDAY2)
   80 CALL LPDAYS(IYR2,IYR0,ITL)
      IT2=(IYR2-IYR0)*365+IDAY2-IDAY0+ITL
C *    DEBUG OPTIONS FOLLOW
C      WRITE(7,500)IDAY0,IYR0,IDAY1,IYR1,IDAY2,IYR2,ITL,IT2
C  500 FORMAT(7X,'IDAY0,IYR0,IDAY1,IYR1,IDAY2,IYR2,ITL,IT2',/7X,8I6)
      T=FLOAT(IT2)
      DO 100 I=1,N
      A50=A500*EXP(-0.693147*T/(CF0HF*365.25))
      A52=A520*EXP(-0.693147*T/(CF2HF*365.25))
      A54=A540*EXP(-0.693147*T/CF4HF)
      ATT=A50+A52+A54
      AT1=ATT/AAT
      F01=A50/ATT
      F41=A54/ATT
      CN1=CN0*AT1
      CALL MODAY(IYR2,IDAY2,A,ID)
      IF ( (REP.EQ.'Y').OR.(REP.EQ.'y') ) THEN
         WRITE(7,2)A,ID,IYR2,F01,F41,AT1,CN1
      END IF
      WRITE(2,2)A,ID,IYR2,F01,F41,AT1,CN1
      IDAY2=IDAY2+1
      IF(IDAY2.LT.366)GO TO 100
      IF(IDAY2.GT.366)GO TO 95
      IMOD=MOD(IYR2,4)
      IF(IMOD.NE.0)GO TO 95
      GO TO 100
   95 IDAY2=1
      IYR2=IYR2+1
  100 T=T+1.
      GO TO 60
  200 CLOSE(2)
      WRITE(7,21)
      STOP
      END
C
C
      SUBROUTINE IDAYS(ID,M,IYR,IDAY)
C      THIS SUBROUTINE CALCULATES THE JULIAN CALENDAR DAY OF THE YEAR
C
    1 FORMAT(7X'YOU TYPED MONTH AND DAY. TYPE DAY AND MONTH '/)
    3 FORMAT(I5)
      IF(IYR.LT.1900)IYR=IYR+1900
      IF(M.GT.12)WRITE(7,1)
      IF(M.GT.12)READ(5,3)ID,M
      IY=MOD(IYR,4)
C      FOR MONTHS AFTER FEBRUARY
      X=2.0
```

```
C       FOR JANUARY AND FEBRUARY
        IF(M.LE.2)X=0.0
C       FOR MONTHS AFTER FEBRUARY IN LEAP YEARS
        IF(M.GT.2.AND.IY.EQ.0)X=1.0
        Y=(M-1)*30.57+0.5+ID-X
        IDAY=INT(Y)
        RETURN
        END
C
C
        SUBROUTINE LPDAYS(I2YR,I1YR,ITL)
C
C       THIS SUBROUTINE CALCULATES THE LEAP YEAR DAYS BETWEEN I1YR
C       AND THE YEAR PRECEDING I2YR.
C
        ITL=0
        IF(I1YR.GE.I2YR)GO TO 60
        DO 50 I=I1YR,I2YR-1
        IMOD=MOD(I,4)
C *      DEBUG OPTIONS FOLLOW
C      3 FORMAT(7X,'ITL,I, AND IMOD ARE ',3I8)
C        WRITE(7,3)ITL,I,IMOD
     50 IF(IMOD.EQ.0)ITL=ITL+1
     60 RETURN
        END
C
        SUBROUTINE MODAY(IYR,IDY,A,ID)
C
C       MODAY--THIS SUBROUTINE CALCULATES THE MONTH AND DAY OF THE YEAR
C       FROM THE JULIAN CALENDAR DAY.                          9/11/81
C
        CHARACTER*4 A0,AJAN,AFEB,AMAR,AAPR,AMAY,AJUN,AJUL,AAUG,ASEP,AOCT,
       1            ANOV,ADEC,A
        DATA A0/'****'/,AJAN/'JAN '/,AFEB/'FEB '/,AMAR/'MAR '/,
       &AAPR/'APR '/,AMAY/'MAY '/,AJUN/'JUN '/,AJUL/'JUL '/,AAUG/'AUG '/,
       &ASEP/'SEP '/,AOCT/'OCT '/,ANOV/'NOV '/,ADEC/'DEC '/
C
        IF(IDY.EQ.0) GO TO 1000
        A=A0
        J=1
C
C       IDY MUST NOT BE CHANGED BECAUSE IT IS RETURNED TO MAIN PGM
C
        IDAY=IDY
        IMOD=MOD(IYR,4)
        IF(IDAY-31)10,10,20
     10 A=AJAN
        ID=IDAY
        GO TO 1000
     20 IF(IDAY-59)30,30,40
     30 A=AFEB
        ID=IDAY-31
        GO TO 1000
     40 IF(IDAY.EQ.60.AND.IMOD.EQ.0)J=0
        IF(J-1)50,60,60
     50 A=AFEB
        ID=IDAY-31
```

```
      GO TO 1000
   60 IF(IMOD.EQ.0)IDAY=IDAY-1
C   ************************************************************
C       IDAY AT THIS LINE OF THE PROGRAM MUST BE 60 OR GREATER.
