# NBS MEASUREMENT SERVICES: ACTIVATION FOIL IRRADIATION BY REACTOR CAVITY FISSION SOURCES

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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 250-14), NBS Measurements Services: Activation Foil Irradiation by Reactor Cavity Fission Sources, by G. P. Lamaze and J. A. Grundl, is the fourteenth 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 44090C 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.

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# ABSTRACT

This document describes the National Bureau of Standards (NBS) calibration service 44090C (formerly 8.1R) 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 measured neutron fluence in a fission-spectrum 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 $3 \times 10^{15}$ neutrons/cm$^2$, with combined ($1\sigma$) uncertainties of ± 2%.

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# 1. DESCRIPTION OF SERVICE

The National Bureau of Standards (NBS) maintains standard fission neutron fields (or benchmark fields) to supply calibration irradiations and certified fluence standards. The NBS Cavity Fission Neutron Source provides a $^{235}U$ thermal-neutron-induced fission neutron field spectrum. Known fluences are based upon the known source strength for the National Standard Ra-Be Neutron Source, NBS-I [1]. The NBS calibration service 44090C [2] (formerly 8.1R) 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 measured neutron fluence in a fission-spectrum 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 and associated uncertainties. Typical maximum fluences are of the order of $3 \times 10^{15}$ neutrons/$cm^2$, with combined ($1\sigma$) uncertainties of $\pm$ 2%. Dosimetry sensors must fit within a cylindrical cadmium enclosure of 20-mm-inside diameter and 6.5-mm-inside height.

For a typical activation detector, the customer measures the activity of the irradiated sensor with his own detector 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 (Bq) at EOI,

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

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

N = number of atoms in the target isotope,

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

and     $\Phi$ = 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

$$\frac{A}{\lambda C T N} = \bar{\sigma} \cdot \phi \qquad (2)$$

1

where: $(A/\lambda CTN)$ = reaction rate probability (disintegrations per
second/nucleus.),

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

T = the length of the irradiation (s) given in the test report.

## 1.1 CONFIGURATION AND CHARACTERISTICS [3], [4], [5]

The NBS cavity fission source operates at the center of a 30 cm diameter
spherical cavity in the NBS Research Reactor graphite thermal column. The
source-detector capsule consists of two coaxial source disks of $^{235}U$ metal
(16 mm dia x 0.13 mm thick) placed outside a cadmium box that encloses passive
detectors for exposure. The cavity and source-detector arrangement are shown
in Fig. 1. The source-detector assembly is designed so that the cadmium box
and detectors may be removed separately from the fission disks. Fission
neutron fluence rates of $\sim 4 \times 10^{10} \ n/cm^2$ s are obtained at the midplane
between the source disks when they are at a separation distance of 11 mm. For
thin disk-shaped detectors with diameters up to 13 mm, axial fluence gradients
are mild:  the fluence at 1 mm from the midplane is 4% greater than at the
midplane. Center-to-edge fluence ratios are more severe — see Section 2.1.

Fission neutron return for spherical cavities in graphite has been
studied extensively by means of neutron transport calculations and by experi-
ment in connection with the general development of standard neutron fields in
a spherical geometry [6]-[10]. For the 30-cm cavity at NBS, the response of
threshold fission detectors to fission neutrons returning from the cavity
walls is less than 0.3% of the uncollided fission neutron fluence from the
fission source disks;  for detectors with thresholds above 1 MeV, the response
to the cavity return fluence is less than 0.1%. These estimates, based on
calculation, have been checked experimentally by means of fission chamber
traverses in the cavity fission sources at Center for Nuclear Studies in Mol,
Belgium (SCK/CEN) [9]. More important is fission neutron scattering in the
source-detector capsule. The scattered neutrons have a different energy
distribution then the original spectrum and therefore will generally have a
different interaction probability. Monte Carlo calculations are employed for
this problem and they provide corrections for threshold detectors which vary
with detection threshold and with the number and arrangement of detector disks
[3]. Corrections for activation rates are generally between 0.5% and 3% (see
section 2.1).

The NBS fission source is not generally used for low-energy integral
detectors because of uncertainties in the graphite return background which has
a near-1/E distribution of neutron fluence. The graphite return correction
for the $^{235}U(n,f)$ reaction, for example, is nearly 10% and self-shielding of
the cavity return fluence by the fission disks and by the $^{235}U$ activation foil
makes this correction uncertain to a factor of about one third. The one meter
cavity arrangement at the SCK/CEN is better suited for low-energy response
detectors [9].

NBS CAVITY FISSION SOURCE

$^{235}U$ SOURCES

DETECTOR STACK

CADMIUM ENCLOSURE

SPHERICAL CAVITY (30 cm dia.)

GRAPHITE ACCESS PLUG

CAVITY FISSION SOURCE
AND CADMIUM-COVERED
DETECTOR CAPSULE

GRAPHITE THERMAL COLUMN
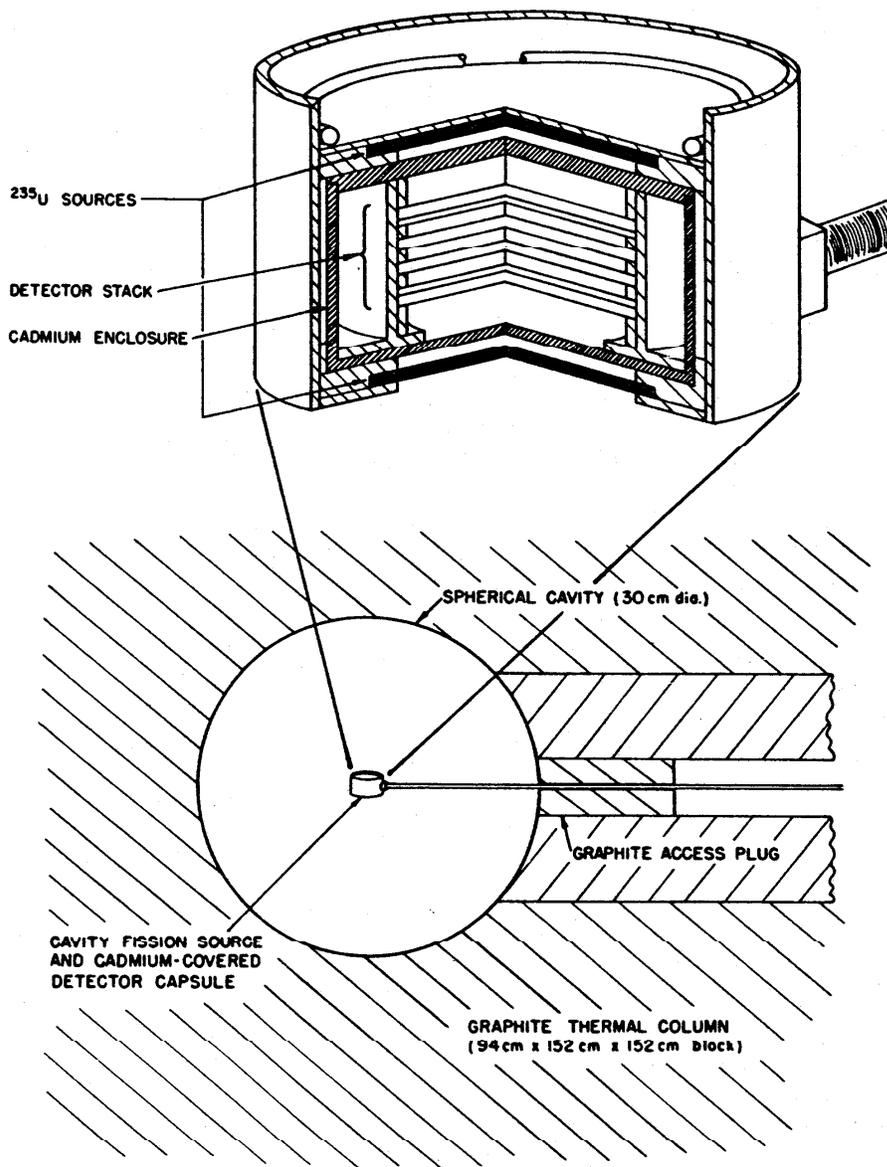(94 cm x 152 cm x 152 cm block)

Fig. 1. $^{235}U$ Cavity fission source. Upper view: relative positions of neutron sensor foils, fission disks and cadmium enclosure. Lower view: the assembly in the thermal column of the NBS reactor.

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## 2. NEUTRON FIELD CHARACTERIZATION

The energy spectra of fission neutrons are very similar for all fissionable nuclides and both test and power reactors exhibit fission-spectrum-like components in the MeV energy range. The $^{235}U$ fission neutron spectrum may be described in terms of a broad energy range,

| | lower bound $E_p(p=0.99)$ | median $E_p(p=0.5)$ | upper bound $E_p(p=0.01)$ |
|---|---|---|---|
| $^{235}U$ | 0.08 MeV | 1.57 MeV | 7.2 MeV |

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

| E limits: | 0 | 0.25 | 0.8 | 1.5 | 2.3 | 3.7 | 8 | 12 MeV |
|---|---|---|---|---|---|---|---|---|
| $\phi(E)dE$: | | 0.054 | 0.197 | 0.229 | 0.195 | 0.192 | 0.127 | 0.006 |

A detailed description of the spectrum is given in Section 2.2.

Although the neutron spectrum listed is specifically for thermal-neutron-induced $^{235}U$ fission, the fission spectrum for fast-neutron-induced fission is little different. The change in median energy is about 0.05 MeV.

### 2.1 NEUTRON FLUENCE (See also Sections 1.1. and 3.)

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 [11-13] and the source strength (n/s into $4\pi$ steradians) is now established as $1.243 \times 10^6 \pm 0.85\%$ ($1\sigma$). The NBS $^{252}Cf$ spontaneous fission neutron source strengths are obtained by measuring their ratio to NBS-I in a manganese sulfate bath [14]. The uncertainty of the source strength obtained in this fashion is $\pm 1.1\%$ ($1\sigma$). The $^{235}U$ fission neutron fluence in the cavity is established by means of fluence transfer from the $^{252}Cf$ fission neutron irradiation facility. The neutron fluence at the midplane of the source-detector capsule for a source disk separation of 12 mm is about 0.12 $n/cm^2$ for each neutron emitted from the fission disks which corresponds to a fluence rate of $\sim 3.5 \times 10^{10}$ $n/cm^2$ s with the reactor at 20 megawatts full power. The fission neutron fluence rate is minimum at the midplane between the two source disks. At 1 mm above or below the midplane, the mean fluence for a 12 mm diameter disk detector is $\sim 4\%$ higher than at the midplane. Radial variations are more severe: center-to-edge fluence ratios are 1.2 for 12 mm diameter disks. This gradient does not require that detector disks be extremely uniform. A radial mass nonuniformity of 20%, for example, distributed linearly in a detector foil changes the activation by less than 0.5% compared to a uniform mass distribution. Also, the nickel fluence monitor included in each irradiation package yields a fluence averaged over the radial variation.

The fission neutron fluence is perturbed by a small cavity return contribution, and also by scattering in fission source disks, in encapsulations, and in the detector foils. The correction for these effects, based on calculation, is given as the fractional departure for each detector ($\mu_{sc}$) from its free-field activity:

$$\begin{array}{l}\text{total} \\ \text{measured} \\ \text{activity}\end{array} = \begin{array}{l}\text{free-field} \\ \text{activity}\end{array} [1 + \mu_{sc}]$$

This correction depends markedly upon reaction threshold because of the change in neutron energy caused by scattering in the nearby materials of the encapsulation. Components of $\mu_{sc}$, obtained from detailed Monte-Carlo calculations [15] of the source-detector capsule with a representative foil stack, are as follows:

|  | change in neutron fluence | changes in threshold detector response for the indicated changes in neutron fluences | | |
|---|---|---|---|---|
|  |  | $E_{95}$ = 1.2 MeV | $E_{95}$ = 2.1 MeV | $E_{95}$ = 6.5 MeV |
| cavity return | +1.2% | +(0.17± 0.1)% | < 0.1% | < 0.1% |
| encapsulations (incl. source disks and cd box) | +2.5% | +(0.5 ± 0.3)% | -(0.2 ± 0.3)% | -(1.3 ± 0.3)% |
| foil stack | +1.5% | +(0.0 ± 0.4)% | -(0.6 ± 0.4)% | -(0.8 ± 0.4)% |
| $\mu_{sc}$(%): |  | +(0.7 ± 0.5)% | -(0.8 ± 0.5)% | -(2.1 ± 0.5)% |

The uncertainties listed are 1σ. The spread of the foil-stack component of $\mu_{sc}$ is associated with foil position. It is possible to derive from the Monte Carlo calculation individual foil-to-foil scattering factors as a function of foil separation. These correction functions can be applied to a range of foil stack arrangements beyond the one calculated as long as the foil-to-foil scattering is mainly first collision. An overall uncertainty of ± 0.5% (1σ) is assigned to $\mu_{sc}$. This uncertainty has been estimated from the distribution of the scattering corrections from a large number of foil positions in several different Monte Carlo calculations.

