NUREG/CR-3391, Vol. 4
R5

LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

QUARTERLY PROGRESS REPORT
OCTOBER 1983 - DECEMBER 1983

Hanford Engineering Development Laboratory

Prepared by
E.P. Lippincott
W.N. McElroy

Prepared for the U.S. Nuclear Regulatory Commission

GPO Printed copy price: $4.50

NUREG/CR-3391, Vol. 4
R5

LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

QUARTERLY PROGRESS REPORT
OCTOBER 1983 - DECEMBER 1983

Hanford Engineering Development Laboratory

Operated by Westinghouse Hanford Company
P.O. Box 1970 Richland, WA 99352
A Subsidiary of Westinghouse Electric Corporation

Prepared by
E.P. Lippincott
W.N. McElroy

Manuscript Completed: March 1984
Date Published: May 1984

Prepared for Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555
NRC FIN No. B5988

PREVIOUS REPORTS IN LWR-PV-SDIP SERIES

NUREG/CR-0038	HEDL-TME 78-4	July 1977 - September 1977
NUREG/CR-0127	HEDL-TME 78-5	October 1977 - December 1977
NUREG/CR-0285	HEDL-TME 78-6	January 1978 - March 1978
NUREG/CR-0050	HEDL-TME 78-7	April 1978 - June 1978
NUREG/CR-0551	HEDL-TME 78-8	July 1978 - September 1978
NUREG/CR-0720	HEDL-TME 79-18	October 1978 - December 1978
NUREG/CR-1240, Vol. 1	HEDL-TME 79-41	January 1979 - March 1979
NUREG/CR-1240, Vol. 2	HEDL-TME 80-1	April 1979 - June 1979
NUREG/CR-1240, Vol. 3	HEDL-TME 80-2	July 1979 - September 1979
NUREG/CR-1240, Vol. 4	HEDL-TME 80-3	October 1979 - December 1979
NUREG/CR-1291	HEDL-SA-1949	October 1978 - December 1979*
NUREG/CR-1241, Vol. 1	HEDL-TME 80-4	January 1980 - March 1980
NUREG/CR-1241, Vol. 2	HEDL-TME 80-5	April 1980 - June 1980
NUREG/CR-1747	HEDL-TME 80-73	October 1979 - December 1980*
NUREG/CR-1241, Vol. 3	HEDL-TME 80-6	October 1980 - December 1980
NUREG/CR-2345, Vol. 1	HEDL-TME 81-33	January 1981 - March 1981
NUREG/CR-2345, Vol. 2	HEDL-TME 81-34	April 1981 - June 1981
NUREG/CP-0029	HEDL-SA-2546	October 1980 - September 1981*
NUREG/CR-2345, Vol. 4	HEDL-TME 81-36	October 1981 - December 1981
NUREG/CR-2805, Vol. 1	HEDL-TME 82-18	January 1982 - March 1982
NUREG/CR-2805, Vol. 2	HEDL-TME 82-19	April 1982 - June 1982
NUREG/CR-2805, Vol. 3	HEDL-TME 82-20	October 1981 - September 1982*
NUREG/CR-2805, Vol. 4	HEDL-TME 82-21	October 1982 - December 1982
NUREG/CR-3391, Vol. 1	HEDL-TME 83-21	January 1983 - March 1983
NUREG/CR-3391, Vol. 2	HEDL-TME 83-22	April 1983 - June 1983
NUREG/CR-3391, Vol. 3	HEDL-TME 83-23	October 1982 - September 1983*

*Annual Reports

FOREWORD

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by NRC to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis methods, procedures and data used to predict the integrated effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the US NRC-supported activities at HEDL, ORNL, NBS, and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany), and several UK laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards.

The primary objective of this multilaboratory program is to prepare an updated and improved set of physics-dosimetry-metallurgy, damage correlation, and associated reactor analysis ASTM Standards for LWR pressure vessel and support structure irradiation surveillance programs. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Standard, Reference, and Controlled Environment Benchmark Fields," research reactor "Test Regions," and operating power reactor "Surveillance Positions."

These studies will establish and certify the precision and accuracy of the measurement and predictive methods recommended in the ASTM Standards and used for the assessment and control of the present and end-of-life (EOL) condition of pressure vessel and support structure steels. Consistent and accurate measurement and data analysis techniques and methods, therefore, will be developed, tested, and verified along with guidelines for required neutron field calculations used to correlate changes in material properties with the characteristics of the neutron radiation field. It is expected that the application of the established ASTM Standards will permit the reporting of measured material property changes and neutron exposures to an accuracy and precision within bounds of 10 to 30%, depending on the measured metallurgical variable and neutron environment.

The assessment of the radiation-induced degradation of material properties in a power reactor requires accurate definition of the neutron field from the outer region of the reactor core to the outer boundaries of the pressure vessel. Problems with measuring neutron flux and spectrum are associated with two distinct components of LWR irradiation surveillance procedures: 1) proper application of calculational estimates of the neutron exposure at in- and ex-vessel surveillance positions, various locations in the vessel wall and ex-vessel support structures, and 2) understanding the relationship between material property changes in reactor vessels and their support structures, and in metallurgical test specimens irradiated in test reactors and at accelerated neutron flux positions in operating power reactors.

The first component requires verification and calibration experiments in a variety of neutron irradiation test facilities including LWR-PV mockups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks serve as a permanent reference measurement for neutron flux and fluence detection techniques, which are continually under development and widely applied by laboratories with different levels of capability. The second component requires a serious extrapolation of an observed neutron-induced mechanical property change from research reactor "Test Regions" and operating power reactor "Surveillance Positions" to locations inside the body of the pressure vessel wall and to ex-vessel support structures. The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than at test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Further, the neutron spectrum at, within, and leaving the vessel is substantially different.

In order to meet the reactor pressure vessel radiation monitoring requirements, a variety of neutron flux and fluence detectors are employed, most of which are passive. Each detector must be validated for application to the higher flux and harder neutron spectrum of the research reactor "Test Region" and to the lower flux and degraded neutron spectrum at "Surveillance Positions." Required detectors must respond to neutrons of various energies so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimates. Detectors being used, developed, and tested for the program include radiometric (RM) sensors, helium accumulation fluence monitor (HAFM) sensors, solid state track recorder (SSTR) sensors, and damage monitor (DM) sensors.

The necessity for pressure vessel mockup facilities for physics-dosimetry investigations and for irradiation of metallurgical specimens was recognized early in the formation of the NRC program. Experimental studies associated with high- and low-flux versions of a pressurized water reactor (PWR) PV mockup are in progress in the US, Belgium, and the United Kingdom. The US low-flux version is known as the ORNL Poolside Critical Assembly (PCA), and the high-flux version is known as the Oak Ridge Research Reactor (ORR) Poolside Facility (PSF). Both are located at Oak Ridge, TN. As specialized benchmarks, these facilities will provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV and support structure neutron field calculations, and temperature-controlled metallurgical specimen exposures are brought together. The two key low flux pressure vessel mockups in Europe are known as the Mol-Belgium-VENUS and Winfrith-United Kingdom-NESDIP facilities. The VENUS facility is to be used for PWR core source and azimuthal lead factor studies, while NESDIP is to be used for PWR cavity and azimuthal lead factor studies.

The results of the measurement and calculational strategies outlined here will be made available for use by the nuclear industry as ASTM Standards. Federal Regulation 10CFR50 already requires adherence to several ASTM Standards that establish a surveillance program for each power reactor and incorporate metallurgical specimens, physics-dosimetry flux-fluence monitors and neutron field evaluation. Revised and new standards in preparation will be carefully updated, flexible, and, above all, consistent.

CONTENTS

FIGURES

TABLES

SUMMARY

Following the 1983 Annual Report issue in this series (NUREG/CR-3391, Vol. 3), these reports have been changed to semiannual progress reports. Two semiannual reports will be issued per year covering the periods October-March and April-September (in place of the previous three quarterlies and one annual). Since no regular quarterly report is, therefore, prepared for October - December 1983, this special report is being issued to provide timely dissemination of the Benchmark Compendium prepared by the National Bureau of Standards (NBS). Contributions from other participating laboratories will appear in the first 1984 semiannual (due April 1984).

NATIONAL BUREAU OF STANDARDS (NBS)

In order to achieve reliable surveillance dosimetry for pressure vessels in light water reactors, it is necessary to perform sensor calibration, other dosimetry reference measurements, and validation of transport calculation in benchmark and test region neutron fields. One element in this effort is the development of a compendium of experimental and calculated dosimetry-related information for these fields. The first two entries in this compendium, the ^{252}Cf and ^{235}U fission spectrum standard fields, were completed in 1978. The revision given in this Progress Report provides updated facility information and evaluated dosimetry cross sections with a cut-off date of June 1983. References are in a separate section following the two entries.

NATIONAL BUREAU OF STANDARDS

COMPENDIUM OF BENCHMARK NEUTRON FIELDS FOR PRESSURE VESSEL SURVEILLANCE DOSIMETRY

J. A. Grundl
C. Eisenhauer
E. D. McGarry

Objective

To provide a compendium of information on benchmark neutron fields that are employed for detector calibration, dosimetry methodology referencing, and transport calculation validation for pressure vessel irradiation surveillance.

Summary

In order to achieve reliable surveillance dosimetry for pressure vessels in light-water power reactors it is necessary to perform sensor calibrations, other dosimetry reference measurements and validation of transport calculation in benchmark and test region neutron fields. One element in this effort is the development of a compendium of experimental and calculated dosimetry related information for these fields. The first two entries in this compendium, the ^{252}Cf and ^{235}U fission spectrum standard fields, were completed in 1978. The revision given in this Progress Report provides updated facility information and evaluated dosimetry cross sections with a cut-off date of June, 1983. References are in a separate section following the two entries.

Accomplishments and Status

The revised compendium entries for the ^{252}Cf and ^{235}U fission spectrum benchmarks were completed in November 1983. The next standard benchmark field to be documented will be the Intermediate-Energy Standard Neutron Field (ISNF).