C       THIS SECTION OF THE PROGRAM DECIDES WHERE THE PROGRAM
C       SHOULD CONTINE IN ORDER TO AVOID MOST OF THE FOLLOWING 'IF'
C       STATEMENTS.
C
      K=(IDAY-6)/30
      GO TO (65,65,80,100,120,140,160,180,200,220,240,240) K
C
C   ************************************************************
   65 IF(IDAY-90)70,70,80
   70 A=AMAR
      ID=IDAY-59
      GO TO 1000
   80 IF(IDAY-120)90,90,100
   90 A=AAPR
      ID=IDAY-90
      GO TO 1000
  100 IF(IDAY-151)110,110,120
  110 A=AMAY
      ID=IDAY-120
      GO TO 1000
  120 IF(IDAY-181)130,130,140
  130 A=AJUN
      ID=IDAY-151
      GO TO 1000
  140 IF(IDAY-212)150,150,160
  150 A=AJUL
      ID=IDAY-181
      GO TO 1000
  160 IF(IDAY-243)170,170,180
  170 A=AAUG
      ID=IDAY-212
      GO TO 1000
  180 IF(IDAY-273)190,190,200
  190 A=ASEP
      ID=IDAY-243
      GO TO 1000
  200 IF(IDAY-304)210,210,220
  210 A=AOCT
      ID=IDAY-273
      GO TO 1000
  220 IF(IDAY-334)230,230,240
  230 A=ANOV
      ID=IDAY-304
      GO TO 1000
  240 A=ADEC
      ID=IDAY-334
      GO TO 1000
 1000 RETURN
      END
```

APPENDIX B

```
*       THIS VERSION OF CFGFAC IS DATED NOVEMBER 5,1985
DOUBLE PRECISION TMI(20),CMI(20)
DOUBLE PRECISION TOUT,DE,HE,TE,DS,HS,TS,TDI,THI,CDI,CHI,T,
     +FLU,PFLU,ALCF,TAU,ALAM,SATFR,B,GFAC,CFAC,BLAM
    1 FORMAT(I5)
    2 FORMAT(D16.0)
    3 FORMAT(7X,
     +/7X'THE G-FACTOR IN THIS PROGRAM IS THE SAME AS DEFINED IN'
     &/7X'THE NBS COMPENDIUM AND THE C-FACTOR IS G/LAMBDA AND IS '
     &/7X'DIMENSIONLESS. IN THIS PROGRAM THE SOURCE IS 252CF WITH A'
     +/7X'2.645 YEAR HALF-LIFE.'/)