It is the small source-to-detector distance which makes it possible to obtain nearly pure fission neutron fluences in small cavities. The cavity return fluence at the cavity center is about 1.2% of the uncollided fission neutron fluence. This return fluence gives rise to threshold detector responses that are between 0.05% and 0.3% of the free-field fission neutron response depending upon the reaction threshold.

## 2.2. NEUTRON SPECTRUM

2.2.1. Calculation. Fission neutron spectra are most accurately determined by laboratory measurement because all the calculations have undetermined fitting parameters. (See, however, Ref. [16] and earlier work cited therein.)

2.2.2. Measurement [17], [18]. From 1952 to the present, well-documented measurements of $^{252}$Cf and $^{235}$U fission neutron spectra have employed every perfected technique of neutron spectrometry [19]. Accordingly, fission spectra are the most widely studied fast-neutron energy distributions in existence. The large body of documented fission spectrum data was evaluated at NBS in 1975 [18]. The evaluation included an estimate of the spectrum uncertainties in a 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 3 of ref. [20].

2.2.3. Evaluated Spectrum. The NBS fission spectrum evaluation has been chosen for presentation in this document because of the consistency of its procedures for both $^{235}$U and $^{252}$Cf fission spectra, and because it includes an articulated uncertainty estimate [17], [18]. Many other evaluations are available.

The NBS evaluated spectrum is described up to 20 MeV by means of a reference Maxwellian, M(E), modified by four piecewise continuous linear segments below 6 MeV and one exponential segment above 6 MeV. The reference Maxwellian for the $^{235}$U fission spectrum is,

$$M(E) = 0.750 \sqrt{E} \exp(-1.5E/1.97), \text{ E in MeV,}$$

and the adjustment functions $\mu(E)$ are as follows (here 25 denotes $^{235}$U):

| Energy Interval (MeV) | $\mu_{25}(E)$ |
|---|---|
| 0 - 0.25 | 1 + 0.8E - 0.153 |
| 0.25 - 0.8 | 1 - 0.14E + 0.082 |
| 0.8 - 1.5 | 1 + 0.040E - 0.062 |
| 1.5 - 6.0 | 1 + 0.01E - 0.017 |
| 6.0 - 20 | 1.043 exp [-0.06(E - 6.0)/1.043] |

The evaluated spectrum is given by $\chi(E) = \mu(E) \cdot M(E)$. A 45-group tabulation of the evaluated $^{235}$U fission spectrum is given in table 1 and plotted in figure 2. The evaluated $^{252}$Cf fission neutron spectrum is given for comparison purposes. Group fluences for other energy group structures may be derived from this tabulation with the interpolation function recommended in the footnote to the table. Spectrum uncertainties, given in table 2, are based on standard deviations from the mean of subsets of measured data from the final adjusted Maxwellian of the evaluation. This estimate, carried out in a seven-group structure, is based on the scatter of the data from the sixteen experiments which were evaluated.

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TABLE 1. EVALUATED FISSION NEUTRON SPECTRA FOR $^{252}$Cf AND $^{235}$U
IN 45-GROUP FORMAT

| Lower Energy Boundary (MeV) | $^{252}$Cf | | $^{235}$U | |
|---|---|---|---|---|
| | 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.

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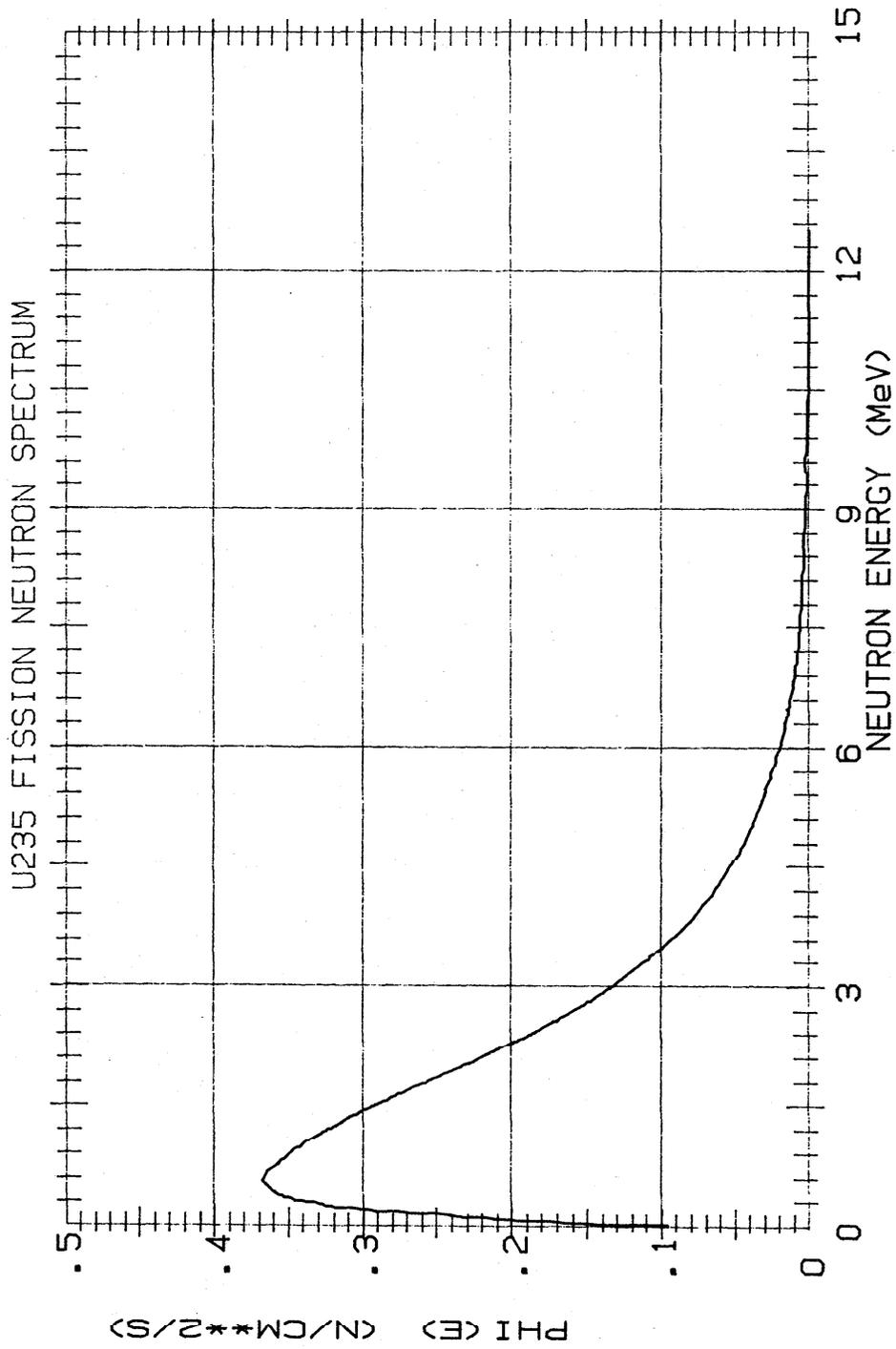
Fig. 2. The NBS evaluation of the $^{235}$U fission neutron spectrum.

TABLE 2.  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 $1\sigma$ (%) | Group Fluence $\phi \cdot \Delta E$ | Uncertainty $1\sigma$ (%) |
| 0.0 | | | | |
| 0.25 | 0.047 | ±13 | 0.054 | ±16 |
| 0.8 | 0.184 | ± 1.1 | 0.197 | ± 4.1 |
| 1.5 | 0.220 | ± 1.8 | 0.229 | ± 3.0 |
| 2.3 | 0.194 | ± 1.0 | 0.195 | ± 3.1 |
| 3.7 | 0.200 | ± 2.0 | 0.192 | ± 2.0 |
| 8 | 0.146 | ± 2.1 | 0.127 | ± 4.8 |
| 12 | 0.0087 | ± 8.5 | 0.0056 | ± 5.3 |
| 20 | (0.00058)* | | (0.00026)* | |

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

Of the many alternative descriptions employed for the $^{235}$U fission spectrum, the most common is the Watt-function fit chosen for ENDF/B-V [21] in June, 1979:

$$\psi(E) = C \exp(-E/a) \sinh(\sqrt{bE})$$

$$a = 0.988 \text{ MeV} \qquad b = 2.249 \text{ MeV}^{-1} \qquad C = 0.4395$$

$$\text{median energy:} \quad E_p(p=0.5) = 1.64 \text{ MeV} .$$

Contemporary time-of-flight data were emphasized in the ENDF/B-V evaluation as compared with the NBS evaluation which considered all well-documented data.

## 3.  IRRADIATION PROCEDURES:

Disk detector irradiations are monitored with nickel or indium monitor foils placed between the detectors.  The neutron fluence in the detector capsule is obtained from the measured $^{58}$Ni(n,p) or $^{115}$In(n,n') activity on the basis of a fluence transfer calibration carried out at the NBS $^{252}$Cf Irradiation Facility.  In this procedure, the ratio of the activity obtained in a known $^{252}$Cf fission neutron fluence to that obtained during a cavity fission source irradiation is translated into a $^{235}$U fission neutron fluence at each monitor foil position in the source-detector capsule.

The expression which governs this fluence transfer is

$$\Phi_{cvy} = \frac{R_{cvy}}{R_{cf}} \cdot \frac{\overline{\sigma}_{cf}}{\overline{\sigma}_{cvy}} \Phi_{cf}$$

$$= \frac{G_{cf}}{G_{cvy}} \cdot \frac{\sigma_{cf}(>E_p)}{\sigma_{cvy}(>E_p)} \cdot \frac{\psi_{cf}(>E_p)}{\psi_{cvy}(>E_p)} \cdot \frac{D_{cvy}}{D_{cf}} \cdot \Phi_{cf}$$

where $\Phi$ = fission neutron fluence and "cvy" denotes the [235]U cavity fission source and "cf" the [252]Cf fission neutron calibration fluence,

$R$ = the reaction probability, and

$E_p$ = truncation energy (p = 0.95 commonly).

The ratios on the right side of the equation involve decay correction factors (G), truncated reaction cross sections $\sigma(>E_p)$, spectrum fractions $\psi(>E_p)$, and the ratio of the monitor foil activity, D, in the cavity to that at the [252]Cf irradiation facility. Details of the fluence transfer procedure are given in Refs. [4], [22]. These references also describe a set of validation experiments, carried out at the SCK/CEN Facility, in which [239]Pu(n,f), with negligible fission spectrum shape sensitivity, was effectively substituted for [58]Ni(n,p) and [115]In(n,n') as the fluence transfer reaction. The fluence transfer procedure yields a free-field [235]U cavity fission neutron fluence to an accuracy of ± 2.0% ($1\sigma$).

Fission neutron fluences of ~ 3 x $10^{15}$ n/cm$^2$ may be obtained at the NBS facility in 24 hour long irradiations. The small irradiation volume restricts multiple foil packages to a diameter of 12 mm and a thickness of a few millimeters. The gamma-ray dose from neutron capture in graphite (~ 0.7 kGy/h) and cadmium (~ 4.4 kGy/h) is about 46% of the neutron and gamma-ray dose from the fission sources.

## 4. DETAILED OPERATING PROCEDURES

### 4.1. ASSEMBLY AND IRRADIATION OF SENSORS

4.1.1. 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 the plastic bag containing the sensor. The masses of the nickel and/or indium foils are necessary to calculate the fluence; other masses are measured to obtain a complete record of the radiation.

4.1.2. The sensors should be mounted into the detector capsule. The positions of the sensors should be recorded on the worksheet (see appendix A) and in the notebook.

4.1.3. Arrange time at the thermal column with other users, principally the radiography group and the activation analysis group.

4.1.4. Source disk assembly. The detector capsule is now ready for assembly with the source disks. Two pairs of source disks are presently in use (CV-3 through CV-6). Each source disk (15.9-mm dia. × 0.15-mm thick) is made up of two 0.08-mm thick enriched uranium metal foils wrapped in 0.04-mm thick aluminum foil. The total weight of a wrapped source disk is about 0.60 g and the total thickness is 0.38 mm. During irradiation the fission neutron source strength of the two disks is about $4 \times 10^{11}$ n/s.