COMPENDIUM OF BENCHMARK NEUTRON FIELDS

FOR

REACTOR DOSIMETRY

STANDARD NEUTRON FIELD ENTRY

Part IA: ^{252}Cf Spontaneous Fission Neutron Spectrum

TABLE OF CONTENTS

1. IDENTIFICATIONS

1.a. NEUTRON FIELD: ^{252}Cf SPONTANEOUS FISSION SPECTRUM

CLASSIFICATION: STANDARD NEUTRON FIELD

1.b. DESIGNATION: XCF-5-N1

NBS 1975 evaluation of documented differential spectrum measurements. References: F1 (1975) and F2 (1975)

1.c. ENTRY DATE: May, 1978 REVISIONS: July, 1983

1.d. GENERIC DESCRIPTION

A standard ^{252}Cf neutron field consists of neutrons from the spontaneous fission of Californium-252 with little or no energy degradation from collided fission or other background neutrons. The median energy of this spectrum is 1.68 MeV, with 98% of the neutrons between 0.1 MeV and 8 MeV. Neutron fluxes in the range of 10^7 n/(cm^{2}s) are obtained in isolated environments near small, intense fission sources. Neutron fluences are established in terms of neutron source strength, irradiation time, and source-detector distance; neither microscopic nuclear data nor an irradiation monitor are required. Certified free-field fluences of up to 10^{13} n/cm^2 may be obtained with uncertainties as low as ± 1.5% (1σ).

Measurements of the Cf fission neutron spectrum and its close relative the ^{235}U fission spectrum are extensive and well-documented. Therefore, these two standard fields are much better known than any other benchmark employed for reactor dosimetry calibration. In the energy range above 2 MeV many neutron fields in and around test and power reactors have fission-spectrum-like components.

1.e. FACILITY LOCATIONS:

National Bureau of Standards (NBS)
Center for Radiation Research
Gaithersburg, Maryland 20878
U. S. A.
Contact: Dale McGarry or James Grundl
Phone: 301-921-2767

Physikalisch-Technische
Bundesanstalt (PTB)
33 Braunschweig
Federal Republic of Germany
Contact: Wolfgang G. Alberts
Phone:

SEFOR Calibration Facility
University of Arkansas
Fayetteville, AR 72701
U.S.A.
Contact: Cecil O. Cogburn
Phone: 501-575-3153

1.f. CONTACT FOR INFORMATION:

James Grundl or Dale McGarry
Center for Radiation Research
National Bureau of Standards
Gaithersburg, Maryland 20878
U. S. A. Phone: 301-921-2767

2. SUMMARY INFORMATION

2.a. AVAILABLE FISSION NEUTRON EXPOSURES (NBS ONLY)

- o flux intensity (5 cm from source) $\sim 2 \times 10^7$ n/(cm^{2}s)
- o nominal maximum fluence 7×10^{12} n/cm^2
- o accuracy of free-field fluence (1σ) $\pm$ 1.5% (1σ)

2.b. CORRECTIONS FOR NEUTRON SCATTERING (5 CM POSITION)

- o Net perturbation for threshold detector reaction rate $<(0.5 \pm 0.5)$%
- o Net perturbation for ^{235}U reaction rate (1.4 ± 0.5)%

3. FACILITY DESCRIPTION

Fission neutron irradiation facilities employing intense ^{252}Cf spontaneous fission sources exist at the National Bureau of Standards, the SEFOR Calibration Center in Arkansas, Westinghouse-Bettis in the U. S. and the PTB standards laboratory in the Federal Republic of Germany. Only the NBS facility which is in general use for neutron detector calibrations and cross section measurements will be summarized in detail. Distinguishing characteristics of the other irradiation facilities may be ascertained from the literature (R1, R2, R3, R4, R5, R6).

3.a. CONFIGURATION AND CHARACTERISTICS (F3)

The two irradiation locations available 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.

At both locations a light-weight source detector assembly is available for irradiation of passive and active neutron detectors. The source-detector assembly is shown in Fig. X-1 with two active fission chambers mounted on a single axis. Alternatively, simultaneous exposure in pairs may be performed along three separate axes.

The californium source capsule shown in Fig. X-2, is made up of a disk-shaped 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 ± 0.5 mm based on constraints of fabrication and x-ray

photographs. Newer capsule designs feature a short attachment stem in place of the first few cm of the source guide tube, the Cf deposit enclosure is unchanged.

Neutron field parameters for a nominal 5 cm source-to-detector distance are given in Table X-1. The source strength uncertainty of ± 1.1% dominates the composite error. The irradiation geometry shown in Fig. X-1 is termed compensated-beam geometry and refers to the experimental practice of placing detector pairs of similar sensitivity on opposite sides of the source, and equidistant from it. The source-to-detector distance error to first-order then is associated only with separation of the detectors; the uncertainty in source position becomes second order. The effective detector response in this case involves the geometric average of the two detector responses. The source and the detectors may be rotated during any irradiation to further ensure proper spatial averaging of the neutron flux at the detectors.

Scattering Corrections. Neutron fluence perturbations in location B attributable to room return, scattering in the source capsule and in support structures are given in Table X-2. Corresponding reaction perturbations are listed for both a threshold, and a low-energy response neutron detector irradiated at a source-to-detector distance of 5 cm. The room-return component is estimated from calculation and response-versus-distance measurements in which the source is moved while the detectors are kept at their normal position. For detectors with low-energy response, a large cadmium basket is placed around the source-detector assembly. The fission neutron return from this basket is less than 0.1%.

Three types of calculations were employed for estimating the free-field fluence: (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 scattering events in individual pieces to obtain corrections for 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 0.5% for source-detector distances up to 15 cm.

The difficulty of correcting a free-field neutron fluence for scattering in a detector depends upon the mass. arrangement, and material of the detector. The NBS has codes available which provide corrections for isotropic scattering of neutrons in lightly constructed detectors with cylindrical symmetry. More extensive Monte-Carlo calculations have been carried out for special detectors, specifically the NBS double fission chamber. Multiple scattering in more massive detectors are difficult to estimate and often require auxiliary experiments. Generally, isolated Cf fission neutron fields are most appropriate for accurate, uncluttered exposures with light weight detectors.

3.b. IRRADIATION PROCEDURES

Detector pairs of similar sensitivity are hung from the mounting ring on opposite sides and equidistant from the source - see Fig. X-1. Alignment and the exact distance between front faces of detectors are determined optically with a coordinate measuring machine. The mounting ring then is placed in the irradiation facility and the Cf source is raised from the storage hole for a timed exposure. Exposure times ranging from a few minutes to a few days are available. As many as three detector pairs may be irradiated simultaneously

along separate axes. No irradiation monitor is required since californium is a natural neutron source with an emission rate governed by its nuclear decay constant (2.2% change per month).

The free-field fission neutron fluence at the detector positions (i.e. the fluence in the absence of all neutron scattering effects) is obtained from the known Cf neutron source strength, the detector separation distance, and the irradiation time. The final certified free-field fission neutron fluence for the detector pair is specified to ± 1.2% (1σ). Reaction rate measurements are expressed in terms of this pair-averaged neutron fluence.

3.c. SPECIFICATION FOR NEUTRON TRANSPORT CALCULATION

Neutron transport calculations are employed for scattering corrections. Material specifications for the source are indicated in Section 3a and in Fig. X-2.

4. NEUTRON FIELD CHARACTERIZATION

The energy spectrum of californium-252 spontaneous fission neutrons is similar to that of U-235 and other fissionable materials. As such it provides a neutron flux characteristic of the driving source for most of nuclear energy. Because fission spectra are similar and were evaluated at the same time, the ^{252}Cf spectrum will be specified together with the ^{235}U spectrum. A concise description of the two fission spectra may be given in terms of broad energy ranges,

Fission Spectrum 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

and a coarse multigroup flux display, $\phi(E)\Delta E$, as follows:

Fission Spectra in Seven-Groups

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

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

4.a. NEUTRON FLUX.

The neutron emission rate for the single-encapsulated sources employed at NBS are $\sim 5 \times 10^9$ n/s when fabricated and decays at a rate of 2.2% per month. Limitations on use of the irradiation facility generally restrict exposure times to less than a week. Thus, for a neutron flux intensity of $\sim 2 \times 10^7 n/(cm^2 s)$ at the 5-cm distance, maximum fluences are in the range of $8 \times 10^{12} n/cm^2$.

Flux gradients for the uncollided neutrons are just those of a point source in free space:

$$\frac{\Delta\phi}{\phi} = \frac{2\Delta R}{R}$$

A small flux anisotropy results from the cylindrical shape of the source. The ratio of the flux averaged over all directions to the flux along a direction perpendicular to the capsule axis is 0.992 when the detectors do not subtend an angle of more than about 20° at the source (see Fig. X-1).

4.b. NEUTRON SPECTRA

4.b.1. Calculation. Fission neutron spectra are most accurately determined by laboratory measurement. (See, however, Ref. F19 and earlier work cited therein.)

4.b.2. Measurement (F2, F1). From 1952 to the present, well-documented fission spectrum measurements of ^{252}Cf and ^{235}U have employed every perfected technique of neutron spectrometry (F4). Fission spectra, therefore, are among the most widely studied of any neutron energy distributions. The large body of available fission spectrum data was evaluated at NBS in 1975 (F2). The evaluation included an estimate of the spectrum uncertainty 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.

4.b.3 Evaluated Spectrum. The NBS fission spectrum evaluation, is chosen for this compendium (F2, F1). The ^{252}Cf spectrum is defined up to 20 MeV by means of a reference Maxwelliam, M(E), modified by four piecewise continuous linear segments below 6 MeV plus one exponential segment above 6 MeV. The reference Maxwellian is

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

and the adjustment functions $\mu(E)$ are 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)/1.0]

The evaluated spectrum then is given by $\chi(E) = \mu(E) \cdot M(E)$. A 45-group tabulation of the evaluated ^{252}Cf fission spectrum is given in Table X-4 along with the corresponding ^{235}U evaluated spectrum. Group fluxes for other energy group structures may be derived from this tabulation with the interpolation recommended in the footnote.

Spectrum uncertainties in the evaluation are based on departures of subsets of measured data from the final adjusted Maxwellian. This estimate, carried out in a seven group structure, includes both 1σ and 2σ errors. Results are given in Table X-5.