    4 FORMAT(7X'HOW MANY TIMES WAS THE IRRADIATION INTERRUPTED?')
    5 FORMAT(7X'GIVE THE DAY, HOUR, AND MINUTE FOR THE END OF THE'
     +/7X'IRRADIATION IN D FORMAT, ONE TO A LINE.'/)
    6 FORMAT(7X'GIVE THE DAY, HOUR, AND MINUTE FOR THE START OF THE'
     +' IRRADIATION'/)
    7 FORMAT(7X'GIVE THE DAY, HOUR, AND MINUTE FOR THE FIRST '
     +'INTERRUPTION'/)
    8 FORMAT(7X'THE SAME FOR THE NEXT INTERRUPTION'/)
    9 FORMAT(7X'IN D FORMAT GIVE THE HALF-LIFE OF THE ACTIVATED '
     +'NUCLEI IN DAYS'/)
   10 FORMAT(7X'TYPE A 1 TO INPUT ANOTHER HALF-LIFE')
   11 FORMAT(7X'THE G-FACTOR FOR THE',1PD14.6,
     +/7X'DAY HALF-LIFE IS',1PD14.6,' PER SECOND'
     &/7X' AND THE C-FACTOR IS ',1PD14.6,' AND IS DIMENSIONLESS')
   13 FORMAT(7X'GIVE THE DAY, HOUR, AND MINUTE FOR THE CONTINUATION'
     +/7X'OF THE IRRADIATION'/)
   14 FORMAT(7X'THE TOTAL IRRADIATION TIME WAS',D16.8,' SECONDS'/)
   15 FORMAT(7X'THE IRRADIATION CYCLE LASTED',D16.8,' SECONDS'/)
   16 FORMAT(7X'THE TOTAL FLUENCE FOR A FLUX DENSITY OF 1 N/CM**2-SEC'
     &/7X'AT THE END OF THE IRRADIATION WAS ',1PD16.8,/)
   18 FORMAT(7X'IF THE CF SOURCE HAD A CONSTANT EMISSION RATE, THE'
     &/7X'TOTAL FLUENCE FOR A FLUX DENSITY OF 1N/CM**2-SEC AT THE START'
     &/7X'OF THE IRRADIATION WOULD HAVE BEEN ',1PD16.8,/)
   19 FORMAT (/7X' THE RESULTS ARE STORED ON FOR003.DAT.')
      OPEN(3,STATUS='NEW')
TOUT=0.D0
WRITE(7,3)
WRITE(7,4)
WRITE(3,3)
READ(7,1)II
WRITE(7,5)
*       READ DAY, HOUR, AND MINUTE FOR THE END OF THE IRRADIATION.
READ(7,2)DE,HE,TE
WRITE(7,6)
*       READ DAY, HOUR, AND MINUTE FOR THE START OF THE IRRADIAT1ION.
READ(7,2)DS,HS,TS
TS=TS+(DS*24.D0+HS)*60.D0
TE=TE+(DE*24.D0+HE)*60.D0-TS
IF(II.EQ.0)GO TO 110
*       READ THE INTERRUPTION AND RESTART TIMES
DO 100 I=1,II
IF(I.EQ.1)GO TO 101
WRITE(7,8)
```

```
READ(7,2)TDI,THI,TMI(I)
GO TO 102
  101 WRITE(7,7)
READ(7,2)TDI,THI,TMI(I)
  102 WRITE(7,13)
READ(7,2)CDI,CHI,CMI(I)
TMI(I)=TMI(I)+(TDI*24.D0+THI)*60.D0-TS
CMI(I)=CMI(I)+(CDI*24.D0+CHI)*60.D0-TS
  100 TOUT=CMI(I)-TMI(I)+TOUT
  110 T=TE-TOUT
      TSS=T*60.D0
      TES=TE*60.D0
WRITE(7,14)TSS
WRITE(7,15)TES
WRITE(3,14)TSS
WRITE(3,15)TES
ALCF=-0.69314718D0/(2.645D0*365.25D0*1440.D0)
      PFLU=0.D0
      BC=1.D0
      IF(II.EQ.0)GO TO 180
      DO 175 I=1,II
  175 PFLU=PFLU+DEXP(ALCF*CMI(I))-DEXP(ALCF*TMI(I))
      BC=(1.D0-PFLU/(DEXP(ALCF*TE)-1.D0))
  180 FLU=(BC/ALCF)*(1.D0-DEXP(-ALCF*TE))*60.D0
      WRITE(3,16)FLU
      WRITE(7,16)FLU
      WRITE(3,18)TSS
      WRITE(7,18)TSS
  200 WRITE(7,9)
READ(7,2)TAU
BLAM=0.69314718D0/(1440.D0*TAU)
ALAM=BLAM+ALCF
PSATFR=0.D0
B=1.D0
IF(II.EQ.0)GO TO 260
DO 250 I=1,II
  250 PSATFR=PSATFR+DEXP(ALAM*CMI(I))-DEXP(ALAM*TMI(I))
B=(1.D0-PSATFR/(DEXP(ALAM*TE)-1.D0))
  260 SATFR=B/ALAM*(1.D0-DEXP(-ALAM*TE))*BLAM
      BLAM=BLAM/60.