When not in use the source disks are kept at the far end of the lead shield cart inside of the lead cup (LC). (To get LC in and out remove lead bricks and short cover plate at front of cart.) Always check the radiation level of the source disks before handling. Use the lead cup as a carrier when moving source disks to the assembly area. Wear glasses for beta shielding.

4.1.5. Set up fission chamber monitor and electronics. Run "timer" program on HP85 (see Appendix C). Check time marking instrument with telephone time (844-2525) or Naval Observatory Time (653-0258). Notify reactor control room. Raise the boral curtain while noting the time. Record the time in notebook, irradiation log and worksheet (see appendix B), and thermal column log sheet.

4.1.6. Leave sign with your name and phone numbers on boral curtain controls.

4.1.7. At the end of the irradiation (EOI), again check time. Notify reactor control room. Lower the boral curtain while noting the time. Record the EOI time in notebook, worksheet, and thermal column log.

4.1.8. Remove sensors from cadmium detector capsule using gloves and package individually in marked plastic containers. Store in shielded card file in building 235, room B123. Aluminum spacers and detector capsule should be stored in pig either in room B123 or in thermal column area. Cadmium box should be returned to its plexiglass holder and also stored as appropriate to its residual radiation level.

## 4.2. FLUENCE AND FLUENCE RATE CALCULATIONS

4.2.1. Monitor Counting. The gamma radiation from the nickel (or indium) foils is counted in the standard geometry with the intrinsic germanium detector. The standard geometry is foil holder #3 mounted directly on the detector with the foil located in the recessed hole. A pulser should be fed into the preamplifier and the rate (usually 50 pps) adjusted to obtain sufficient counts in the pulser peak for the chosen counting interval. Detailed instructions for the operation of the counting system are in appendix D. The stored spectra are analyzed with the code STP2D8, (a FORTRAN code). A listing of this code is in appendix E. The code includes a time history correction and gives a final result as a counting rate at EOI. All results should be corrected for deadtime and pileup by finding the live-time correction which

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would restore the pulser results to the known, true pulser rate. The determined counting rate must be corrected for losses of events due to attenuation of the outgoing gamma rays. The correction factor, $C_\gamma$, is equal to $(1-e^{-\mu t})/\mu t$ where $\mu$ is the mass attenuation coefficient, for example, for the 810 keV line in nickel, $\mu$ is 0.589 cm$^{-1}$. The resulting correction for a .0254 cm nickel foil is 0.9925. This corrected result is then divided by the mass of the foil and entered as D in the worksheet.

4.2.2. <u>Scattering corrections and gradient corrections</u>. The FORTRAN code "SCATFO" (see appendix F) is run on a Pro350 computer. Input data are taken from the foil assembly worksheet and the program gives $\mu_{sc}$, the activation by scattered neutrons. The code also gives the field gradient, $\phi(z)/\phi(0)$. These parameters should be entered on the fluence calculation worksheet. A value of D at the midplane of the foil holder can now be calculated for each foil and an average value obtained for the set of foils.

4.2.3. <u>Neutron Fluence Calculation</u>. The code "SAT260" (see appendix F) is run to obtain corrections for loss of activity during irradiation. If the power level was approximately constant, a good correction can be obtained from $C_t = (1-e^{-\lambda t})/\lambda t$ where $\lambda$ is the decay constant and t is the length of the irradiation. The activation decay rate factor, G, is equal to $C_t \lambda$. The neutron midplane fluence is now calculated using equation 4 of the worksheet. Table 6 of the worksheet can now be completed which gives the fluence and the scattering correction for each sensor. Test reports should also be prepared at this point for each sensor. A sample test report is in appendix H.

## 4.3. <u>SUMMARY OF UNCERTAINTIES</u>

The uncertainties of the certified fluence are given in Table 3. The first part of the table gives uncertainties from the fluence transfer procedure from the $^{252}$Cf field to the $^{235}$U field. The second part of the table gives uncertainties for preparation of the fluence standards at the cavity fission source. The resultant total uncertainty is for the free field fluence. Additional uncertainties due to scattering corrections in the fluence standard must also be added. This last term varies depending on the activation reaction and the placement of the sensors in the detector capsule.

The table indicates which uncertainties are mostly systematic and which are mostly random. In some cases, the uncertainty terms consist of both random and systematic components. A more important distinction in the uncertainty terms is whether the uncertainty could lead to an overall change in scale of the NBS neutron fluence values. Two components could have this effect: 1) the measured value of NBS I could change (thereby changing all source strength values, and 2) improved corrections for neutron scattering for Pu(n,f) in the Cf and $^{235}$U fission fields.

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TABLE 3.   ASSIGNMENT OF UNCERTAINTIES (± 1 σ)

- Fluence Transfer Measurements from the $^{252}$Cf Standard Fission Neutron Field to the $^{235}$U Cavity Fission Source:

|  | Percent | |
|---|---|---|
| (1) Absolute neutron source strength of the $^{252}$Cf source | 0.9 | S+R* |
| (2) Distance measurement | 0.6 | R |
| (3) $^{252}$Cf to $^{235}$U spectrum-averaged cross section ratio for Pu(n,f) monitor | 0.1 | S |
| (4) Correction for neutron scattering for Pu(n,f) | 0.9 | S |
| (5) Reproducibility of fluence transfer procedure (including counting statistics, and electronic system stability over period of measurement) | 0.4 | R |

Subtotal (summed in quadrature) = 1.5

- Preparation of Neutron Fluence Standards at the $^{235}$U Cavity Fission Source:

|  | Percent | |
|---|---|---|
| (1) Gradient corrections and positioning | 0.4 | S+R |
| (2) Corrections for neutron scattering for $^{115}$In(n,n') | 1.2 | R |
| (3) Reproducibility of fluence monitoring including random summing corrections and uncertainties in activation decay corrections | 0.4 | R |

Subtotal (summed in quadrature) = 1.3

TOTAL UNCERTAINTY = 2.0

*S is for systematic uncertainty and R is for random uncertainty.

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# 5. 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 detector to a known neutron fluence, $\chi_{25}$, in a $^{235}U$ fission spectrum establishes a calibration factor,

$$G = \frac{[\text{observed detector response}]}{[\chi_{25} \text{ neutron fluence}]} \qquad (1)$$

which may be used in conjunction with the calculated fission-spectrum-averaged cross section, $\sigma_\chi$, for the detector reaction to obtain a consistent benchmark calibration. The cross section, $\sigma_\chi$, 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 $\sigma_0$ and $\sigma_a$ (commonly from the ENDF/B-V dosimetry file).

[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-ray counting system and known reaction parameters, branching ratios, isotopic abundances, fission yields etc.), then the "observed detector response" in Eq. (1) becomes the reaction probability, and G is an observed cross section for the fission spectrum.]

The calibration factor may then be applied to the investigation of a study spectrum.

By irradiating samples nearly identical to those irradiated in the $^{235}U$ field, the fluence of the study field, $\Phi_{study}$, can be obtained as a simple ratio of detector responses, D, and spectrum averaged cross sections, $\bar{\sigma}$, so that:

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

In practice, $\frac{D_{study}}{D_X}$ is simply the time history corrected counting rate ratio of the sample irradiated in the unknown field to that irradiated in the $^{235}U$ 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 $\Phi_X$ is that given in the test report and one then solves equation (2) for $\Phi_{study}$.

## 6. SAFETY

NRC and NBS regulations require that all personnel working at the NBS reactor undergo a radiation safety and awareness training course. This course acquaints the worker with the applicable radiation regulations as well as the hazards of radiation. Further training of any workers using the thermal column facilities is also mandatory. Very high radiation doses can be obtained through carelessness in the vicinity of the thermal column. A specific hazard for the cavity fission source irradiations is the fission-product gamma activity from the $^{235}U$ disks after they have been irradiated. Radiation doses can be reduced by: (1) waiting as long as possible before handling the disks, (2) having a well thought out plan of action for handling the disks, (3) working rapidly, and (4) using shielding whenever possible. Since the latter two often conflict, personal judgment based on experience must be used. Item 2 can be facilitated by performing "dry runs" with "cold" or dummy sources.

The irradiated samples will be radioactive, but at a very low level (usually a few microcuries). There is therefore little hazard from the samples, but all radioactive shipping regulations must still be followed. The samples should be stored in a lead shielded storage area while awaiting analysis or shipping.

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## DETECTOR ASSEMBLY WORKSHEET FOR CAVITY FISSION SOURCE

**Assembly Date:** 1/17/86   **Irradiation** 8601          **EOI:** 21:37:00  Jan 20, 1986

SOI:11:01:00 EST, Jan 18, 1986

| (1) Assembly Piece | | Depth Measurement (239-S) | | S (mils) | (2) Z |
| Identification | thk. | piece sum | micro-meter | assigned | (D= 443 ) = 324H19 |
|---|---|---|---|---|---|
| | | 239 | | | Z = S −120 |
| Spacer | 70 | | | | |
| Au-1 | 5 | 167 | | 72(Au-1) | −48 |
| Spacer | 10 | | | | |
| Ni-3 | 11 | 148 143 | 142.5 | 91(Ni-3) | −29 |
| Spacer | 11 | | | | |
| Ni-4 | 11 | 126 121 | | 113(Ni-4) | −7 |
| Spacer | 11 | | | | |
| Ni-5 | 11 | 105 99 | 101 | 134(ni-5) | +15 |
| Spacer | 11 | | | | |
| Ni-6 | 11 | 83 77 | 80 | 159(Ni-6) | +40 |
| Spacer | 15 | | | | |
| Sc: S-10 | 6 | 59 | | 180(Sc) | +61 |
| Au-2 | 5 | 53 51 | 55 | 186(Au) | +67 |
| Spacer | 60 | | | | |
| Σ=249 | | | | | |

**Assembled Cd. Capsule:**   324 = f

**Cd. Capsule Holder:**   367

(1) Pieces are assembled with i.d. number up

(2) Error does not include uncertainty in midplane position. Position error for Indium monitor is relative to adjacent foil.

f = dimension of assembled cadmium capsule (0.326" empty)

D = source disk separation: (f + 119)

Z = S + (102 - D/2)

   = detector position measured from midplane

| Foil | Mass (g) |
|------|----------|
| Ni-3 | 0.28609 |
| Ni-4 | 0.28775 |
| Ni-5 | 0.28564 |
| Ni-6 | 0.28721 |
| Au-1 | 0.28298 |
| Au-2 | 0.29142 |
| Sc S-10 | 0.04824 |

## IRRADIATION LOG AND WORKSHEET FOR CALCULATING NEUTRON FLUENCE

Irradiation Identification:        8601

Materials Irradiated:    Au, Ni, Sc

1. Source Assembly              i.d.      R/hr @ 10cm

   Top Source Disk:

   Bottom Source Disk:

   Source Separation Distance: 443

2. Irradiation Log:

   SOI: 11:01:00 EST   Jan 18, 1986   (DAY 18 )

   EOI: 21:37:00 EST   Jan 20, 1986   (DAY 20)

   Duration  2 days. 10 hr, 36 min   = 210960S

   Fission chamber monitor: Not used

3. Nickel and Indium Fluence Monitor Response at Midplane

| i.d. | z | $^{(a)}\Phi(z)/\Phi(0)$ | D(c/s,g) @ EOI | $^{(b)}$Activation by Scattered Flux | D(c/s,g) @ EOI (corrected) |
|------|-----|-----------|----------|----------|----------|
| Ni-3 | -29 | 1.0178 | 5491.1 | -0.007 | 5433 |
| Ni-4 | -7 | 1.0010 | 5395.1 | -0.007 | 5428 |
| Ni-5 | +15 | 1.0048 | 5368.0 | -0.007 | 5380 |
| Ni-6 | +40 | 1.0339 | 5436.0 | -0.007 | 5295 |

$$5385 \pm 33 (0.6\%)$$

4. Activation decay rate factors:

| | T= 2.10960S | $^{58}$Co | $^{198}$Au | $^{46}$Sc |
|---|---|---|---|---|
| $\lambda-$ | | $1.1334 \times 10^{-7}$ | $2.9762 \times 10^{-6}$ | $9.5700 \times 10^{-8}$ |
| satfac | | $2.363 \times 10^{-2}$ | $4.663 \times 10^{-1}$ | $1.999 \times 10^{-2}$ |
| G= satfac/T | | $1.1200 \times 10^{-7}$ | $2.210 \times 10^{-6}$ | $9.474 \times 10^{-8}$ |
| C= G/$\lambda$ = | | .9881 | .7426 | .9900 |