5. INTEGRAL DETECTOR RESPONSE

Performing neutron dosimetry measurements with integral detectors requires the use of benchmark neutron fields for calibrating measurement techniques and for referencing data interpretation methods. In order to carry out these measurement assurance procedures, and in particular for estimating uncertainties, it is necessary to distinguish between calculated and measured reaction rates and to identify parameters which characterize the energy response of each detector in the various spectra to which it is exposed. A formulation which meets these requirements is described briefly in this section and includes expressions for first-order error propagation. Tables of response parameters list present cross sections, spectral indexes, and energy response ranges for a number of reaction rate detectors. Measured reaction rates are presented in Section 5.b. followed by comparisons of measured and calculated cross sections in Section 5.c.

5.a. CALCULATED REACTION PROBABILITIES

The calculated reaction probability, R_c, for an integral detector is equal to the product of the neutron fluence to which it has been exposed and to the spectrum-averaged reaction cross section:

$$R_c = \bar{\sigma} \cdot \Phi \qquad \text{(X-1)}$$

Φ = total spectrum integrated neutron fluence, (n/cm^2).

$\bar{\sigma}$ = spectrum-averaged reaction cross section: $\int_0^\infty \sigma(E)\psi(E)dE$,

where $\psi(E)$ is the neutron spectrum normalized to unity. In many applications the lower limit of integration is 0.4 eV, the cadmium cut-off.

The reaction probability is sometimes referred to as the total reactions per target nucleus or the time-integrated reaction rate. In this formulation all experimental quantities such as detection efficiencies, time history corrections, and neutron field perturbations are relegated to the expression for the measured reaction probability given in Eq. X-10.

Integral detector response also may be expressed in terms of an average neutron fluence rate, $\langle\phi\rangle = \Phi/T$, where T is the irradiation interval. In this case, the quantity of interest (experimentally, the saturated specific activity for activation detectors) is the average reaction rate: $\bar{\sigma}\langle\phi\rangle$.

When neutron spectra extend over a large energy range, a truncated neutron fluence, $[\psi(>E_o)\cdot\Phi]$, may be defined where $\psi(>E_o)$ is the spectrum fraction above E_o. An example from materials damage dosimetry is $[\psi(>1\ \text{MeV})\cdot\Phi]$, the fluence greater than 1 MeV. Similarly, for integral detectors with a restricted energy response range (e.g. threshold detectors), a cross section truncated near the lower bound of the detector response range is a measure of detector response invariance to spectrum shape:

$$\bar{\sigma}(>E_p) = \frac{\int_{E_p}^{\infty} \sigma(E)\psi(E)dE}{\int_{E_p}^{\infty} \psi(E)dE}\ ,\ E_p = \text{truncation energy.} \qquad (X\text{-}2)$$

For a fractional response p, the truncation energy is defined by

$$p \cdot \bar{\sigma} = \int_{E_p}^{\infty} \sigma(E)\psi(E)dE$$

$$= \bar{\sigma}(>E_p) \cdot \psi(>E_p) \qquad (X\text{-}3)$$

where $\psi(>E_p)$ = the spectrum fraction above E_p

p = fraction of detector response above E_p;

$E_p(p=1) = 0.4$ eV; $E_p(p=0) = 20$MeV; $E_p(p=0.5)$ = median energy.

The advantage of a truncated cross section $\bar{\sigma}(>E_p)$ is its independence of spectrum in the energy region where the detector does not respond.

In terms of these energy-truncated quantities the calculated reaction rate of Eq. X-1 becomes

$$R_c = \frac{\psi(>E_p)}{p \cdot \psi(>E_o)} \cdot \bar{\sigma}(>E_p) \cdot \left[\psi(>E_o) \cdot \Phi\right]. \qquad (X\text{-}4)$$

An alternative formulation of truncated fluence which deals with $\bar{\sigma}(>E_o)$ in place of $\bar{\sigma}(>E_p)$ has been in some use. The relationship between the two formulations follows directly from Eq. X-4,

$$\bar{\sigma}(>E_o) \equiv \frac{[\text{reaction probability}]}{[\text{fluence above } E_o]} = \frac{\psi(>E_p)}{p \cdot \psi(>E_o)} \cdot \bar{\sigma}(>E_p)$$

5.a.1. Spectrum Response Table. Basic integral detector response parameters for ^{252}Cf fission neutrons are given in Table X-6 (B5). The spectrum averaged cross sections listed in column 2 are the full-spectrum averaged values above a cadmium cut-off energy of 0.4 eV and are followed in column 3 by cross sections truncated at p = 0.95. The energy dependent cross sections employed are those of the ENDF/BV dosimetry file reduced to 620 energy groups; spectrum averaging is carried out with the NBS DETAN code. The spectrum fraction, $[\psi(>E_p), p = 0.95]$, follows in column 4. Energy response characteristics given in the last three columns, are the median response energy and the enery response range, i.e., the lower- and upper-energy bounds

that include 90% of the detector response. The lower-energy bound, $E_p(p = 0.95)$, is the truncation energy corresponding to the truncated cross section given in column 3.

5.a.2. Spectral Indexes. Elementary spectrum indicators associated with integral detector responses are the spectrum-averaged cross section ratios among detector pairs with distinguishable energy response ranges. This ratio is generally called a spectral index:

$$S_{\alpha/\beta} = \bar{\sigma}_\alpha / \bar{\sigma}_\beta \quad ,$$

where α and β refer to the two detector reactions involved.

A selected set of spectral indexes calculated for full-spectrum-averaged cross sections is given in Table X-7 (B5). The corresponding set for truncated cross sections is in Table X-8 (B5). The ratio of a calculated spectral index to the observed value, $C_{\alpha/\beta}$, is the basic spectrum information available from an integral detector measurement.

$$C_{\alpha/\beta} = \left[S_{\alpha/\beta}\right]_{\text{calc.}} \Big/ \left[S_{\alpha/\beta}\right]_{\text{obs.}} \qquad \text{(X-5)}$$

These double ratios are established for benchmark neutron fields and for neutron fields under study. Then, by a variety of analytic means, the double ratios may be employed for (1) directly adjusting the calculated spectrum of the neutron field under study; (2) normalizing neutron transport calculations; (3) establishing spectrum consistency among the benchmark fields themselves; or (4) validating or adjusting reaction rate cross sections. In particular, calibration against a benchmark neutron field is established by normalizing the double ratio for a study spectrum to the corresponding double ratio for a relevant benchmark:

$$\left[T_{\alpha/\beta}\right]_C = \left[C_{\alpha/\beta}\right]_{study} \Big/ \left[C_{\alpha/\beta}\right]_{bchmk} \quad . \qquad (X\text{-}6)$$

5.a.3 Error Propagation. Apart from the complexities of assessing correlations, expressions to estimate the uncertainty of calculated spectrum response parameters are obtained directly from first-order statistical propagation. Three basic error propagation expressions for uncorrelated errors in spectrum and cross sections errors are included here, others may be found in Reference R-7 along with an example of their application to reactor pressure vessel irradiation surveillance dosimetry.

A full-spectrum-averaged cross section, $\bar{\sigma}$, in multigroup format is

$$\bar{\sigma} = \sigma_o \cdot \sum_i s_i \psi_i \Delta E_i \Big/ \sum_i \psi_i \Delta E_i \quad ,$$

where $\sigma_i = \sigma_o s_i$ is the energy dependent reaction cross section and ψ_i the neutron spectrum. The standard error for $\bar{\sigma}$, without covariance, is

$$(\delta\sigma)^2 = \sum_i \left[\left(\frac{\partial\bar{\sigma}}{\partial s_i}\right)^2 (\delta s_i)^2 + \left(\frac{\partial\bar{\sigma}}{\partial\psi_i}\right)^2 (\delta\psi_i)^2 + \left(\frac{\partial\bar{\sigma}}{\partial\sigma_o}\right)^2 (\delta\sigma_o)^2\right]$$

The fractional standard error, $\delta\bar{\sigma}/\bar{\sigma}$, follows directly

$$\left(\frac{\delta\bar{\sigma}}{\bar{\sigma}}\right)^2 = \sum_i \left[\frac{\sigma_i}{\bar{\sigma}} \psi_i \Delta E_i\right]^2 \left(\frac{\delta s_i}{s_i}\right)^2 + \sum_i \left[1 - \frac{\sigma_i}{\bar{\sigma}}\right]^2 (\psi_i \Delta E_i)^2 \left(\frac{\delta\psi_i}{\psi_i}\right)^2 + \left(\frac{\delta\sigma_o}{\sigma_o}\right)^2 \qquad (X\text{-}7)$$

The quantities $\delta\psi_i/\psi_i$ and $\delta s_i/s_i$ are the fractional errors in the group fluxes and the group cross section shape, respectively.

The fractional error for a spectral index, $S_{\alpha/\beta} = \bar{\sigma}_\alpha/\bar{\sigma}_\beta$, is similar,

$$\left(\frac{\delta S}{S}\right)^2 = \sum_i \left[\left(\frac{\sigma_{\alpha i}}{\sigma_\alpha}\right)^2 \left(\frac{\delta s_{\alpha i}}{s_{\alpha i}}\right)^2 + \left(\frac{\sigma_{\beta i}}{\sigma_\beta}\right)^2 \left(\frac{\delta s_{\beta i}}{s_{\beta i}}\right)^2\right] (\psi_i \, \Delta E_i)^2$$
$$+ \sum_i \left(\frac{\sigma_{\alpha i}}{\sigma_\alpha} - \frac{\sigma_{\beta i}}{\sigma_\beta}\right)^2 (\psi_i \, \Delta E_i)^2 \left(\frac{\delta \psi_i}{\bar{\psi}_i}\right)^2 + \left(\frac{\delta \sigma_{o\alpha}}{\sigma_{o\alpha}}\right)^2 + \left(\frac{\delta \sigma_{o\beta}}{\sigma_{o\beta}}\right)^2 \quad . \quad (X\text{-}8)$$

The truncated spectrum-averaged cross section in multigroup form is,

$$\bar{\sigma}\,(>E_p) = \sigma_o \sum_{i>i_p} s_i \psi_i \Delta E_i \quad \sum_{i>i_p} \psi_i \Delta E_i,$$

and the fractional error is

$$\left[\frac{\delta \bar{\sigma}(>E_p)}{\bar{\sigma}(>E_p)}\right]^2 = \sum_{i>i_p} \left[\mu_i \frac{\sigma_i}{\bar{\sigma}(>E_p)} \cdot \frac{\delta s_i}{s_i}\right]^2$$
$$+ \sum_{i>i_p} \left[1 - \frac{\sigma_i}{\bar{\sigma}(>E_p)}\right]^2 \mu_i^2 \left(\frac{\delta \mu_i}{\mu_i}\right)^2 + \left(\frac{\delta \sigma_o}{\sigma_o}\right)^2 \quad . \quad (X\text{-}9)$$

where $\mu_i = \psi_i \Delta E_i / \psi(>E_p)$.