      GFAC=SATFR/FLU
      CFAC=GFAC/BLAM
WRITE(7,11)TAU,GFAC,CFAC
WRITE(3,11)TAU,GFAC,CFAC
WRITE(7,10)
READ(7,1)MM
IF(MM.EQ.1)GO TO 200
      WRITE(7,19)
      CLOSE(3)
STOP
END
```

APPENDIX C

TEST REPORT FOR NEUTRON FLUENCE STANDARDS PRODUCED BY AN IRRADIATION AT THE NBS ^{252}Cf FISSION NEUTRON SOURCE

Irradiation No.: Report Date:

Neutron Source:

Sensor Materials Irradiated:

(designations:

End of Irradiation (EOI):

REPORT FOR:

Free-field, fission-neutron fluence in compensated beam geometry (averaged over the volume of the detectors):

Φ = n/cm^2

Physical description of material: Mass =

Form: mm dia. X mm thk disk

Material:

Additional parameters of irradiation:

Decay correction factor: C = using λ =

Length of Irradiation: T =

Departure from free-field activity due to neutron scattering:

The gradient of the neutron fluence across the thickness of the detector package is %/mm. The fluence at the center of the detector disk is % greater than at the edge.

Inventory of Uncertainties (1σ)

(1) Neutron source strength of ^{252}Cf	1.1%
(2) Distance measurement for compensated geometry	0.2%
(3) Neutron scattering including sensors	0.5%

Other Radioactivities:

Since the sensors are natural metal samples, activities from neutron reactions in other isotopes and impurities may be present. Ordinarily, this will not be a problem for Ge(Li) and other high resolution gamma detectors. The following lines have been observed above the annihilation energy:

TEST REPORT FOR NEUTRON FLUENCE RATE STANDARDS PRODUCED BY AN IRRADIATION AT THE NBS ^{252}Cf FISSION NEUTRON SOURCE

Irradiation No.: Report Date:

Neutron Source:

Sensor Materials Irradiated:

(designations:

End of Irradiation (EOI):

REPORT FOR:

Free-field, fission-neutron fluence rate in compensated beam geometry averaged over the volume of the detectors):

ϕ = n/cm^{2}s

Physical description of material: Mass =

Form: mm dia. X mm thk disk

Material:

Additional parameters of irradiation:

Decay correction factor: [λTC] = using λ =

Length of irradiation: T =

Departure from free-field activity due to neutron scattering:

The gradient of the neutron fluence rate across the thickness of the detector is %/mm. The fluence at the center of the detector disk is % greater than at the edge.

Inventory of Uncertainties (1σ)

(1) Neutron source strength of ^{252}Cf	1.1%
(2) Distance measurement for compensated beam geometry	0.2%
(3) Neutron scattering including sensors	0.5%

Other Radioactivities:

Since the sensors are natural metal samples, activities from neutron reactions in other isotopes and impurities may be present. Ordinarily, this will not be a problem for Ge(Li) and other high resolution gamma detectors. The following lines have been observed above the annihilation energy:

Comments:

1) This service is offered as SP No. 44080C. A document describing the service is in preparation and will be forwarded when available.

2) Reference Docoument: "Compendium of Benchmark Neutron Fields for Reactor Dosimetry," NBS Document NBSIR 85-3151 (January 1986).