Note: For a constant fluence rate irradiation, satfac = $(1-e^{-\lambda T})$

5. <u>Fission Neutron Fluence at Midplane (z = 0)</u>: $\Phi_O$

$$\Phi_O = K_{cf} \cdot \frac{\sigma(x_{cf})}{\sigma(x_{25})} \cdot \lfloor D/G \rfloor$$

counter response of fluence monitors corrected for saturation (SATFAC code)

calculated cross section ratio for $X_{cf}$ to $x_{25}$ fluence transfer reaction (before 9-83: In(n,n); after 9-83:this factor is included in $K_{cf}$)

calibration factor (i.e. composite of detector efficiency parameters)(For GEM detector, single shelf, zero thickness nickel foil, $K_{cf}$ = 1.711 x 10$^5$)

$$\Phi_O = 1.711 \times 10^5 \cdot \frac{5385}{1.12 \times 10^{-7}} = 8.226 \times 10^{15}$$

6. <u>Fission Neutron Fluence at Neutron-Fluence-Standards Position</u>

| i.d. | Z mils | $\Phi(z)/\Phi(0)$ | free-field fluence @ foil pos. | $\mu_{sc}$ activity by scattered flux(%) | Total Error |
|---|---|---|---|---|---|
| Ni-3 | -29 | 1.0178 | $8.373 \times 10^{15}$ | -.7 | |
| Ni-4 | -7 | 1.0010 | $8.235 \times 10^{15}$ | -.7 | |
| Ni-5 | +15 | 1.0048 | $8.266 \times 10^{15}$ | -.7 | |
| Ni-6 | +40 | 1.0339 | $8.506 \times 10^{15}$ | -.7 | |
| Au-1 | -48 | 1.0488 | $8.628 \times 10^{15}$ | +.4 | |
| Au-2 | +67 | 1.0952 | $9.010 \times 10^{15}$ | +.5 | |
| S-10 | +61 | 1.0789 | $8.876 \times 10^{15}$ | +.6 | |

# NOTES FOR FLUENCE CALCULATION

## (a) Disk geometry gradient (empirical fit of calculation and measurement)

$\phi(z)$ = neutron fluence rate per source neutron at z = distance from midplane between coaxial disks a distance D apart.

$$= 0.0470 \, D^{-1.16} \, (1 + 21.2 \, z^2)$$

$$\frac{d\phi}{\phi} = 2 \, [1 + 1/(21.2 \, z^2)]^{-1} \quad \frac{dz}{z}$$

$$D = \underline{\hspace{1.5cm}} \qquad 0.0470 \, D^{-1.16} = \underline{\hspace{1.5cm}}$$

## (b) Corrections for scattered neutrons

Monte Carlo calculations from LANL provide net detector activation associated with collided neutrons:

THESE ARE THE INSTRUCTIONS FOR RUNNING THE ADCAM DIRECTLY FROM PRO/TOOL KIT

1.  LOAD ADCAM MENU BY TYPING @MENU OR RUN DIRECTLY BY TYPING @START AND SKIP TO STEP 3.1

2.  THE "A" MENU WILL BE DISPLAYED; A CARRIAGE RETURN WILL CHANGE TO THE "B" MENU; ANOTHER ONE CHANGES YOU BACK

3.  FROM THE "B" MENU, TYPE "3" OR "START"

 3.1    ANSWER QUERIES ABOUT MCA#

4.  WHEN IT ASKS FOR PARAMETERS, TYPE LT

5.  THEN TYPE IN THE PRESET TIME IN SECONDS

6.  WHEN IT ASKS IF YOU WANT TO TRANSFER PARAMETERS FROM ANOTHER FILE, SELECT NO (DEFAULT IS NO)

7.  WHEN IT ASKS IF YOU WANT TO ALTER SAMPLE DESCRIPTION, TYPE Y

8.  ENTER TWO LINE DESCRIPTION   (DETECTOR DESCRIPTION IS ALREADY ENTERED IN ANOTHER PLACE)

9.  MAKE SURE SAMPLE IS IN PLACE!

10. WHEN IT ASKS IF YOU WANT TO ALTER ANALYSIS PARAMETERS, TYPE N OR CR (DEFAULT)

11. DATA TAKING HAS NOW BEGUN, TIME IS AUTOMATICALLY RECORDED AND DATA TAKING WILL END AT PRESET ·

12. YOU MAY DO OTHER THINGS WITH COMPUTER AT THIS TIME, e.g., UPDATE GEMLOG.DOC; TYPE CONTROL Z TO RETURN TO TOOLKIT FROM ADCAM MENU MODE

13. TO SEE DATA, SELECT EMULATE FROM "A" MENU, OR TYPE @EMULATE FROM TOOLKIT.  FOR ADCAM 0 ANSWER ALL QUERIES WITH CARRIAGE RETURNS.