5.b. MEASURED REACTION PROBABILITIES

The measured reaction probability, R, is proportional to some detector response mechanism, D:

$$R = \frac{1}{NG} \cdot [\varepsilon \cdot \mu(\bar{N}, Y, F, \ldots)] \cdot D \quad (X\text{-}10)$$

D = observed integral detector response during or after irradiation (e.g. counts per sec from a fission chamber or from a gamma counting system, or total number of permanently registered nuclear particle tracks).

$G(\lambda,t)$ = activation decay rate factor, (sec^{-1})

$$= \lambda \frac{\int_{-T}^{o} \phi(t)\ e^{\lambda t} dt}{\int_{-T}^{o} \phi(t) dt} \quad ,$$

where $\phi(t)$ is the irradiation time history of the neutron flux intensity, T is the irradiation interval, and λ is the decay constant. For an uninterrupted irradiation of length T at constant flux, $G = [1 - \exp(-\lambda T)]/T$. The dimensionless quantity G/λ is sometimes used in place of G. For detectors that register reaction rate directly (e.g., fission chambers), G is replaced by 1/T.

N = number of detector atoms.

ε = detection efficiency factor

μ = composite of response factors exclusive of ε/NG needed to connect observed counts per sec or its equivalent to reaction probability; e.g., effective number of detector atoms including neutron self absorption corrections ($\bar{N}$), fission yield (Y), neutron field perturbations (F), isotopic abundance, branching ratio, extrapolation of pulse height distribution, effects of competing reactions, background subtraction, pulse losses, etc.

Benchmark referencing tests the combined validity of many of the factors included within the bracket of Eq. X-10. In dosimetry experiments where absolute reaction rates are not determined, or are separately interpreted, a comprehensive calibration factor may be established on the basis of a standard neutron field exposure. In this way, most of the response factors in Eq. X-10 are either eliminated or their error contribution reduced. Subsequent fluence determinations with the detection system thus calibrated will be as accurate

as the calibration factor if the neutron spectrum matches that of the standard neutron field. Spectrum differences between the standard and the field under study will introduce additional errors, mainly through differences in spectrum-averaged cross sections in Eq. X-1. Error propagation formulas similar to Eq. X-7 and X-9 may be used to obtain an estimate of the latter error contribution.

The important observed quantities for reactor dosimetry are the ratios of reaction probabilities among detectors which, according to Eq. X-1, are proportional to ratios of spectrum-average cross sections. These cross section ratios, when they are sensitive to spectrum shape, are called spectral indexes as noted above. An observed spectral index is obtained from the measured reaction probabilities of two detectors by replacing the calculated reaction probabilities in Eq. X-1 with the measured values in Eq. X-10 and dividing:

$$[S_{\alpha/\beta}]_{obs} = (R_\alpha/R_\beta)_{obs} \quad , \qquad \text{(X-11)}$$

The subscripts α and β refer to the two detector reactions involved.

5.c. MEASURED CROSS SECTIONS AND CALCULATED-TO-OBSERVED RATIOS

For the two fission spectrum benchmarks in this compendium, the primary integral detector quantities are cross sections and spectral indexes taken with respect to the $^{238}U(n,f)$ reaction. This index will be denoted $S_{\alpha/U8}$ or $S_{U8/\alpha}$ depending upon whether the energy response range of a particular detector is higher (U8/α) or lower (α/U8) than that of $^{238}U(n,f)$.

Since absolute free-field neutron fluxes may be established directly for a natural neutron source such as californium, absolute integral cross sections may be determined without recourse to microscopic nuclear data. A substantial amount of such data exists in spite of relatively low fluxes. Data for the californium fission spectrum are reported and summarized in references R2, R3, R4, and R8 through R22. Collections of evaluated measured cross sections may be found in references R16, R19, R23, and R24.

A new and restricted evaluation of measurement reported before July 1983 was performed at NBS for detector reactions considered important for reactor technology. Results are given in Table X-9 and X-11. The spectral indexes in Table X-11 are based either on independent measurements for each detector or on true ratio measurements - the latter most notably for the fission reactions. The uncertainties given at one standard deviation include estimates of systematic errors.

Calculated-to-observed ratios of reaction cross sections are shown in Table X-10(B5). The errors in calculated cross sections correspond to ^{252}Cf fission spectrum uncertainties from Table X-5 propagated according to the second term of Eq. X-7 in Section 5.a. Two reactions, $^{47}Ti(n,p)$ and Th232(n,f), show calculated-to-observed discrepancies in excess of 10% and six of the eighteen reactions, some of considerable significance for reactor dosimetry and neutron flux monitoring, show discrepancies of 5% or more. Standard deviations of the

seventeen C/E ratios in Table X-10(B5), excluding ^{47}Ti(n,p) and Th232(n,f), are ± 0.025 about an average ratio of 1.034 and ± 0.043 about a mean C/E of 0.995.

Calculated-to-observed ratios of spectral indexes, $C_{\alpha/U8}$, are presented in Table X-12(B5). The calculated indexes are obtained with spectrum-averaged cross sections given in column 2 of Table X-6(B5). Non-overlapping energy response intervals are given in column 2 of Table X-12(B5) and contributing errors for observed and calculated indexes are listed separately in columns 4 and 5. As is appropriate for cross section validation, propagated errors for the calculated indexes assess only the contribution of fission spectrum uncertainties given in Table X-5. The propagation formula for this partial error is that of the second term of Eq. X-8, Section 5.a.3.

The $C_{\alpha/U8}$ ratios in Table X-12(B5) indicate how calibration in a reference neutron field can improve the accuracy of a neutron fluence measurement with integral detectors. The $C_{\alpha/U8}$ value for ^{47}Ti(n,p), for example, departs from unity by 29%. Similarly, the $C_{U8/\beta}$ value for U235 fission departs by more than 5%, a significant disagreement considering the amount of cross section data which exists, and the accuracy which is expected of these two reaction. Without benchmark calibration, a neutron fluence obtained with these detectors would be biased by the full amount of the departures shown in Table X-12(B5). The standard deviations of sixteen $C_{\alpha/U8}$ ratios in Table X-12(B5) (^{47}Ti excluded) are ± 0.031 about an average ratio of 1.041, and ± 0.055 about a mean C/E of 1.008.

TABLE X-1. ^{252}Cf FISSION NEUTRON FIELD PARAMETERS AND ERROR COMPONENTS

Free-field fission neutron flux (5-cm distance; source 6×10^9 n/s)	2×10^7 n/(cm^2 s)
Source decay rate	2.2%/month
Free-field fluence for 100-h exposure	7×10^{12} n/cm^2
Source capsule scattering (inelastic plus net elastic inscatter)	1.1%
Gamma-ray exposure at 5 cm (2.8 yr after separation)	~ 190 R/h

Error components for free-field fission neutron flux (1σ):	
Source strength	± 0.9%
Source capsule and support scattering	± 0.7%(max)
Distance measurements (typical for compensated-beam geometry)	± 0.5%
Free-field flux error (rms sum):	± 1.2%

TABLE X-2. TYPICAL NEUTRON FLUENCE AND REACTION RATE PERTURBATIONS IN LOCATION B DUE TO NEUTRON SCATTERING

Distance to nearest boundary:	2.8m
neutron fluence perturbations: (above 0.4 eV)	
room return	<0.0%
source capsule scattering	1.1%
support structure scattering	0.5%
air scatter	<0.1%
Net reaction rate perturbation due to neutron scattering:	
$^{235}U(n,f)$ detector with cadmium cover:	
room return	0.1%
source capsule	0.8%
support structures	0.5%
$^{238}U(n,f)$ threshold detector:	
room return *	0.0%
source capsule	0.0%
support structures	0.5%

*Source capsule perturbations depend markedly upon detector threshold.

TABLE X-3. DIFFERENTIAL SPECTRUM MEASUREMENTS FOR NBS EVALUATION OF ^{252}Cf AND ^{235}U FISSION NEUTRON SPECTRA

C A L I F O R N I U M - 2 5 2 S P O N T A N E O U S F I S S I O N

Reference	Date	Measurement Range (MeV)	Detection Scheme
Green et al.	F5(1973)	0.5 -13	Time-of-flight (TOF); energy scale based on carbon scattering resonances. Scintillator NE-213 efficiency from long counter response (E < 4 MeV); calculations for E > 4 MeV.
Knitter et al.	F5(1973)		TOF; energy scale calibrated with monoenergetic neutrons (0.4 to 2.3 MeV). Liq. scint. eff. from angular distribution of T(p,n), D(d,n), Li(p,n), H(n,n), etc. (0.05 to 8.2 MeV).
Werle and Bluhm (n,p) (^{3}He)	F6(1972)	 1.5 - 7 1 - 4	Proton-recoil proportional counters (PR); response functions from monoenergetic response. ^{3}He-spectrometer and prop. Gas counter.
Meyer et al.	F5(1974)	0.05 - 8	PR (0.05 to 1.2 MeV). Scint. NE-213; unfolded spectrum (1.1 to 8 MeV).
Jeki et al.	F5(1971)	0.002- 1	TOF; ^{6}Li glass scint. efficiency same as Meadows.
Meadows	F6(1967)	0.003-15	TOF; ^{6}Li glass scint. (E < 2.6 MeV); liq. scint. (E > 1.0 MeV); calculated efficiencies.
Condé and During	F6(1965)	0.07 - 7.5	TOF; ^{6}Li Glass scint. (E < 1.1 MeV), Calc. eff.; NE 102A plastic scint. (E > .3 MeV), eff. measured against calibrated long counter.
Zamyatnin et al.	F6(1970)	0.04 - 6	TOF; ^{6}Li glass and plastic scint.