This fluence standard prepared by:

_____

Neutron Dosimetry Group

Reviewed by:

_____

J. A. Grundl
Neutron Dosimetry Group

For the Director

_____

R. S. Caswell, Chief
Ionizing Radiation Division

Distribution (File copy only)

laboratory	contact	test report sent	fluence standard sent

Relationship of Test Report Data to the Measured Activity at End of Irradiation (EOI) for ^{252}Cf Fission Neutron Irradiations

Compensated beam geometry. Neutron fluence standards are irradiated in pairs on opposite sides and equidistant from the source. In this arrangement, which is called compensated beam geometry, the fluence or fluence rate given in the test report is related to the geometric mean, $\bar{A}_m$, of the two measured detector activities:

$$\bar{A}_m = \sqrt{A_1 A_2} = [\text{const.}] \times [\text{fluence or fluence rate}]. \quad (1)$$

Average neutron fluence or fluence rate. This test report certifies a neutron fluence or fluence rate which is a free-field value (i.e. neutron scattering is neglected) averaged over the volume of the detectors. A correction for neutron scattering(μ_{sc}) may be included in the test report. Based on a neutron transport calculation, it is equal to the fractional departure from free-field activation attributable to neutron scattering:

$$\bar{A}_m = \bar{A}\,(1 + \mu_{sc}) \quad (1a)$$

where $\bar{A}_m$ = mean of detector activities measured at EOI (Eq. 1)

$\bar{A}$ = net free-field mean of detector activities at EOI

μ_{sc} = fractional departure from free-field activity due to neutron scattering. (A minus sign means that the observed activity is lower than it would be in free-field conditions because of a net outscatter of neutrons.)

Activation equation for neutron fluence. For simple decay, the activity of the neutron fluence standard in disintegrations per second is related to the free-field fission neutron fluence by the activation equation:

$$\frac{\bar{A}}{\lambda C N} = \sigma \cdot \Phi \quad (2)$$

where: $\bar{A}$ = net free-field mean of detector activities at EOI (dps) from Eq. 1a

λ = decay constant of reaction product

C = decay correction factor given in test report. For an uninterrupted irradiation of length T at a constant fluence rate, C is equal to $[(1-\exp(-\lambda T))/\lambda T]$. (When it is appropriate to report average neutron fluence rate, the decay correction factor is given as the product λTC -- see below.)

N = number of reaction isotope atoms in the fluence standard

σ = fission-spectrum-averaged reaction cross section (cm^2)

Φ = free-field fission neutron fluence (n/cm^2). (Note: the symbol Φ is recommended by ICRU and ASTM E170.)

$\sigma \cdot \Phi$ = reaction probability

Activation equation for average neutron fluence rate. For decay half-lives short compared to the length of irradiation, the activation equation is commonly expressed in terms of the average reaction rate (experimentally, the saturated specific activity):

$$\frac{\bar{A}}{[\lambda TC]N} = \sigma \cdot \langle\phi\rangle \qquad (3)$$

$\frac{\bar{A}}{[\lambda TC]N}$ = saturated specific activity

$[\lambda TC]$ = decay correction factor for average neutron fluence rate. For an uninterrupted irradiation of length T at constant fluence rate, $[\lambda TC]$ is equal to $[1-e^{-\lambda T}]$.

T = length of irradiation given in the test report.

$\langle\phi\rangle$ = the neutron fluence rate ($n/cm^2 s$)

$\sigma\langle\phi\rangle$ = average reaction rate

Activation equation for fission product activity. When the observed gamma disintegration is from direct formation of a fission product, Equation 2 becomes

$$\bar{A} = Y \cdot \lambda \cdot C \cdot N \cdot (\sigma \cdot \Phi), \qquad (4)$$

where Y is the chain fission yield. When the fission product activity is from the daughter of a radioactive parent with a half life longer than that of the daughter, the appropriate free-field activity at equilibrium is

$$\bar{A} = [\lambda'/(\lambda'-\lambda)] \cdot Y \cdot \lambda \cdot C \cdot N \cdot (\sigma \cdot \Phi), \qquad (5)$$

where λ is the decay constant of the parent activity and λ' that of the daughter.

Recommended Procedure for Establishing a Calibration Factor (χ_{cf})

Use Equations (2) or (3) to derive a free-field neutron fluence or average fluence rate, respectively, from the measurement of a particular activity of the neutron sensor. Define a calibration factor for this activity as the ratio of the NBS certified fission neutron fluence or fluence rate to the derived neutron fluence or fluence rate. Then, to obtain an NBS calibrated result for subsequent measurements with this activity, either multiply observed reaction probability by the calibration factor or divide the detector cross section by the calibration factor. This calibration procedure will be most effective if the derived fluence or fluence rate for the NBS Standard is obtained with the same reaction parameters and detector cross section set as is used to obtain spectrum-averaged detector cross-sections for subsequent neutron field measurements.

Revised: November 3, 1986