14. TO STORE DATA, ENTER EMULATE MODE (STEP 13)

 14.1   HIT MENU KEY

 14.2   SELECT SETUP MCA PARAMETERS

 14.3   SET PRIMARY SOURCE TO MCB

 14.4   SELECT ADCAM # AND SEGMENT # (1)

 14.5   WHEN DONE, HIT EXIT, SELECT SAVE PRIMARY SPECTRUM, GIVE NAME, DO

15. WHEN DONE, HIT EXIT TO RETURN TO STEP ONE FOR NEXT SPECTRUM

16. PLEASE UPDATE GEMLOG.DOC WITH NAMES OF STORED SPECTRA.  PUT NAME OF FLOPPY IN COMMENT SECTION IF NOT ON HARD DISC.

```
C*******************************************************************
C       STP2D8.FTN
C       MODIFIED 10/1/84 TO MAKE VARIABLE Y 'VIRTUAL' FOR 8000 CHANNELS
C       FORMERLY STPBG.FOR FROM PDP-11 RT-11 OPERATING SYS.
C       MODIFIED 1984 FOR ADCAM FILES, I.E., .SPC FILES
C       MODIFIED 9/25/82 BY MCGARRY (HAD TO CHANGE READ STATE IN READFI)
C       BACKGROUND IS COMPUTED FROM AVERAGES OVER REGIONS TO THE LEFT AND
C       TO THE RIGHT OF THE PEAK REGION, RESPECTIVELY. THE BACKGROUND
C       REGIONS NEED NOT BUTT UP AGAINST THE PEAK AREA.
C       THIS PROGRAM WILL DO AN ENERGY CALIBRATION, LOCATE PEAKS,
C       CALCULATE AREAS, WIDTHS, PERCENT ERRORS AND COUNTS PER SECOND.
C       THE DATA OF THE SPECTRA MUST BE STORED ON A FILE WITH ANY STANDARD
C       ORTEC .SPC FILE SPECIFICATION.  PEAK AND END OF IRRAD-
C       IATION PARAMETERS ARE ENTERED IN SUBROUTINE PARAMS AND ARE READ
C       FROM FILE LIMITS.DOC OR FROM THE KEYBOARD OR JUST NOT USED. WHEN
C       THEY ARE NOT USED, THE PROGRAM WILL FIND THE PEAK PARAMETERS AND
C       OUTPUT THEM ON FILE LIMITS.PKS. IF THEY ARE INPUT FROM THE KEYBOARD
C       THEY WILL ALSO BE OUTPUT TO LIMITS.TEM. THEY SHOULD BE CONCATENATED
C       TO LIMITS.DOC, IF ONE WISHES TO SAVE THEM.  THE PROGRAM WILL STORE
C       THE INTERCEPT, SLOPE AND SECOND ORDER CONSTANT FOR THE ENERGY CALI-
C       BRATION CURVE ON KEV.DOC FROM THE ENERGY CALIBRATION ROUTINE OR USE
C       AN EXISTING FILE WITH THAT DATA. THE RESULTS ARE WRITTEN ON THE FILE-
C       NAME.RES AS WELL AS ECHOED TO THE TERMINAL IN SUBROUTINE EPEAK.
C
C*******************************************************************
C*******************************************************************
C
C       THE MAIN PROGRAM CALLS SUBROUTINES PARAMS, GETFIL,
C       READFI, CAL, LOCATE, AND EPEAK IN THAT ORDER.
C
C*******************************************************************
C*******************************************************************
C
        DIMENSION IWL(40),IWH(40),PK(40),
       &IC(40),ID(40),NL(40),NR(40)
        CHARACTER IFILI(20),IFILO(20),REP
        LOGICAL TEM
        DIMENSION IWORD(34)
        CHARACTER*128 TITL1,TITL2,ADATA
        CHARACTER*5 AMONTH
        CHARACTER*1 ADUM
        DOUBLE PRECISION XME(4)
        VIRTUAL Y(8192)
        LOGICAL*1 PRMS
        COMMON/ADCAM/IADCAM
        COMMON /NCHAN/ ICHAN
        COMMON /PKLIM/ NUM,IWL,IWH,IC,ID,NL,NR,PK,IBW,SIG
        COMMON /FILNMS/ IFILI,IFILO
        COMMON/IPUT/XME
        COMMON/LAMBD/IYRO,IDAYO,IHRO,MINITO,SECO,ALAM,CASRAT,
       &IYR,IDAY,IHR,MINIT,SEC,PRMS
        COMMON /SPEC/TIME,CLTM,PLSRTE
    2 FORMAT(A1)
    3 FORMAT(I4)
```

```
    4 FORMAT(I1)
    5 FORMAT('.P') ! THIS IS USED BY PGM EDPAGE
      TYPE 8
    8 FORMAT(/' LOCK SHIFT KEY'/)
      CALL PARAMS(TEM)
   10 CALL GETFIL(5, IFILI, IFILO)
C FORMERLY SUBROUTINE READFI
C
C     READ AND PRINT TITLE AND COLUMN HEADS AND INPUT SPECTRUM DATA
C     FROM THE INPUT FILE. IT IS ONLY CALLED BY THE MAIN PROGRAM.
C
      READ(3'1)(IWORD(I),I=1,34)
      READ(3'IWORD(5))ADATA
      PRINT *,ADATA
      READ(ADATA,7)ADATA(:16),NDAY,AMONTH,NYR,IHR,ADUM,MINIT,ADUM,
     1NSEC,TIME,CLTM
    7 FORMAT(A16,I2,A5,I2,I5,A1,I2,A1,I2,2F11.0)
      PRINT 7,ADATA(:16),NDAY,AMONTH,NYR,IHR,ADUM,MINIT,ADUM,NSEC,
     1TIME,CLTM
      WRITE(4,7)ADATA(:16),NDAY,AMONTH,NYR,IHR,ADUM,MINIT,ADUM,NSEC,
     1TIME,CLTM
      SEC=NSEC
      IYR=1900+NYR
      NLP=IMOD(NYR,4)
      IF(AMONTH.EQ.'-JAN-')IMON=1
      IF(AMONTH.EQ.'-FEB-')IMON=2
      IF(AMONTH.EQ.'-MAR-')IMON=3
      IF(AMONTH.EQ.'-APR-')IMON=4
      IF(AMONTH.EQ.'-MAY-')IMON=5
      IF(AMONTH.EQ.'-JUN-')IMON=6
      IF(AMONTH.EQ.'-JUL-')IMON=7
      IF(AMONTH.EQ.'-AUG-')IMON=8
      IF(AMONTH.EQ.'-SEP-')IMON=9
      IF(AMONTH.EQ.'-OCT-')IMON=10
      IF(AMONTH.EQ.'-NOV-')IMON=11
      IF(AMONTH.EQ.'-DEC-')IMON=12
      IDAY=(IMON-1)*30+NDAY
      IF(IMON.EQ.2.OR.IMON.EQ.6.OR.IMON.EQ.7)IDAY=IDAY+1
      IF(IMON.EQ.3)IDAY=IDAY-1
      IF(IMON.EQ.8)IDAY=IDAY+2
      IF(IMON.EQ.9.OR.IMON.EQ.10)IDAY=IDAY+3
      IF(IMON.EQ.11.OR.IMON.EQ.12)IDAY=IDAY+4
C    IF LEAP YEAR(NLP=0)ADD A DAY AFTER FEBRUARY
      IF(NLP.EQ.0.AND.IMON.GE.3)IDAY=IDAY+1
      READ(3'IWORD(6))TITL1
      READ(3'IWORD(7))TITL2
      PRINT *,TITL1,TITL2
      WRITE(4,40)TITL1,TITL2
   40 FORMAT(4(1X,A64/))
      INC=0
      DO 20 IN=IWORD(31),IWORD(31)+255
      READ(3'IN)(Y(I),I=INC+1,INC+32)
   20 INC=INC+32

      ICHAN=INC
```

```
C      FORMERLY   END    ! OF SUBROUTINE READFI
D      WRITE(4,5)
   21 TYPE 200
  200 FORMAT('  ENTER ADCAM NUMBER(0 OR 1)')
       ACCEPT 4,IADCAM
       IF (IADCAM.NE.0 .AND. IADCAM.NE.1)GO TO 21
       TYPE 9
    9 FORMAT(/' THE ENERGY CALIBRATION DATA SHOULD BE ON KEV.DOC'//
      &'$DO YOU WANT A NEW ENERGY CALIBRATION ?  : ')
       ACCEPT 2, REP
       IF(REP.EQ.'Y') CALL CAL(Y)
       TYPE 13
   13 FORMAT(' DO YOU WANT TO LOCATE PEAKS, USING THE ENERGY '
      &//'$CALIBRATION FROM KEV.DOC ?    ')
       ACCEPT 2, REP
       IF(REP.EQ.'Y') CALL LOCATE(Y)
       TYPE 12
   12 FORMAT('$DO YOU WANT TO ANALYSE THE SPECTRUM : ')
       ACCEPT 2, REP
       IF(REP.EQ.'Y') CALL EPEAK(Y)
       TYPE 15
   15 FORMAT(' DO YOU WANT TO INPUT ANOTHER FILE USING THE SAME',
      &/'$ PARAMETERS AND INPUT FORMAT ?   ')
         CLOSE(3)
         CLOSE(4)
       ACCEPT 2, REP
       IF(REP.EQ.'Y') GO TO 10
       TYPE 25, IFILO
   25 FORMAT(7X'THE RESULTS ARE ON FILE ',20A1)
       IF (TEM) GO TO 27
       OPEN(UNIT=2,FILE='LIMITS.PKS',STATUS='OLD')
C  THE PREVIOUS LINE MODIFIED FROM RT-11 VERSION WHICH USED CALL ASSIGN
       DO 65 I=1,NUM
   65 WRITE(2,105)IWL(I),IC(I),NL(I),ID(I),IWH(I),NR(I),PK(I)
  105 FORMAT(6I8,F8.2)
       CLOSE(2)
       TYPE 28
   28 FORMAT(' THE PEAK LIMITS HAVE BEEN WRITTEN ON LIMITS.PKS.')
       GO TO 30
   27 TYPE 26
   26 FORMAT(7X'DO NOT FORGET FILE LIMITS.TEM  ')
   30 STOP
       END       ! OF MAIN PROGRAM
C*********************************************************************
C
C
       SUBROUTINE PARAMS(TEM)
C
C    THIS SUBROUTINE PROVIDES THE INPUT FROM THE KEYBOARD OR FROM FILE
C    LIMITS.DOC FOR READING THE VARIOUS PARAMETERS FOR THE PROGRAM.
C    THOSE SUPPLIED FROM THE KEYBOARD WILL BE WRITTEN TO FILE LIMITS.TEM
C    SO THAT THEY CAN BE CONCATENATED TO LIMITS.DOC LATER.
C    IF ONE IS NOT INTERESTED IN PARAMETERS THE PROGRAM WILL
C    SUPPLY A SET THAT WILL GIVE AN OUTPUT OF PEAK SUMS ONLY.
```

```
C      IN THE CASE OF A DECAY LIKE BA, WHICH DECAYS TO LA, ONE MUST
C      INPUT BOTH OF THE HALF LIVES, THE LONGEST FIRST (BA). IT CALCULATES
C      THE CASCADING RATIO OF BA ACTIVITY TO LA ACTIVITY AT THE END OF
C      IRRADIATION, (TA-TB)/TA.
C
       DIMENSION IPK(40),IWL(40),IWH(40),PK(40),
      &IC(40),ID(40),NL(40),NR(40)
       DOUBLE PRECISION KEY, TOKEN
       CHARACTER REPL
       LOGICAL*1 PRMS,TEM
       COMMON/LAMBD/IYRO,IDAYO,IHRO,MINITO,SECO,ALAM,CASRAT,
      &IYR,IDAY,IHR,MINIT,SEC,PRMS
       COMMON /FILNMS/ IFILI,IFILO
       COMMON /NCHAN/ ICHAN
       COMMON /PKLIM/ NUM,IWL,IWH,IC,ID,NL,NR,PK,IBW,SIG
       DATA NUM,IYRO,IYR,IDAYO,IDAY,IHRO,IHR,MINITO,MINIT /9*0/,
      &TA /1.0 E13 /, TB,SECO,SEC /3*0.0/,PRMS/.FALSE./,IBW/3/,
      &SIG /40./
       TEM=.FALSE.
       TYPE 1
1      FORMAT ('$DO YOU WANT TO USE INTEGRATION PARAMETERS ?  ')
2      FORMAT(1A1)
       ACCEPT 2, REPL
       IF(REPL.NE.'Y') GO TO 120
       PRMS = .TRUE.
       TYPE 8
8      FORMAT('$DO YOU WISH TO READ THEM FROM LIMITS.DOC ?  ')
       ACCEPT 2,REPL
       IF (REPL.EQ.'Y') GO TO 100
       TYPE 9
9      FORMAT(' THESE PARAMETERS WILL BE WRITTEN ON FILE LIMITS.TEM',
      &/' SO THAT YOU CAN LATER ADD THEM TO FILE LIMITS.DOC')
       OPEN(UNIT=2,FILE='LIMITS.TEM',STATUS='NEW')
       TYPE 101
       ACCEPT 102, KEY
       WRITE(2,102) KEY
       WRITE(5,122) KEY
       TEM = .TRUE.
       TYPE 3
3      FORMAT(' GIVE YEAR,DAY,HR,MIN,SEC FOR THE END OF IRRADIATION  '
      &/' IN 4I10, 1F10.0 FORMAT, SEPARATED BY COMMAS  '/)
       ACCEPT 4,IYRO,IDAYO,IHRO,MINITO,SECO
       TYPE 4,IYRO,IDAYO,IHRO,MINITO,SECO
4      FORMAT(4I10,F10.0)
       TYPE 5
5      FORMAT(/' GIVE THE HALF LIFE OF THE SINGLE DECAY OR THE TWO HALF'
      &/' LIVES FOR A CASCADING DECAY LIKE BA-LA, LONGEST FIRST AND IN'
      &/' F OR E14.0 FORMAT, SEPARATED BY A COMMA, IF TWO '/)
       ACCEPT 6, TA, TB
       TYPE 206, TA, TB
6      FORMAT(2F14.0)
206    FORMAT(1X,2E14.6)
       TYPE 7
7      FORMAT(/'$UNITS OF TIME FOR THE FIRST (Y, D, H, M, OR S) ?  ')
```

```
      ACCEPT 2,REPL
      IF(REPL.EQ.'Y') GO TO 20
      IF(REPL.EQ.'D') GO TO 21
      IF(REPL.EQ.'H') GO TO 22
      IF(REPL.EQ.'M') GO TO 23
      IF(REPL.EQ.'S') GO TO 26
20    TA=TA*365.25
21    TA=TA*24.
22    TA=TA*60.
23    TA=TA*60.
26    IF(TB.LT.1.0E-5) GO TO 50
27    TYPE 28
28    FORMAT('$UNITS OF TIME FOR THE SECOND (Y, D, H, M, OR S) ?  ')
      ACCEPT 2,REPL
      IF(REPL.EQ.'Y') GO TO 30
      IF(REPL.EQ.'D') GO TO 31
      IF(REPL.EQ.'H') GO TO 32
      IF(REPL.EQ.'M') GO TO 33
      IF(REPL.EQ.'S') GO TO 50
30    TB=TB*365.25
31    TB=TB*24.
32    TB=TB*60.
33    TB=TB*60.
50    TYPE 51
51    FORMAT(' HOW MANY PEAKS DO YOU WISH TO USE (I2)   ')
      ACCEPT 53,NUM
      TYPE 52
52    FORMAT(/'$PEAK SIGNIFICANCE : ')
      ACCEPT 54,SIG  ! USED AFTER ST 70 IN SUB CHPEAK
54    FORMAT(F6.0)
53    FORMAT(I6)
55    FORMAT(5X,' ENTER THE LEFT BKG. CHANNEL, PEAK CHANNEL AND',/,
     &' THE NUMBER OF BKG. CHANNELS TO BE AVERAGED ON THE LEFT. ENTER',/
     &' UPPER PEAK CHANNEL ,THE BKG. CHANNEL, AND NUMBER OF CHANNELS ',/
     &' TO BE AVERAGED ON THE RIGHT. FOLLOW THIS WITH THE PEAK POSI-',/
     &' TION. ALL ENTRIES IN 6I, 1F FORMAT ON ONE LINE WITH A COMMA',/
     &' BETWEEN EACH.')
      TYPE 55
      DO 63 I=1,NUM
      TYPE 59,I
63    ACCEPT 61,IWL(I),IC(I),NL(I),ID(I),IWH(I),NR(I),PK(I)
59    FORMAT(7X,' PEAK NO.  ',I4)
61    FORMAT (6I10,F14.0)
      WRITE(2,104)NUM,IYRO,IDAYO,IHRO,MINITO,SECO,SIG,TA,TB
      DO 65 I=1,NUM
65    WRITE(2,105)IWL(I),IC(I),NL(I),ID(I),IWH(I),NR(I),PK(I)
      CLOSE(2)
      GO TO 120
100   OPEN(UNIT=2,FILE='LIMITS.DOC',STATUS='OLD')
      TYPE 101
101   FORMAT('$WHAT IS THE ISOTOPE KEYWORD ?  ')
      ACCEPT 102,KEY
102   FORMAT(A8)    ! DOUBLE PRECISION = 8 CHARACTERS
      WRITE(5,122) KEY
```

```fortran
  122 FORMAT(1X,A8)
  103 READ(2,102,END=110) TOKEN
      IF(TOKEN.NE.KEY)GO TO 103
  104 FORMAT(5I8,F8.0,F8.0,/,2E16.6)
      READ(2,104)NUM,IYRO,IDAYO,IHRO,MINITO,SECO,SIG,TA,TB