TABLE X-3. DIFFERENTIAL SPECTRUM MEASUREMENTS FOR NBS EVALUATION OF ^{252}Cf AND ^{235}U FISSION NEUTRON SPECTRA (Continued)

U - 2 3 5 T H E R M A L - N E U T R O N - I N D U C E D F I S S I O N

Reference	Date	Measurement Range (MeV)	Detection Scheme
Rosen	F7(1956)	0.3 -13	Incident neutron energy (E_{in}) = thermal. Photographic plate method, background measurements included.
Islam and Knitter	F8(1973)	0.55 - 7	E_{in} = 0.4 MeV, TOF; energy scale checked with monoenergetic neutrons. NE 102 scint. eff. from H(n,n) up to 5.75 MeV.
Barnard et al.	F7(1965)	0.1 - 4	E_{in} = 0.1 MeV, TOF; Plastic scint. calibrated with long ctr. (E < 2.3 MeV).
Condé and During	F6(1965)	1 - 7.5	E_{in} = 0.04 MeV, TOF; Plastic scint. only, same as for ^{252}Cf by Condé and During.
Werle and Bluhm (n,p) (^{3}He)	F6(1972)	 1.5 - 7 1 - 4	Same as for ^{252}Cf by Werle and Bluhm. E_{in} = thermal.
Cranberg et al.	F7(1956)	0.18 - 2.7	E_{in} = 0.08 MeV, TOF; Plastic scint. calibrated against long ctr. (0.1 - 3.4 MeV).
Johansson et al.	F8(1975)	0.5 -14	E_{in} = 0.1 - 2.0 MeV, TOF; energy scale based on scattering resonances (0.5 - 21 MeV). NE 213 scint. eff. from H(n,n) and T(p,n) angular distribution.
Watt	F7(1952)	2.6 -16	E_{in} = thermal. Proton-recoil, gas counter telescope.

TABLE X-4. EVALUATED FISSION NEUTRON SPECTRA FOR ^{252}Cf AND ^{235}U IN 45-GROUP FORMAT

Lower Energy Boundary (MeV)	Designation: XCF-5-N1 Californium-252 Group Flux $\phi \cdot \Delta E$	Californium-252 Cumulative Flux to Lower Boundary[a]	Designation: XU5-5-N1 Uranium-235 Group Flux $\phi \cdot \Delta E$	Uranium-235 [a]Cumulative Flux to lower Boundary
0.0	0.0039	1.0000	0.0048	1.0000
0.05	0.0074	0.9961	0.0088	0.9952
0.111	0.0219	0.9887	0.0249	0.9864
0.183	0.0140	0.9668	0.0154	0.9615
0.25	0.0152	0.9528	0.0166	0.9461
0.302	0.0323	0.9377	0.0350	0.9295
0.4	0.0337	0.9054	0.0363	0.8945
0.498	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
1.35	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
3.679	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
6.065	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	0.00021	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 $E^{1/2} \cdot \exp(-aE)$;

$a = 0.70$ for ^{252}Cf; $a = 0.76$ for ^{235}U.

[a]Spectrum fraction above lower energy bound.

TABLE X-5. ERROR ESTIMATES FOR ^{252}CF and ^{235}U EVALUATED FISSION SPECTRA

Energy Boundaries	Californium-252 (Spontaneous Fission) Group Flux $\phi \cdot \Delta E$	Error 1σ (%)	Error 2σ (%)	Uranium-235 (Thermal-Neutron-Induced Fission) Group Flux $\phi \cdot \Delta E$	Error 1σ (%)	Error 2σ (%)
0.0						
	0.047	±13	±26	0.054	±16	±32
0.25						
	0.184	± 1.1	± 3.3	0.197	± 4.1	± 6.2
0.8						
	0.220	± 1.8	± 3.6	0.229	± 3.0	± 4.8
1.5						
	0.194	± 1.0	± 3.1	0.195	± 3.1	± 5.2
2.3						
	0.200	± 2.0	± 3.0	0.192	± 2.0	± 3.0
3.7						
	0.146	± 2.1	± 4.8	0.127	± 4.8	± 8.0
8						
	0.0087	± 8.5	±17	0.0056	± 5.3	±11
12						
	(0.00058)			(0.00026)		
20						

TABLE X-6(B5). INTEGRAL DETECTOR RESPONSE PARAMETERS

Spectrum: NBS-Evaluated ^{252}Cf Spontaneous Fission Neutron — Designation: xCF-5-N1

Entry Date: May, 1978

Cross Sections: ENDF/B-V Dosimetry File (Ref. R25) — Revised: March, 1982

Detector Reaction	Cross Section[a] $\sigma(>E_p)$		Spectrum[b] Fraction	Median[c] Response Energy	Response Range[c]	
	E_p=0.4 eV	E_p(p=0.95)	$\varphi(>E_p)$	E_p(p=0.5)	E_p(p=0.95)	E_p(p=0.05)
	(barns)	(barns)		(MeV)	(MeV)	(MeV)
Spectrum Check						
Constant	1.000	1.000	0.950	1.68	0.260	5.5
recip. vel., (1/v)	0.1107	0.1062	0.991	0.995	0.089	4.3
Fissionable Mat'ls.						
$^{239}Pu(n,f)$	1.792	1.807	0.942	1.75	0.286	5.7
$^{235}U(n,f)$	1.236	1.223	0.960	1.68	0.224	5.8
$^{233}U(n,f)$	1.904	1.888	0.958	1.62	0.232	5.5
$^{238}U(n,f)$	0.3136	0.541	0.550	2.79	1.50	7.2
$^{237}Np(n,f)$	1.352	1.591	0.807	2.06	0.69	6.2
$^{232}Th(n,f)$	7.81 E-2	0.1360	0.545	3.01	1.51	7.7
$^{240}Pu(n,f)$	1.356	1.598	0.806	2.06	0.69	6.1
$^{241}Pu(n,f)$	1.595	1.575	0.962	1.63	0.22	5.6
$^{238}U(n,\gamma)$	6.83 E-2	6.55 E-2	0.991	0.90	0.088	2.8
$^{232}Th(n,\gamma)$	8.97 E-2	8.63 E-2	0.987	0.92	0.109	2.9
Capture Reactions						
^{23}Na	0.2712 E-3	0.259 E-3	0.996	0.95	0.055	4.7
^{58}Fe	1.660 E-3	1.587 E-3	0.994	1.07	0.067	4.8
^{59}Co	6.03 E-3	5.76 E-3	0.994	1.08	0.068	4.0
^{63}Cu	9.65 E-3	9.22 E-3	0.994	0.98	0.067	4.0
^{115}In	0.1212	0.1169	0.986	1.12	0.117	3.0
^{197}Au	0.0766	0.0731	0.994	0.74	0.068	3.0

TABLE X-6(B5). INTEGRAL DETECTOR RESPONSE PARAMETERS (continued)

Spectrum: NBS-Evaluated ^{252}Cf Spontaneous Fission Neutron

Designation: XCF-5-N1

Entry Date: May, 1978

Cross Sections: ENDF/B-V Dosimetry File (Ref. R25)

Revised: March, 1982

Detector Reaction	Cross Section[a] $\sigma(> E_p)$		Spectrum[b] Fraction	Median[c] Response Energy	Response Range[c]	
	E_p=0.4 eV	E_p(p=0.95)	$\psi(> E_p)$	E_p(p=0.5)	E_p(p=0.95)	E_p(p=0.05)
	(barns)	(barns)		(MeV)	(MeV)	(MeV)
Helium Production						
$^{10}B(n,\alpha)$	0.489	0.468	0.992	1.44	0.078	5.6
$^{6}Li(n,\alpha)$	0.465	0.454	0.972	1.55	0.179	6.0
Threshold Reactions						
$^{115}In(n,n')$	0.1819	0.270	0.641	2.71	1.20	6.2
$^{47}Ti(n,p)$	0.2407 E-1	0.520 E-1	0.440	3.9	1.92	8.0
$^{32}S(n,p)$	0.0760	0.206	0.350	4.1	2.3	7.9
$^{58}Ni(n,p)$	0.1138	0.285	0.380	4.2	2.2	7.9
$^{54}Fe(n,p)$	8.83 E-2	0.249	0.337	4.3	2.4	8.0
$^{46}Ti(n,p)$	1.347 E-2	8.85 E-2	0.1445	5.9	3.8	9.9
$^{27}Al(n,p)$	5.14 E-3	2.79 E-2	0.1749	6.0	3.5	9.8
$^{56}Fe(n,p)$	1.414 E-3	2.79 E-2	0.0482	7.6	5.6	12
$^{63}Cu(n,\alpha)$	0.758 E-3	9.72 E-3	0.0741	7.7	4.9	12
$^{27}Al(n,\alpha)$	1.059 E-3	4.08 E-2	0.0247	8.6	6.6	12
$^{48}Ti(n,p)$	0.409 E-3	1.099 E-2	0.0354	8.4	6.0	13
Additions[e]						
$^{103}Rh(n,n')$	0.703	0.849	0.787	2.37	0.75	6.0
$^{93}Nb(n,n')$	0.1605	0.220	0.694	2.61	1.02	5.9
$^{60}Ni(n,p)$	3.44 E-3	4.57 E-2	0.0715	7.3	4.9	11
$^{55}Mn(n,2n)$	0.440 E-3	0.401	1.044 E-3	13	11	16
dpa	895	1030	0.825	2.65	0.64	6.8

Footnotes for TABLE X-6(B5)

(a) The value given in column 2 is the full-spectrum averaged cross section above a cadmium cut-off of 0.4 eV. The truncated cross section on column 3 is for a truncation energy (column 6) above which 95% of the detector response occurs. A spectrum average cross section truncated at energy E_p is given by

$$\sigma(>E_p) = \int_{E_p}^{\infty} \sigma(E)\psi(E)dE \int_{E_p}^{\infty} \psi(E)dE$$

(b) The fraction of the spectrum above E_p(p = 0.95): $\psi(>E_p) = \int_{E_p}^{\infty} \psi(E)dE$. The full-spectrum-averaged cross section $\sigma(>0.4\ eV)$ for p = 0.95 is equal to $\sigma(>E_p) \cdot \psi(>E_p)/0.95$.