      WRITE(5,104)NUM,IYRO,IDAYO,IHRO,MINITO,SECO,SIG,TA,TB
      DO 106 I=1,NUM
      READ(2,105)IWL(I),IC(I),NL(I),ID(I),IWH(I),NR(I),PK(I)
  106 WRITE(5,105)IWL(I),IC(I),NL(I),ID(I),IWH(I),NR(I),PK(I)
  105 FORMAT(6I8,F8.2)
      CLOSE(2)
      GO TO 120
  110 TYPE 112
  112 FORMAT(' KEY WAS NOT FOUND IN LIMITS.DOC FILE  ')
  120 ALAM=ALOG(2.)/TA
      CASRAT=(TA-TB)/TA
      RETURN ! TO MAIN PROGRAM
      END    ! OF SUBROUTINE PARAMS
C
C
C**********************************************************************
C
C
C**********************************************************************
C
      SUBROUTINE GETFIL(LUN,AREAI,AREAO)
C
C    THIS SUBROUTINE IS CALLED FROM MAIN IN ORDER TO ASSIGN
C    THE INPUT AND OUTPUT FILES TO LOGICAL UNITS
C
      CHARACTER*20 AREAI, AREAO
      TYPE 1
    1 FORMAT('$ENTER INPUT FILENAME ')
      READ(LUN,9) IQI, AREAI(:IQI)
      IQ=IQI
      K=0
      DO 10 I = 1,IQ
   10 IF (AREAI(I:I) .EQ. '.') K = I    ! DECIMAL POINT
      IF (K.EQ.0)AREAI(IQ+1:IQ+4)='.SPC'
   14 IF(K .NE. 0) IQ=K-1
      AREAO(:IQ) = AREAI(:IQ)
      AREAO(IQ+1:IQ + 4) = '.RES'
    9 FORMAT(Q, A20)
      WRITE(LUN,8)AREAI(:IQ+4),AREAO(:IQ+4)
    8 FORMAT(2X,A20,3X,A20)
      OPEN(UNIT=3,FILE=AREAI(:IQ+4),STATUS='OLD',ACCESS='DIRECT',
     1RECORDSIZE=32)
      OPEN(UNIT=4,FILE=AREAO(:IQ+4),STATUS='NEW')
      RETURN ! TO MAIN PROGRAM
      END    ! OF SUBROUTINE GETFIL
C**********************************************************************
C


C**********************************************************************
```

```fortran
C
      SUBROUTINE CAL(Y)
C
C     SUBROUTINE CAL USES SUBROUTINE ELOC TO FIND PEAKS
C     IT IS ONLY CALLED BY THE MAIN PROGRAM
C

      DIMENSION X(40),E(40)
      VIRTUAL Y(8192)
      COMMON /PEAK/X,E,N
      COMMON/ADCAM/IADCAM
      DATA SX,SX2,SX3,SX4,SY,SXY,SX2Y/7*0.0/
      CALL ELOC(Y)
      TYPE 2
    2 FORMAT(' PEAK      CHANNEL      ENERGY'/)
      DO 11 I=1,N
   11 TYPE 12,I,X(I),E(I)
C
C     CALCULATE PEAK ENERGY : E = A + BX + CX**2
C
   12 FORMAT(I4,5X,F9.2,2X,F9.2)
      IF(N.EQ.1) GO TO 20
      DO 10 I=1,N
      SX=SX+X(I)          ! SUM X'S
      X2=X(I)*X(I)
      SX2=SX2+X2          ! SUM X**2
      SX3=SX3+X2*X(I)     ! X**3
      SX4=SX4+X2*X2       ! SUM X**4
      SY=SY+E(I)          ! SUM Y'S
      SXY=SXY+X(I)*E(I)   ! SUM X*Y
      SX2Y=SX2Y+X2*E(I)   ! SUM Y*X**2
   10 CONTINUE
      T=N
      CKSX3=T*SX3-SX*SX2          ! THIS IS A CLOSED SOLUTION TO EQ (8-4)
      CKSX4=T*SX4-SX2*SX2         ! OF BEVINGTONS'S DATA REDUCTION AND ERROR
      CKSX2=T*SX2-SX*SX           ! ANALYSIS FOR THE PHYSICAL SCIENCES
      CKSXY=T*SXY-SX*SY
      CKSX2Y=T*SX2Y-SX2*SY
      C1=CKSX2*CKSX2Y-CKSX3*CKSXY
      C2=CKSX2*CKSX4-CKSX3*CKSX3
      C=C1/C2
      B1=CKSXY-C*CKSX3
      B=B1/CKSX2
      A1=SY-C*SX2-B*SX
      A=A1/T
      GO TO 30
   20 A=0.0
      C=0.0
      B=E(1)/X(1)
   30 TYPE 40
   40 FORMAT(/'  DATA FILE KEV.DOC = '/)
      TYPE 45,A,B,C
   45 FORMAT('  A, B, & C = ',1P3E14.7)

      IF(IADCAM.GT.0)GO TO 200
```

```
        OPEN(UNIT=2,FILE='KEV0.DOC',STATUS='OLD')
        GO TO 201
   200 OPEN(UNIT=2,FILE='KEV1.DOC',STATUS='OLD')
   201 WRITE(2,50)A,B,C
        CLOSE(2)
    50 FORMAT(3E17.10)
        TYPE 55
    55 FORMAT(/' FILE KEV UPDATED. '/)
        RETURN ! TO THE MAIN PROGRAM
        END    ! OF SUBROUTINE CAL
C*****************************************************************************
C
        SUBROUTINE ELOC(Y)

C     IT RETURNS PEAK LOCATIONS WITH THE ENERGY SUPPLIED BY THE
C     TERMINAL FOR THE ENERGY CALIBRATION, WHICH IS DONE BY CAL.
C     IT CALLS SUBROUTINE CHPEAK.
C
        DIMENSION X(40),E(40)
        VIRTUAL Y(8192)
        COMMON /NCHAN/ ICHAN
        COMMON /PEAK/X,E,NUM
        NUM=0
        TYPE 1
     1 FORMAT(/'$PEAK SIGNIFICANCE : ')
        ACCEPT 2,SIG  ! USED AFTER ST 70 IN SUB CHPEAK
     2 FORMAT(F6.0)
        TYPE 3
     3 FORMAT(//4X,'PEAK LOCATOR'//'   CHAN  ENERGY '/)
        I=1
    10 CALL CHPEAK(CPK,IH,IL,I,SIG,Y)
        IF(CPK .LT. 0) GO TO 100
        NUM=NUM+1
        X(NUM)=CPK
        TYPE 4,CPK
     4 FORMAT(1H ,$F7.1,2X/)
        ACCEPT 5,E(NUM)
     5 FORMAT(F10.0)
        IF(E(NUM).LT.1.0) NUM=NUM-1
   100 IF(I.LT.ICHAN) GO TO 10
        RETURN ! TO SUBROUTINE CAL
        END    ! OF SUBROUTINE ELOC
C
C*****************************************************************************
C
        SUBROUTINE CHPEAK(CPK,IH,IL,I,SIG,Y)
C
C     THIS SUBROUTINE LOCATES THE PEAK FROM THE INPUT DATA.
C     IT STARTS AT CHANNEL I AND STEPS FROM CHANNEL TO CHANNEL
C     LOOKING FOR PEAKS. IT IS CALLED BY SUBROUTINES ELOC, EPEAK
C     AND LOCATE. A PEAK MUST BE AT LEAST TWO CHANNELS WIDE IN WHICH
C     CASE THE SECOND CHANNEL COUNT MUST BE LARGER THAN THE FIRST.
C     THE ALTERNATIVE IS TO ENTER THE REGION OF INTEREST INTO THE
C     LIMITS.DOC FILE, WHICH IS ACCESSED BY SUBROUTINE PARAMS, AND
```

```
C        ANSWER NO TO THE GAINSHIFT REQUEST IN SUBROUTINE EPEAK. THEN
C        THE PROGRAM WILL NOT ACCESS THIS SUBROUTINE.
C
         VIRTUAL Y(8192)
         COMMON /SPEC/ TIME,CLTM
         ISTEP=1
         I=I+ISTEP
         IF (Y(I).LT.0.)Y(I)=-Y(I)
         YL=Y(I-1)+SQRT(Y(I))
         YH=Y(I+1)-SQRT(Y(I))
         IF((YL.LT.Y(I)).AND.(Y(I).LT.YH)) GO TO 50
         GO TO 100
     50 IL=I
     52 DIF=Y(I+1)-Y(I)
         IF(DIF.LT.0.0) GO TO 51
         I=I+1
         GO TO 52
     51 YP=Y(I)
         IK=I
     53 I=I+1
         YL=Y(I-1)+SQRT(Y(I))
         YH=Y(I+1)-SQRT(Y(I))
         IF((YL.GT.Y(I)).AND.(Y(I).GT.YH)) GO TO 53
     54 IH=I
         YL=Y(IL)
         YH=Y(IH)
         BK=(YL+YH)/2.0
         PW=IH-IL
         IF(PW.LE.3) GO TO 100
     70 CNT=YP-BK
         TOT=YP+BK
         IF(TOT.LE.0.) GO TO 100
         IF(CNT.LT.SIG*SQRT(TOT)) GO TO 100 ! SIG = PEAK SIGNIFICANCE
         SXY=0.                             ! SELECTION CRITERION
         SY=0.
         ILOW=IK-2
         IHI=IK+2
         DO 75 K=ILOW,IHI
         CK=K
         SXY=SXY+CK*Y(K)*Y(K)
         SY=SY+Y(K)*Y(K)
     75 CONTINUE
         CPK=SXY/SY
         GO TO 101
    100 CPK= -10.0
    101 RETURN ! TO SUBROUTINE ELOC, LOCATE, OR EPEAK
         END    ! END OF SUBROUTINE CHPEAK
C*******************************************************************************
C
         SUBROUTINE LOCATE(Y)
C
C        SUBROUTINE LOCATE DETERMINES PEAKS AND BOUNDARIES AND
C        PASSES INFORMATION TO SUBROUTINE EPEAK VIA COMMON BLOCKS.
C        IT WRITES ON THE OUTPUT FILE FILENAME.RES.
```

```
C
      DIMENSION EGY(40),IPK(40),IWL(40),IWH(40),PK(40),
     &IC(40),ID(40),NL(40),NR(40)
      VIRTUAL Y(8192)
      COMMON/ADCAM/IADCAM
      COMMON /NCHAN/ ICHAN
      COMMON /ENEG/ EGY
      COMMON /PKLIM/ NUM,IWL,IWH,IC,ID,NL,NR,PK,IBW,SIG
      IF(IADCAM.GT.0)GO TO 200
      OPEN(UNIT=2,FILE='KEVO.DOC',STATUS='OLD')
      GO TO 201
  200 OPEN(UNIT=2,FILE='KEV1.DOC',STATUS='OLD')
  201 READ(2,1)A,Z,C
    1 FORMAT(3E17.10)
      CLOSE(2)
      NUM=0
      TYPE 2
    2 FORMAT(/'$PEAK SIGNIFICANCE : ')
      ACCEPT 3,SIG   ! USED AFTER STMT 70 IN SUB CHPEAK
    3 FORMAT(F6.0)
      WRITE(4,4)
      WRITE(4,4)
    4 FORMAT(' ****************************************
     &*********************************')
      WRITE(4,5)
      TYPE 5
    5 FORMAT(4X,'PEAK LOCATOR'//' PEAK    CHAN   ENERGY  FWIDTH'/)
      I=1
   10 CALL CHPEAK(CPK,IH,IL,I,SIG,Y)
      IF(CPK .LT. 0) GO TO 100
      EN=A+Z*CPK+C*CPK*CPK
      NUM=NUM+1
      EGY(NUM)=EN
      IFW=IH-IL+1
      IWL(NUM)=IL
      IWH(NUM)-IH
      IC(NUM)=IL
      ID(NUM)=IH
      NL(NUM)=IBW
      NR(NUM)=IBW
      PK(NUM)=CPK
      WRITE(4,6)NUM,CPK,EN,IFW
      TYPE 6,NUM,CPK,EN,IFW
    6 FORMAT(I5,F8.2,F9.2,3X,I5)
  100 IF(I.LT.ICHAN) GO TO 10
      RETURN ! TO SUBROIUTINE EPEAK
      END    ! OF SUBROUTINE LOCATE
C*******************************************************************************
C
      SUBROUTINE EPEAK(Y)
C
C      SUBROUTINE EPEAK USES PEAKS AND BOUNDARIES FOUND BY
C      SUBROUTINE LOCATE OR THOSE FROM SUBROUTINE PARAMS
C      IT CREATES ALL THE OUTPUT RESULTS WHICH ARE WRITTEN
```

```
C        TO FILENAME.RES. IT WILL ADJUST THE PEAK PARAMETERS WITH RESPECT
C        TO THE PEAK FOR CHANGES IN GAIN OF THE PEAK. IT CALLS SUBROUTINE
C        CHPEAK AND LOCATE. IT IS CALLED BY THE MAIN PROGRAM.
C
         DIMENSION EGY(40),IPK(40),IWL(40),IWH(40),PK(40),
        &IC(40),ID(40),NL(40),NR(40)
         DIMENSION BLOW(40),BHIGH(40),BKG(40),TOTAL(40),ER(40)
         CHARACTER IFILI(20),IFILO(20),IREP
         LOGICAL*1 GAIN,PRMS,PULSE
         DOUBLE PRECISION XME(4)
         VIRTUAL Y(8192)
         COMMON/ADCAM/IADCAM
         COMMON/IPUT/XME
         COMMON/LAMBD/IYRO,IDAYO,IHRO,MINITO,SECO,ALAM,CASRAT,
        &IYR,IDAY,IHR,MINIT,SEC,PRMS
         COMMON /ENEG/ EGY
         COMMON /SPEC/ TIME,CLTM,PLSRTE
         COMMON /ENFIT/ A,Z,C
         COMMON /PKLIM/ NUM,IWL,IWH,IC,ID,NL,NR,PK,IBW,SIG
         COMMON /FILNMS/ IFILI,IFILO
         DATA GAIN/.FALSE./
         IF(IADCAM.GT.0)GO TO 200
         OPEN(UNIT=2,FILE='KEVO.DOC',STATUS='OLD')
         GO TO 201
  200 OPEN(UNIT=2,FILE='KEV1.DOC',STATUS='OLD')
  201 READ(2,1)A,Z,C
    1 FORMAT(3E17.10)
    2 FORMAT(A1)
         CLOSE(2)
         TYPE 22
   22 FORMAT('$DO YOU WANT TO CORRECT FOR GAIN SHIFT ?  ')
         ACCEPT 2,IREP
         IF(IREP.EQ.'Y') GAIN = .TRUE.
C
C        OUTPUT RESULTS
C
         TYPE 3
         WRITE(4,3)
    3 FORMAT(' ****************************************
        &*********************************')
C
         WRITE(4,4)IFILI
         TYPE 4,IFILI
    4 FORMAT(/'          DATA FILENAME: '21A1)
         WRITE(4,5)TIME
         TYPE 5,TIME
    5 FORMAT(/'          LIVE  TIME (SEC): ',F8.0,/)
   21 CONTINUE
    7 FORMAT(' PEAK CHANNELS ENERGY      BKGND         AREA   % ERR
        &      CTS/SEC'/)
    9 FORMAT(/)
C
C        DETERMINE AREAS
C
```

```
      IDAYS = 0
      SECS = 0.0
      IF(.NOT. PRMS) GO TO 72
   10 IF(IYRO.LE.IYR)GO TO 70
      WRITE(5,12)IYRO,IYR
   12 FORMAT(' IYRO,',I5,', CANNOT BE LARGER THEN IYR,',I5)
      TYPE 18
   18 FORMAT(' INPUT NEW IYRO   ')
      ACCEPT 19,IYRO
   19 FORMAT(I6)
      GO TO 10
   70 IF(IYRO.EQ.IYR)GO TO 71
      IF(MOD(IYRO,4).EQ.0)IDAYS =IDAYS+1
      IYRO=IYRO+1
      IDAYS=IDAYS+365
      GO TO 70
   71 IDAYO=IDAYO-IDAYS
      FDAYS=IDAY-IDAYO
      FIHRS=FDAYS*24.+(IHR-IHRO)
      FMINIT=FIHRS*60.+(MINIT-MINITO)
      SECS=FMINIT*60.+(SEC-SECO)
   72 TYPE 88,SECS
      WRITE(4,88)SECS
      WRITE(4,7)
      TYPE 7
   88 FORMAT(' SECONDS AFTER END OF IRRADIATION = ',E14.6,/)
      PULSE = .FALSE.
      IF(PLSRTE.LT.2) GO TO 885
      PULSE = .TRUE.        ! PULSER PEEK IN SPECTRUM
      NNUM=NUM+1
      DO 870 I=NNUM,2,-1
      EGY(I)=EGY(I-1)
      IPK(I)=IPK(I-1)
      IWL(I)=IWL(I-1)
      IWH(I)=IWH(I-1)
      PK(I)-PK(I-1)
      IC(I)=IC(I-1)
      ID(I)=ID(I-1)
      NL(I)=NL(I-1)
  870 NR(I)=NR(I-1)
      EGY(1)=EGY(NNUM)
      IPK(1)=IPK(NNUM)
      IWL(1)=IWL(NNUM)