(c) The fractions p = 0.95, 0.5, and 0.05 define energies above which 95%, 50% (median), and 5% of the detector response occurs, respectively. E_p is defined by the relation

$$\int_{E_p}^{\infty} \sigma(E)\psi(E)dE = p \cdot [\sigma(>0.4\ eV)]$$

where $E_p(p = 1) = 0.4$ eV, and $E(p = 0) = 20$ MeV; and $\int_{0.4\ eV}^{\infty} \psi(E)dE = 1$.

(d) Normalization is $\int_{0.4\ eV}^{18\ MeV} \sigma_{1/v}(E)\psi(E)dE = 1$.

(e) Cross sections not on the ENDF/BV Dosimetry File:

- Nb(n,n') is from the IRDF Dosimetry File (R25a).
- Rh(n,n') is from data reported in Ref. (R26).
- dpa, the atom displacement cross section, is from ASTM Standard Practice E693-79. (R27)

TABLE X-7(B5). CALCULATED SPECTRAL INDEXES

Spectrum: ^{252}Cf Spontaneous Fission Neutron

Designation: XCF-5-N1
Entry Date: Feb. 1980
Revised: March, 1982

Spectral Index: Full-Spectrum Cross Sections from Table X-6(B5) (ENDF/B-V Dosimetry File)

$$\left[S_{\alpha/\beta}\right]_{calc.} \equiv \left[\bar{\sigma}_\alpha/\bar{\sigma}_\beta\right]_{calc.}$$

α/β	$^{235}U(n,f)$ ($\bar{\sigma}$=1.236b)	$^{238}U(n,f)$ ($\bar{\sigma}$=0.314b)	α/β	$^{235}U(n,f)$ ($\bar{\sigma}$=1.236b)	$^{238}U(n,f)$ ($\bar{\sigma}$=0.314b)
Fission			Threshold		
^{239}Pu	1.450	5.71	$^{115}In(n,n')$	0.1472	0.580
^{235}U	1.000	3.941	$^{47}Ti(n,p)$	0.01947	0.0768
^{233}U	1.540	6.07	$^{58}Ni(n,p)$	0.0921	0.363
^{238}U	0.2537	1.000	$^{54}Fe(n,p)$	0.0714	0.2816
^{237}Np	1.094	4.31	$^{46}Ti(n,p)$	0.01089	0.0430
^{240}Pu	1.097	4.32	$^{63}Cu(n,\alpha)$	6.13 E-4	2.417 E-3
He Prod.					
			$^{56}Fe(n,p)$	1.144 E-3	4.51 E-3
$^{10}B(n,\alpha)$	0.395	1.559	$^{48}Ti(n,p)$	3.31 E-4	1.304 E-3
$^{6}Li(n,\alpha)$	0.376	1.483	$^{27}Al(n,\alpha)$	8.57 E-4	3.38 E-3
Capture			Additions		
^{238}U	0.0553	0.2178	$^{103}Rh(n,n')$	0.569	2.242
^{232}Th	0.0725	0.2860	$^{93}Nb(n,n')$	0.1299	0.511
^{197}Au	0.0619	0.2443			
^{59}Co	4.88 E-3	0.01923			
^{58}Fe	1.343 E-3	5.29 E-3			

TABLE X-8(B5). CALCULATED SPECTRAL INDEXES -- TRUNCATED

Spectrum: ^{252}Cf Spontaneous Fission Neutron

Designation: XCF-5-N1
Entry Date: Feb. 1980
Revised: March, 1982

Spectral Index: *Truncated Cross Section from TABLE X-6 (ENDF/B-V Dosimetry File)

$$\left[S_{\alpha/\beta}\right]_{calc.} \equiv \left[\bar{\sigma}_{\alpha}(>E_p) \Big/ \bar{\sigma}_{\beta}(>E_p)\right]_{calc.}$$

α/β	$^{238}U(n,f)$ $\bar{\sigma}(>E_p)$=0.541b	α/β	$^{238}U(n,f)$ $\bar{\sigma}(>E_p)$=0.541b
Fission		Threshold	
^{239}Pu	3.34	$^{115}In(n,n')$	0.500
^{235}U	2.261	$^{47}Ti(n,p)$	0.0961
^{238}U	1.000	$^{32}S(n,p)$	0.381
^{237}Np	2.941	$^{58}Ni(n,p)$	0.527
^{240}Pu	2.954	$^{54}Fe(n,p)$	0.460
		$^{46}Ti(n,p)$	0.1636
Additions		$^{56}Fe(n,p)$	0.0516
$^{103}Rh(n,n')$	1.569	$^{63}Cu(n,\alpha)$	0.01797
$^{93}Nb(n,n')$	0.407	$^{48}Ti(n,p)$	0.0203
		$^{27}Al(n,\alpha)$	0.0754

*p = 0.95; E_p for each isotope is given in column 5 of TABLE X-6.

TABLE X-9. OBSERVED INTEGRAL CROSS SECTIONS

Spectrum: ^{252}Cf Spontaneous Fission Neutrons

Designation: XCF
Entry Date: May 1978
Revised: June, 1983

Reaction	[a]Cross Section Value ($\times 10^{-27}$ cm^2)		Median Response Energy, E_{50}(MeV)	[b]Measurement References
Threshold Reactions				
$^{238}U(n,f)$	326.	± 2.0%	2.8	R20 (1983), R10, R17, R28
$^{237}Np(n,f)$	1365	± 2.0%	2.1	R20 (1983), R10, R17, R28
^{240}Pu	1337	± 2.4%	2.1	R20 (1983)
^{232}Th	89.4	± 3.0%	3.0	R20 (1981)
$^{93}Nb(n,n')$	149	± 7%	2.5	R28a (1982)
$^{115}In(n,n')$	195.	± 1.9%	2.7	R1 (1979), R3
$^{47}Ti(n,p)$	19.4	± 2.5%	3.9	R3 (1975), R12
$^{58}Ni(n,p)$	119.4	± 1.5%	4.2	R3 (1975), R12
$^{54}Fe(n,p)$	87.8	± 2.2%	4.3	R13 (1977), R3, R12
$^{46}Ti(n,p)$	14.2	± 2.5%	5.9	R3 (1975), R12
$^{63}Cu(n,\alpha)$	0.696	± 3.0%	7.7	R18 (1981), R19
$^{56}Fe(n,p)$	1.45	± 2.5%	7.6	R3 (1975)
$^{48}Ti(n,p)$	0.425	± 2.5%	8.4	R3 (1975), R12
$^{27}Al(n,\alpha)$	1.024	± 2.5%	8.6	R3 (1975), R12
Non-Threshold Reactions				
$^{197}Au(n,\gamma)$	77.4	± 2.5%	0.74	R1 (1979), R4
$^{239}Pu(n,f)$	1824.	± 1.9%	1.75	R20 (1983), R10, R17, R28
$^{235}U(n,f)$	1216.	± 1.6%	1.68	R20 (1983), R8, R9, R17, R22
$^{233}U(n,f)$	1893	± 2.5%	1.62	R20 (1983)
$^{241}Pu(n,f)$	1616	± 5.0%	1.63	R20 (1983)

[a] The total error includes a consideration of the agreement among independent measurements when they are available, as well as the error estimate given in the primary reference.

[b] The primary measurement chosen for each reaction is listed first.

TABLE X-10(B5). CALCULATED-TO-OBSERVED RATIOS OF CROSS SECTIONS

Spectrum: ^{252}Cf Spontaneous Fission Neutrons (NBS Evaluation)
Cross Section: Calculated: ENDF/B-V
Designation: XCF-5-N1
Entry Date: March 1982
Revised: June, 1983

Reaction	Cross Section (10^{-27} cm^2) Observed (Table X-9)	[a] Calculated (Table X-6(B5))	[b] Calculated-to-Observed
Fission			
^{239}Pu(n,f)	1824 ± 1.9 %	1792 ± 0.1 %	0.982 ± 1.9 %
^{235}U(n,f)	1216 ± 1.6 %	1236 ± 0.1 %	1.016 ± 1.6 %
^{233}U(n,f)	1893 ± 2.5 %	1904 ± 0.1 %	1.006 ± 2.5 %
^{238}U(n,f)	326 ± 2.0 %	313 ± 0.9 %	0.960 ± 2.2 %
^{237}Np(n,f)	1365 ± 2.0 %	1352 ± 0.6 %	0.990 ± 2.1 %
^{232}Th(n,f)	89.4 ± 3.0 %	78.1 ± 0.9 %	0.874 ± 3.1 %
^{240}Pu(n,f)	1337 ± 2.4 %	1356 ± 0.6 %	1.014 ± 2.5 %
^{241}Pu(n,f)	1616 ± 5.0 %	1595 ± 0.1 %	0.987 ± 5.0 %
Capture			
^{197}Au(n,α)	77.4 ± 2.5 %	76.6 ± 2.0 %	0.990 ± 3.2 %
Threshold			
^{93}Nb(n,n')	149 ± 7%	160.5 ± 0.8 %	1.08 ± 7%
^{115}In(n,n')	195 ± 1.9 %	182 ± 0.8 %	0.93 ± 2.1 %
^{47}Ti(n,p)	19.4 ± 2.5 %	24.1 ± 1.1 %	1.242 ± 2.7 %
^{58}Ni(n,p)	119.4 ± 1.5 %	114 ± 1.3 %	0.955 ± 2.8 %
^{54}Fe(n,p)	87.8 ± 2.2 %	88.3 ± 1.4 %	1.006 ± 2.6 %
^{46}Ti(n,p)	14.2 ± 2.5 %	13.5 ± 2.1 %	0.95 ± 3.3 %
^{63}Cu(n,α)	0.696 ± 3.0 %	0.758 ± 3.7 %	1.09 ± 4.8 %
^{56}Fe(n,p)	1.45 ± 2.5 %	1.414 ± 3.2 %	0.975 ± 4.1 %
^{27}Aℓ(n,α)	1.024 ± 2.5 %	1.059 ± 5.3 %	1.034 ± 5.9 %
^{48}Ti(n,p)	0.425 ± 2.5 %	0.409 ± 4.7 %	0.962 ± 5.3 %

(a) Errors correspond to ^{252}Cf fission spectrum uncertainties given in Table X-5 and propagated according to the second term of Eq. X-7, Section 5.a.3.

(b) Errors are quadrature sum of errors in columns 2 and 3.

TABLE X-11. OBSERVED SPECTRAL INDEXES

Spectrum: ^{252}Cf Spontaneous Fission Neutrons $\left[S_{\alpha/U8}\right]_{obs} = \left[\bar{\sigma}_{\alpha}/\bar{\sigma}_{f}(U238)\right]_{obs}$

Designation: XCF
Entry Date: May 1978
Revised: June, 1983

Reactions[a]	Reaction[b] Non-overlap Interval (MeV)	Spectral[c] Index (1σ)
Threshold Reactions		
$^{237}Np(n,f)/U8$	0.69 - 1.5	4.19 ± 1.5%
$^{232}Th/U8$	1.5 - 1.5	0.274 ± 2.6%
$^{240}Pu/U8$	0.69 - 1.5	4.10 ± 1.6%
$^{115}In(n,n')/U8$	1.2 - 1.5	0.598 ± 2.8%
$^{47}Ti(n,p)/U8$	1.9 - 1.5	0.0595 ± 3.2%
$^{58}Ni(n,p)/U8$	2.2 - 1.5	0.366 ± 2.1%
$^{54}Fe(n,p)/U8$	2.4 - 1.5	0.269 ± 3.0%
$^{46}Ti(n,p)/U8$	3.8 - 1.5	0.0436 ± 3.2%
$^{63}Cu(n,\alpha)/U8$	4.9 - 1.5	2.14 E-3± 3.6%
$^{56}Fe(n,p)/U8$	5.6 - 1.5	4.45 E-3± 3.2%
$^{48}Ti(n,p)/U8$	6.0 - 1.5	1.304E-3± 3.2%
$^{27}Al(n,\alpha)/U8$	6.6 - 1.5	3.14 E-3± 3.2%
Non-Threshold Reactions		
$^{239}Pu(n,f)/U8$	0.29 - 1.5	5.60 ± 1.2%
$^{235}U(n,f)/U8$	0.22 - 1.5	3.73 ± 1.2%
$^{233}U(n,f)/U8$	0.23 - 1.5	5.81 ± 2.1%
$^{241}Pu(n,f)/U8$	0.22 - 1.5	4.96 ± 4.8%
$^{197}Au(n,\gamma)/U8$	0.06 - 1.5	0.237 ± 3.2%

(a) All spectral indexes are taken relative to the U238(n,f) reaction labeled "U8".

(b) Calculated values of E_p are for p = 0.95 for each reaction in the ratio--see Table X-6(B5).

(c) Fission detector indexes are from simultaneous measurement of fission rates from both isotopes and hence show smaller uncertainties because of reduced neutron from flux errors. Other spectral indexes are formed from independent cross section measurements listed in Table X-9.

TABLE X-12(B5). CALCULATED-TO-OBSERVED RATIOS OF SPECTRAL INDEXES - NBS EVAL.

Spectrum: ^{252}Cf Spontaneous Fission Neutrons Designation: XCF-5-N1

Entry Date: March 1982

Spectral Index: Calculated: ENDF/BV/Table X-7(B5) Revised: June, 1983

Observed: Table X-11

$$C_{\alpha/U8} = \left[S_{\alpha/U8}\right]_{calc.} \Big/ \left[S_{\alpha/U8}\right]_{obs} \text{ or } C_{U8/\alpha}$$

Reactions[a]	Reaction[b] Non-Overlap Interval (MeV)	Calculated-to-Observed Values	ERROR (1σ) Observed Value	Calculated[c] Value	Total
Threshold Reactions					
U8/Np(n,f)	0.7 - 1.5	0.972	± 1.5%	± 0.5%	± 1.7%
U8/^{240}Pu	0.7 - 1.5	0.951	± 1.6%	± 0.5%	± 1.7%
U8/^{232}Th	1.5 - 1.5	1.100	± 2.6%	± 0.1%	± 2.6%
U8/In(n,n')	1.2 - 1.5	1.031	± 2.8%	± 0.2%	± 2.8%
^{47}Ti(n,p)/U8	1.5 - 1.9	1.29	± 3.2%	± 0.5%	± 3.2%
^{58}Ni(n,p)/U8	1.5 - 2.2	0.992	± 2.1%	± 0.7%	± 2.2%
^{54}Fe(n,p)/U8	1.5 - 2.4	1.04	± 3.0%	± 0.8%	± 3.1%
^{46}Ti(n,p)/U8	1.5 - 3.8	0.986	± 3.2%	± 1.7%	± 3.6%
^{63}Cu(n,α)/U8	1.5 - 4.9	1.129	± 3.6%		
^{56}Fe(n,p)/U8	1.5 - 5.6	1.013	± 3.2%	± 3.0%	± 4.4%
^{48}Ti(n,p)/U8	1.5 - 6.0	1.000	± 3.2%		-
Al(n,α)/U8	1.5 - 6.6	1.076	± 3.2%	± 1.5%	± 3.5%
Non-Threshold Reactions					
U8/^{239}Pu(n,f)	0.3 - 1.5 low 5.7 - 7.2 high	0.981	± 1.2%	± 0.8%	± 1.4%
U8/^{235}U(n,f)	0.2 - 1.5 low 5.8 - 7.2 high	0.946	± 1.2%	± 1.0%	± 1.6%
U8/^{233}U(n,f)	0.2 - 1.5 low 5.5 - 7.2 high	0.957	± 2.1%	± 1.0%	± 2.3%
U8/^{241}Pu(n,f)	0.2 - 1.5 low 5.6 - 7.2 high	0.973	± 4.8%	± 1.0%	± 4.9%
U8/Au(n,γ)	0.06 - 1.5 low 3.0 - 7.2 high	0.970	± 3.2%	-	-

[a] Spectral indexes are taken with respect to the ^{238}U(n,f) reaction labeled "U8" with the lower energy response detector in the denominator. The calculated-to-observed ratio for σ_f(U238) is 0.960 from Table X-10(B5).

[b] Non-overlapping energy response interval between the 95% response energy boundary for each detector. Also see footnote (b) Table X-11. For non-threshold reactions two non-overlapping energy intervals are listed: The 95% exclusion interval is labeled "low" and the 5% exclusion interval "high." See Table X-6(B5).

[c] Errors in the calculated value correspond to ^{252}Cf fission spectrum uncertainties only, given in Table X-5, and propagated according to the second term of Eq. X-8, Sect. 5.a.3. Error in σ(U238) calculated is ± 0.9% (1σ) from Table X-10(B5); error in the observed value is from Table X-11.

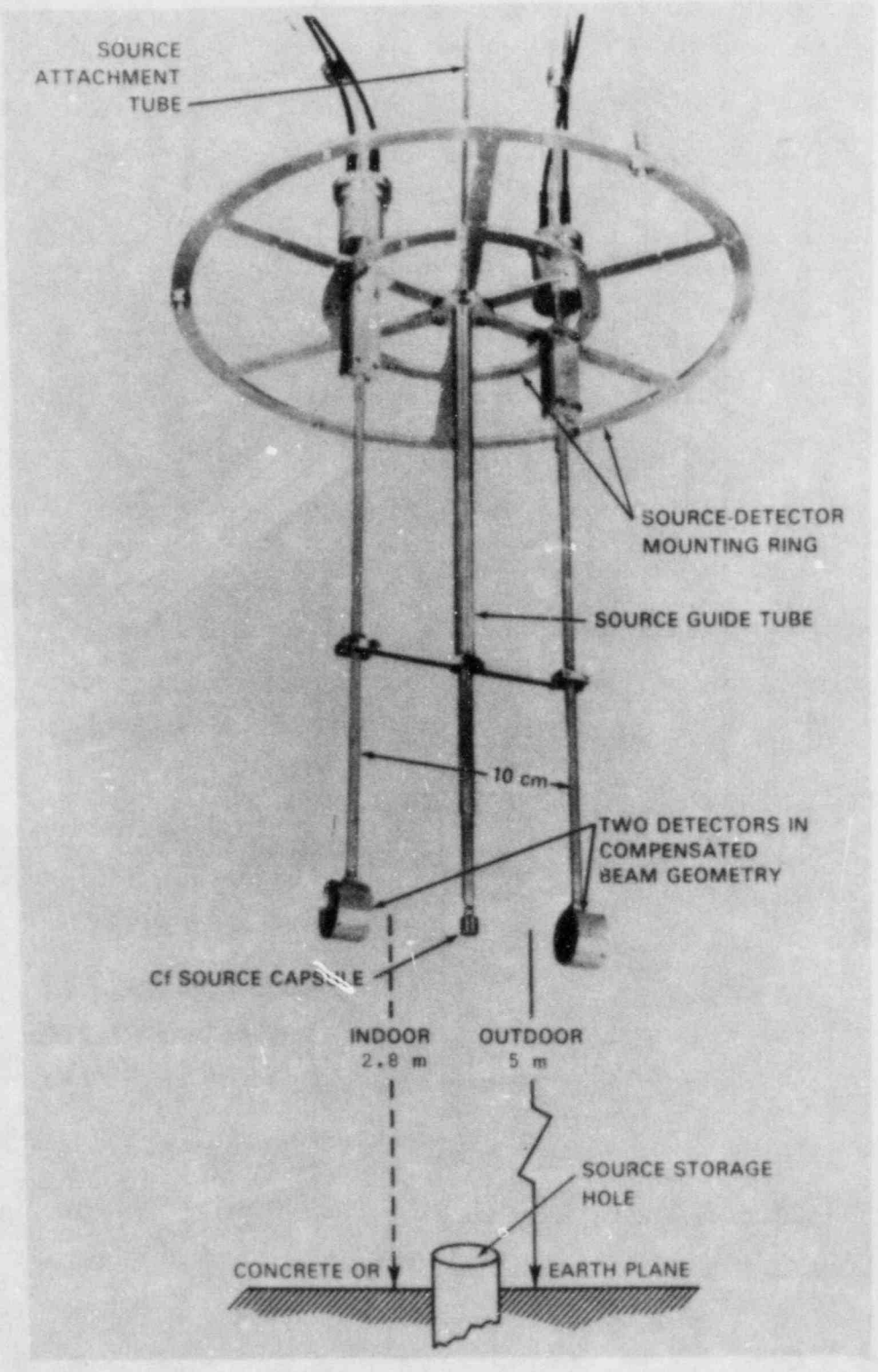

FIGURE X-1. Schematic of typical source and detector arrangement at NBS ^{252}Cf Fission Spectrum Irradiation Facilities.