      IWH(1)=IWH(NNUM)
      PK(1)=PK(NNUM)
      IC(1)-IC(NNUM)
      ID(1)=ID(NNUM)
      NL(1)=NL(NNUM)
      NR(1)=NR(NNUM)
      GO TO 886
  885 PLSRTE=1      ! USED IN ST 86
      PLSRAT=1      ! USED IN ST 86
  886 DO 40 I=1,NUM
      IDIFF=0
```

```
      IF(.NOT. PRMS)GO TO 15
      IF(.NOT. GAIN)GO TO 15
      IF((PK(1).LT.IWL(I)).OR.(PK(I).GT.IWH(I)))TYPE 16
   16 FORMAT(' INPUT PEAK IS OUT OF BOUNDS ')
      LL=IWL(I)-NL(I)
      ILL=LL
      LH=IWH(I)+NR(I)
   17 CALL CHPEAK(CPK,IH,IL,ILL,SIG,Y)
      IF(CPK.GT.5.0)GO TO 14
      IF(ILL.LT.LH)GO TO 17
      TYPE 13,LL,LH,SIG
   13 FORMAT(' NO PEAK WAS FOUND BETWEEN CHANNEL',I5,' AND',I5,
     &/' WITH PEAK SIGNIFICANCE =',F5.1)
      GO TO 40
   14 CPKK=CPK+0.5
      TPK=PK(I)+0.5
      IDIFF = IFIX(CPKK)-IFIX(TPK)
      PK(I)=PK(I)+IDIFF
      IWH(I)=IWH(I)+IDIFF
      IWL(I)=IWL(I)+IDIFF
      IC(I)=IC(I)+IDIFF
      ID(I)=ID(I)+IDIFF
      IF(IDIFF.NE.0)WRITE(5,8)IDIFF
    8 FORMAT(' THE PROGRAM PARAMETERS HAVE BEEN SHIFTED',I5,
     &/' CHANNELS DUE TO A CHANGE IN PEAK POSITION.'/)
   15 T=0.0
      IMINS=IC(I)
      IPLUS=ID(I)
      CC=FLOAT(IC(I))
      DD=FLOAT(ID(I))
      DO 41 J=IMINS,IPLUS
      T=T+Y(J)
   41 CONTINUE
C
C        PEAK LOCATION
C
      SXY=0.
      SY=0.
      DO 45 K=IMINS,IPLUS
      CK=K
      SXY=SXY+CK*Y(K)*Y(K)
   45 SY=SY+Y(K)*Y(K)
      CPK=SXY/SY
C
C      PROGRAM USES NL AND NR CHANNELS ON EACH SIDE TO DETERMINE
C      BACKGROUND AT PEAK BOUNDARIES. DEFAULT IS IBW.
C
   42 IMINS=IWL(I)
      IPLUS=IWH(I)
      BL=0.0
      DO 64 JK=1,NL(I)
   64 BL=BL+Y(IMINS-JK+1)
      NTL=NL(I)
      XL=FLOAT(IMINS)-(FLOAT(NTL)/2.)
```

```
      BL=BL/FLOAT(NTL)
      BLOW(I)=BL
      BH=0.0
      DO 66 JK=1,NR(I)
   66 BH=BH+Y(IPLUS+JK-1)
      NTR=NR(I)
      XH=FLOAT(IPLUS)+(FLOAT(NTR)/2.0)
      BH=BH/FLOAT(NTR)
      BHIGH(I)=BH
C
C        BACKGROUND EQUATION
C
      B=BL*(CPK-CC+0.5)+BH*(DD-CPK+0.5)
      YAA=T-B
      BKG(I)=B
      TOTAL(I)=T
C
C     PEAK ENERGY
C
      EGY(I)=A+Z*CPK+C*CPK*CPK
C
C   PERCENT ERROR = SQRT (T+((U-L)/2)*(BL+BH))/N
C
      CONST=(DD-CC)/2.
      SUM=BL+BH
      ER(I)=SQRT(T+CONST*SUM)/YAA
      ER(I)=ER(I)*100.
      BINT=YAA*CLTM/TIME
      RATEUN=BINT/CLTM        !DEADTIME CORRECTED RATE
C
C  RATESC IS RATE CORRECTED FOR WITHIN COUNTING TIME DECAY
C  FOR VERY SMALL CORRECTIONS, THIS CORRECTION IS NOT MADE
C
      IF(.NOT. PULSE)GO TO 74
      RATESC=BINT/CLTM
      RATETO=RATESC
      PLSRAT=RATESC        ! ANALYZER PULSER COUNT
      PULSE = .FALSE. ! (PLSRTE/PLSRAT) HAS BEEN DETERMINED FOR ST 86
      CORR=PLSRTE/PLSRAT
      GO TO 90
   74 CORR=1.
      IF(CLTM*ALAM .LT. .001)GO TO 87
      RATESC=BINT*ALAM/(1-EXP(-ALAM*CLTM))
      GO TO 86
   87 RATESC=BINT/CLTM
   86 RATETO=CASRAT*RATESC*CORR*EXP(ALAM*SECS)
   90 TYPE 81,I,IC(I),ID(I),EGY(I),B,YAA,ER(I),RATEUN
      WRITE(4,81)I,IC(I),ID(I),EGY(I),B,YAA,ER(I),RATEUN
   81 FORMAT(1X,I4,I4,I5,F7.1,F10.1,F13.1,F7.2,1X,1PE12.5)
      IF (SECS .LT. 1.) GO TO 40
      TYPE 89,RATESC,RATETO
      WRITE(4,89)RATESC,RATETO
   89 FORMAT (/1X,'DECAY CORRECTED RATE = ',F10.3,'  COUNTING
     & RATE AT TO = ',1PE12.5/)
```

```
   40 CONTINUE
      TYPE 9              ! LINE FEED
      TYPE 3
      WRITE(4,3)
      WRITE (4,50)
   50 FORMAT (/,15X,'DETAILS REGARDING THE BACKGROUND SUBTRACTION'/)
      WRITE (4,55)
      WRITE (4,56)
   55 FORMAT(2X,'PEAK   AREA      LOW BKG   HIGH BKG'
     &,'   BLOW    BHIGH    BL/BH    BKG/T  % ERROR')
   56 FORMAT(2X,'        LIMITS     LIMITS      LIMITS')
      WRITE (4,9)
   57 FORMAT(2X,I2,1X,I4,'-',I4,1X,I4,'-',I4,1X,I4,'-',I4,
     &1X,F7.1,1X,F7.1,1X,F7.3,1X,F7.4,1X,F7.2)
      DO 58 I=1,NUM
D     WRITE (4,59) CPK,IC(I),ID(I),NL(I),NR(I)
D59 FORMAT (5X,F9.2,I4,I4,I2,I2)
      I1=IWL(I)-NL(I)+1
      I2=IWH(I)+NR(I)-1
      Z1=BLOW(I)/BHIGH(I)
      Z2=BKG(I)/TOTAL(I)
   58 WRITE(4,57)I,IC(I),ID(I),I1,IWL(I),IWH(I),I2,BLOW(I),
     &BHIGH(I),Z1,Z2,ER(I)
      WRITE (4,9)          ! LINE FEED
      WRITE (4,3)
      WRITE(4,3)
   52 WRITE(4,9)
      TYPE 9              ! LINE FEED
      RETURN ! TO MAIN PROGRAM
      END    ! OF SUBROUTINE EPEAK
C
C
C*********************************************************************************
C
```

```
C  PROGRAM SCATFO TO CALCULATE SCATTERING CORRECTION FOR ACTIVATION
C  FOILS IN CAVITY FISSION SOURCE.  DATA MAY BE ENTERED FROM A DISK FILE
C    OR FROM THE KEYBOARD
C      INPUT DATA:
C      ANAME(J)= FOIL ID
C      N       = THE NUMBER OF FOILS
C      IT(J) = FOIL TYPE
C      POS(J)= POSITION OF FOIL IN MILS
.C      TH(J) = THICKNESS OF FOIL IN MILS
       CHARACTER*4  ANAME(50)
       DIMENSION GRAD(50)
       DIMENSION IT(50),POS(50),TH(50)
       OPEN (3,STATUS='NEW')
       TYPE 97
   97  FORMAT(1X,'ENTER FILE NUMBER FOR DATA; 5 FOR KEYBOARD')
       READ(5,100)NF
  100  FORMAT(I10)
       IF(NF.NE.5)GO TO 101
       TYPE 99
   99  FORMAT (1X,'ENTER NUMBER OF FOILS, C.R.')
       GO TO 104
  101  OPEN(NF,STATUS='OLD')
  104  READ(NF,100) N
       IF(NF.NE.5)GO TO 103
       TYPE 98
   98  FORMAT(1X,'ENTER FOIL ID IN A4 FORMAT FOLLOWED BY TYPE,'
      &,/' POSITION IN MILS, THICKNESS IN MILS, FOLLOWED BY'/
      &' C.R. AND THEN NEXT FOIL.  USE 1 FOR IN, 2 FOR NI AND 3 FOR AL')
  103  READ(NF,110)(ANAME(J),IT(J),POS(J),TH(J),J=1,N)
       WRITE(7,111)(ANAME(J),IT(J),POS(J),TH(J),J=1,N)
  110  FORMAT(A4,I10,2F10.1)
  111  FORMAT(1X,A4,I10,2F10.1)
       WRITE(3,96)
   96  FORMAT(2X,'ID',6X,'GRAD.',8X,'TYPE',2X,'POSITION',2X,'THICKNESS
      &    FOIL SCAT      MUSC')
       DO 1000 I=1,N
       GRAD(I)=1+21.2E-6*POS(I)*POS(I)
       SCAT=0
       IF(IT(I)-2)400,500,600
  400  DO 450 J=1,N
       IF(I.EQ.J)GO TO 450
       X=ABS(POS(J)-POS(I))
       Y=-.001+(.0054*EXP(-.0728*X))
       Y=Y*TH(J)/10.
  120  FORMAT(1X,E10.3)
       SCAT=SCAT+Y
  450  CONTINUE
       ALSCAT=SCAT+.0046+(.00255*TH(I)/5.)
       GO TO 800
  500  DO 550 J=1,N
       IF(I.EQ.J)GO TO 550
       X=ABS(POS(J)-POS(I))
```

```
      Y=-.0017+(.00356*EXP(-.0815*X))
      Y=Y*TH(J)/10.
      SCAT=SCAT+Y
  550 CONTINUE
      ALSCAT=SCAT-.002+(.0017*TH(I)/10.)
      GO TO 800
  600 DO 650 J=1,N
      IF(I.EQ.J)GO TO 650
      X-ABS(POS(J)-POS(I))
      Y=-.002+(.002*EXP(-.147*X))
      Y=Y*TH(J)/10.
      SCAT=SCAT+Y
  650 CONTINUE
      ALSCAT=SCAT-.011
      GO TO 800
  800 WRITE(7,102)ANAME(I),GRAD(I),IT(I),POS(I),TH(I),SCAT,ALSCAT
      WRITE(3,102)ANAME(I),GRAD(I),IT(I),POS(I),TH(I),SCAT,ALSCAT
  102 FORMAT(1X,A4,F10.4,I10,2F10.1,3X,2E12.3)
 1000 CONTINUE
      CLOSE(3)
      IF (NF.NE.5)CLOSE(NF)
      STOP
      END
```

```
10 REM SAT260 ...11/21/83
20 ! THE DATA VARIABLES IN
30 ! SEQUENCE:
40 !      X1(I)...FIRST COUNTER
50 !      C(I)....Cu
60 !      X3(I)...Cgc
70 !      T(I)....sec*32
80 !      X5(I)...DAY COUNTER
90 !      X6(I)...hhmmss
100 !      X7(I)...TIME INTERVAL
110 REM    F1$ IS THE DATA FILE
120 REM    NAME
125 PRINTER IS 701,80
126 PRINT CHR$(27)&"&l1L"
130 CLEAR
140 DIM C(260),X1(260),X5(260),X3(260),X7(260),X6(260),T(260)
150 DISP "DATA FILE TO BE USED:"
160 INPUT F1$
170 ASSIGN# 1 TO F1$
180 READ# 1,1 ; N
190 FOR I=1 TO N
200 READ# 1,I+1 ; X1(I),C(I),X3(I),T(I),X5(I),X6(I),X7(I)
210 NEXT I
220 ASSIGN# 1 TO *
230 T(1),X1(1),C(1),X3(1),X7(1)=0
240 PRINT "SAT260 DATA ANALYSIS"
250 PRINT "    11/83 version"
260 PRINT
270 PRINT "N = ";N
280 PRINT
300 PRINT "DATA VERIFICATION FOR ";F1$
310 PRINT "          Cl       Cu       Cgc       hhmmss     sec*32     HPsec"
320 FOR I=1 TO N
330 PRINT USING 340 ; I,X1(I),C(I),X3(I),X6(I),T(I),X7(I)
340 IMAGE 3D,2X,7D,3X, 7D,3X,7D,5X,6D,3X,6D,3X,6D
350 NEXT I
360 PRINT
380 FOR I=1 TO N
410 NEXT I
420 PRINT
430 PRINT
440 A1,A0=X5(1)
450 FOR I=2 TO N
460 IF X5(I)<>A1 THEN A1=X5(I)
470 NEXT I
480 PRINT USING 490 ; A0
490 IMAGE "DATA GOES FROM DAY",4D
500 PRINT USING 510 ; A1
510 IMAGE "            TO DAY",4D
520 PRINT
530 DISP
540 PRINT "DIFINING THE IRRADIATION PERIOD"
550 DISP "DEFINING THE IRRADIATION PERIOD"
560 DISP
570 PRINT
580 DISP "WITH WHICH INTERVAL WOULD YOU"
590 DISP "LIKE ANALYSIS TO START ";
600 INPUT Z5
610 IF Z5>1 THEN 630
```

```
620 Z5=2
630 Q5=Z5-1
640 DISP "WITH WHICH INTERVAL DO YOU WANT "
650 DISP "ANALYSIS TO CONCLUDE";
660 INPUT N
670 T(Q5),X1(Q5),C(Q5),X3(Q5),X7(Q5)=0
680 ZO=N-Z5+1 @ REM ZO IS CORRECT DIVISOR FOR AVERAGES INSTEAD OF N, WHEN ZO I
NOT 1
690 PRINT "ANALYSIS INCLUDES INTERVAL:";Z5
700 PRINT "THROUGH INTERVAL:";N
710 PRINT
720 PRINT "E.S.T. AT S.O.I. :";X6(Z5-1)
730 PRINT "E.S.T. AT E.O.I. :";X6(N)
740 PRINT
750 X1=N
760 T=0
770 T1=0
780 C1,S1=0
790 FOR I=Q5 TO N
800 T(I)=T(I)/32
810 T1=T1+T(I)
820 S1=S1+C(I)
830 Z1=X6(I)*10^(-4)
840 X6(I)=IP(Z1)*3600
850 Z1=FP(Z1)*100
860 X6(I)=X6(I)+IP(Z1)*60+FP(Z1)*100
870 X6(I)=X6(I)+X5(I)*86400
880 IF I=Q5 THEN 900
890 C1=C1+C(I)/T(I)*(X6(I)-X6(I-1))
900 NEXT I
910 T=X6(N)-X6(Q5)
915 PRINT "IRRADIATION TIME   :";T;"s"
916 PRINT
920 C1=C1/T
2180 PRINT
2190 PRINT "THE CORRECTION IS THE RATIO OF"
2200 PRINT "ACTUAL ACTIVITY AT EOI TO THAT"
2210 PRINT "WHICH WOULD HAVE BEEN PRODUCED"
2220 PRINT "BY THE SAME FLUENCE AT CONSTANT"
2230 PRINT "POWER" @ PRINT
2240 CLEAR
2250 DISP "WHICH LAMBDA DO YOU WANT?"
2260 DISP "    NUCLIDE   t1/2         "
2270 DISP "1.  In        4.486h   4.292E-5"
2280 DISP "2.  Sc-46     83.85d   9.568E-8"
2290 DISP "3.  Ni        70.81d   1.1330E-7"
2300 DISP "4.  Mn-56     2.576h   7.4744E-5"
2310 DISP "5.  Mn-54     312.5d   2.567E-8"
2320 DISP "6.  Au        2.696d   2.9757E-6"
2330 DISP "7.  Na        15.00h   1.2836E-5"
2340 DISP "8.  Sc-47     3.40d    2.3596E-6"
2350 DISP "9.  Sc-48     43.8h    4.3959E-6"
2355 DISP "10. Ba-La     12.789d 6.2373E-7"
2360 INPUT A
2370 DISP "WAIT"
2380 ON A GOTO 2390,2400,2410,2420,2430,2440,2450,2460,2470,2475
2390 L=.00004292 @ L$="In" @ GOTO 2480
2400 L=.00000009568 @ L$="Sc-46" @ GOTO 2480
2410 L=.0000001133 @ L$="Ni" @ GOTO 2480
2420 L=.000074744 @ L$="Mn-46" @ GOTO 2480
```

```
2430 L=.00000002567 @ L$="Mn-54" @ GOTO 2480
2440 L=.0000029757 @ L$="Au" @ GOTO 2480
2450 L=.000012836 @ L$="Na" @ GOTO 2480
2460 L=.0000023596 @ L$="Sc-47" @ GOTO 2480
2470 L=.0000043959 @ L$="Sc-48" @ GOTO 2480
2475 L=.0000006273 @ L$="Ba-La"
2480 S3,D1,S2=0
2490 FOR J=Q5+1 TO N
2520 C=L*(X6(J)-X6(J-1))
2530 S2=S2+C(J)/T(J)*(1-EXP(-C))/L*EXP(-(L*(T-(X6(J)-X6(Q5)))))
2540 S3=S3+1*(1-EXP(-C))/L*EXP(-(L*(T-(X6(J)-X6(Q5)))))
2550 NEXT J
2560 S=L*S2/C1
2570 Q=L*S3
2580 C3=1-EXP(-(L*T))
2590 C2=S/C3
2600 PRINT
2610 PRINT "      --------------------        "
2620 PRINT @ PRINT
2630 PRINT "NUCLIDE ... ";L$
2640 PRINT "LAMBDA = ";L;"  1/SEC"
2650 PRINT @ PRINT @ PRINT
2660 PRINT USING 2670 ; S
2670 IMAGE "SATFAC            = ",D.DDDDe
2680 PRINT USING 2690 ; Q
2690 IMAGE "FLAT CASE SFAC        = "D.DDDDe
2700 PRINT USING 2710 ; C3
2710 IMAGE "XACT FLAT CASE SFAC = ",D.DDDDe
2720 PRINT USING 2730 ; C2
2730 IMAGE "CORRECTION          = ",D.DDDDe
2740 PRINT @ PRINT @ PRINT @ PRINT
2750 CLEAR
2760 GOTO 2240
2770 END
```

## TEST REPORT FOR NEUTRON FLUENCE STANDARDS
## PRODUCED BY AN IRRADIATION AT THE NBS
## $^{235}$U CAVITY FISSION SOURCE

Irradiation No.: 8601                    Report Date: March 6, 1986


Sensor Materials Irradiated: Nickel, Gold, and Scandium
      (designations: Au-1,2; Sc-S-10; Ni-3,4,5,6)


End of Irradiation (EOI): Jan 20, 1986 at 2137:00 EST (Day 20 of 1986)


REPORT FOR:    Ni-5


Free Field Fission Neutron Fluence: $\Phi$ = 8.266 x 10$^{15}$ n/cm$^2$ ± 2 % (1 $\sigma$)

Physical Description of Material:   Mass = 0.28564 g

                Form:   12.7 mm dia. X 0.25 mm thk disks

                Type:   Natural element nickel

Additional Parameters of Irradiation:

      Decay Correction Factor:  C = 0.9881

      Length of Irradiation:  T = 210960 s

      Departure from free-field activity due to Neutron Scattering ($\mu_{sc}$):

            $\mu_{sc}$(Ni)   = -(0.7 ± 0.3) %


Other Radioactivities:

      Since the sensors are natural metal samples or in some cases special
      alloys, 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:

            $^{60}$Co < 0.1 % of $^{58}$Co activity on 1/28/86


This fluence standard prepared by:            For the Director



---

G.P. Lamaze                            R.S. Caswell, Chief
Ionizing Radiation Division            Ionizing Radiation Division

## Relationship of Test Report Data to the Measured Activity at End of Irradiation (EOI) for Cavity Fission Neutron Source Irradiations

The neutron fluence or fluence rate standard has been exposed to a fission neutron fluence that is perturbed by a small cavity return flux and by the effects of neutron scattering in fission source disks, encapsulations, and the fluence standards themselves. This test report certifies a free-field fission neutron fluence or fluence rate, i.e. corrected for neutron scattering. This correction, based on calculation, is reported as the fractional departure from the free-field activation attributable to neutron scattering including cavity return. On this basis the total activity measured at EOI is

$$A_m = A (1 + \mu_{sc}) \tag{1}$$

where: $A_m$ = total activity measured at EOI

$A$ = net free-field activity at EOI

$\mu_{sc}$ = 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 unperturbed activity of the neutron fluence standard in disintegrations per second is related to the free-field fission neutron fluence by the activation equation:

$$\frac{A}{\lambda C N} = \sigma \cdot \Phi \tag{2}$$

where: $A$ = net free-field activity at EOI (dps)

$\lambda$ = 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 $\lambda T C$ -- see below.)

$N$ = number of reaction isotope atoms in the fluence standard

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

$\Phi$ = free-field fission neutron fluence $(n/cm^2)$ given in this test report. (Note: the symbol $\Phi$ 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):

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

$\dfrac{A}{[\lambda TC]N}$ = 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.

$\langle \phi \rangle$ = average, free-field fission neutron fluence rate $(n/cm^2 s)$ given in this test report.

$\sigma \langle \phi \rangle$ = 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

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

where Y is the chain fission yield. When the fission product actvity 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

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

where $\lambda$ is the decay constant of the parent activity and $\lambda'$ that of the daughter.

## Recommended Procedure for Establishing a Calibration Factor ($\chi_{25}$)

Use Equation (1) and either Equation (2) or Equation (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 with the same fission spectrum shape and detector cross section set as are used in the transport calculations which provide spectrum-averaged detector cross-sections for subsequent neutron field measurements.

Revised: November 3, 1986

## ASSIGNMENT OF UNCERTAINTIES (± 1 σ)

- Flus Transfer Measurements from the $^{252}$Cf Standard Fission Neutron Field to the $^{235}$U Cavity Fission Source:

|  | Percent | |
|---|---|---|
| (1) Absolute neutron source strength of the $^{252}$Cf source | 0.9 | S+R* |
| (2) Distance measurement | 0.6 | R |
| (3) $^{252}$Cf to $^{235}$U spectrum-averaged cross section ratio for Pu(n,f) monitor | 0.1 | S |
| (4) Correction for neutron scattering for Pu(n,f) | 0.9 | S |
| (5) Reproducibility of flux transfer procedure (including counting statistics, and electronic system stability over period of measurement) | 0.4 | R |

Subtotal (summed in quadrature) = 1.5

- Preparation of Neutron Fluence Standards at the $^{235}$U Cavity Fission Source:

|  | Percent | |
|---|---|---|
| (1) Gradient corrections and positioning | 0.4 | S+R |
| (2) Corrections for neutron scattering for $^{115}$In(n,n') | 1.2 | R |
| (3) Reproducibility of fluence monitoring including random summing corrections and uncertainties in activation decay corrections | 0.4 | R |

Subtotal (summed in quadrature) = 1.3

TOTAL UNCERTAINTY = 2.0

*S is for systematic uncertainty and R is for random uncertainty.