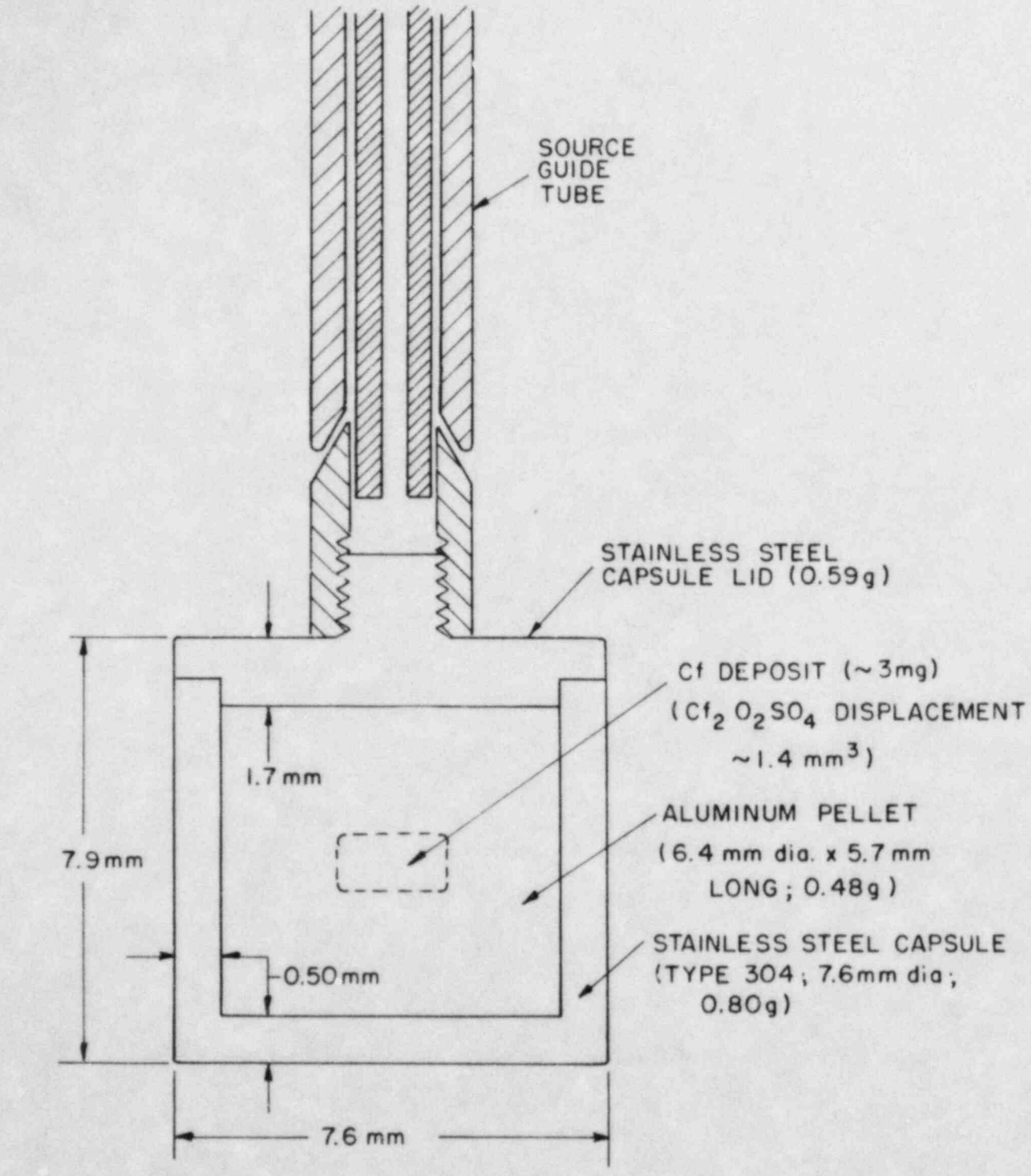

FIGURE X-2. Scattering materials in the low-mass ^{252}Cf fission neutron source capsule.

COMPENDIUM OF BENCHMARK NEUTRON FIELDS

FOR

REACTOR DOSIMETRY

STANDARD NEUTRON FIELD ENTRY

Part IB: ^{235}U Thermal-Neutron-Induced Fission Neutron Spectrum

TABLE OF CONTENTS

1. IDENTIFICATIONS

1.a. NEUTRON FIELD: ^{235}U THERMAL-NEUTRON-INDUCED FISSION SPECTRUM

CLASSIFICATION: STANDARD NEUTRON FIELD

1.b. DESIGNATION: XU5-5-N1

NBS evaluation in 1975 of documented differential spectrum measurements. References: F1 (1975) and F2 (1975)

1.c. ENTRY DATE: June, 1978 REVISIONS: July, 1983

1.d. GENERIC DESCRIPTION

The standard spectrum consists of neutrons from the thermal-neutron-induced fission of ^{235}U. The median energy of the spectrum is 1.57 MeV with 98% of the spectrum between 0.1 and 7 MeV. Contributions from scattered neutrons and other backgrounds must be small and well defined if a particular ^{235}U fission source facility is to qualify as a standard neutron field. Fission neutron fluxes of up to 2×10^{10} n/(cm^{2}s) can be generated in reactor thermal-columns with spherical cavities. Source to detector distances are small in such arrangements and the pure fission neutron field is restricted to a volume of one to ten cubic centimeters.

Extensive and well documented measurements exist for the ^{235}U fission neutron spectrum and its close relative the ^{252}Cf fission spectrum. Therefore these two standard fields are much better known than any other benchmark employed for reactor dosimetry calibration. In the energy range above 2 MeV many neutron fields in and around test and power reactors have fission-spectrum-like components.

1.e. FACILITY LOCATIONS:

National Bureau of Standards
Center for Radiation Research
Gaithersburg, Md 20878
U.S.A.

Contact: Dale McGarry and James Grundl
Phone: 301-921-2767

Research Reactor Institute
Kyoto University
Kyoto, Japan

Contact: Itsuro Kimura

Centre D'Etude De L'Energie Nucléaire
SCK/CEN
BOERETANG 200
B-2400 MOL
Belgium

Contact: Albert Fabry or Reactor Physics Section
Phone: (011) 32 14 311801

1.f. CONTACT FOR INFORMATION:

James Grundl or Dale McGarry
Center for Radiation Research
National Bureau of Standards
Gaithersburg, Maryland 20878
U.S.A.

Phone: 301-921-2767

2. SUMMARY INFORMATION

2.a. AVAILABLE FISSION NEUTRON EXPOSURES (NBS ONLY)

o	flux intensity	$\sim 2 \times 10^{10}$ n/(cm^{2}s)
o	nominal maximum fluence	1.5×10^{15} n/cm^2
o	accuracy of free-field fluence	±2.6% (1σ)

2.b. CORRECTIONS FOR NEUTRON SCATTERING IN SOURCE AND ENCAPSULATIONS

o	neutron fluence	$\lesssim$ 2.5%
o	reaction rates (E_{95} < 3 MeV)	(+0.6 to 0)%
o	reaction rates (E_{95} > 3 MeV)	(0 to -1.0)%

3. FACILITY DESCRIPTION

Fission neutron sources in spherical cavities are in operation at two reactor thermal-column facilities. At SCK/CEN in Belgium a one meter diameter cavity in graphite provides fission neutron fluxes in an arrangement where low-energy background components may be established experimentally (F9, F10, F11). At NBS in the United States a 30-cm diameter spherical cavity generates a higher fission neutron flux than is available at the SCK/CEN facility, but, because of the smaller diameter cavity and a less simple source detector geometry, low-energy backgrounds are more difficult to estimate (F12, F13, F14). At Kyoto University, a high- and low-power fission plate converter system is enclosed in a cubical concrete shield with ~2.5 m wall-to-wall separation (F15). Only the NBS facility will be described in this compendium.

3.a. CONFIGURATION AND CHARACTERISTICS (F12)

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 thk) placed outside a cadmium box that encloses passive detectors for exposure. The cavity and source-detector arrangement are shown in Fig X-3. In 1983, the source-detector assembly was reworked so that the cadmium box and detectors may be removed separately from the fission disks. The main features of the source detector geometry otherwise are unchanged. Fission neutron fluxes of $\sim 2 \times 10^{10}$ 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 12 mm, axial flux gradients are mild: the flux at 1 mm from the midplane is 4% greater than at the midplane. Center-to-edge flux ratios are more severe--see section 4a.

Fission neutron return for spherical cavities in graphite has been studied extensively by means of neutron transport calculations and by experiment in connection with the general development of standard neutron fields in spherical geometry (F9, F14, F16, F17, F18). For the 30-cm cavity at NBS, the response of threshold fission detectors to fission neutrons returning from the cavity walls to the cavity center is less than 0.3% of the uncollided fission neutron flux from the fission source disks; for detectors with thresholds above 1 MeV, the response to the cavity return flux 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 SCK/CEN (F36). More important is fission neutron scattering in the source-detector capsule. 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 (F12). Corrections for activation rates are generally between 0.5% and 3%.

The NBS fission source is not used generally for low-energy integral detectors because of uncertainties in the graphite return background. The graphite return correction for the $^{235}U(n,f)$ reaction, for example, is nearly 10% and self-shielding of the near-1/E cavity return flux by the fission disks and by a ^{235}U activation foil makes this correction uncertain to within a factor of about one third. The one meter cavity arrangement at SCK/CEN is better suited for low-energy response detectors (F9, F10, F11).

3.b. IRRADIATION PROCEDURES:

Disk detector irradiations are monitored with indium monitor foils placed between the detectors. The neutron fluence in the detector capsule is

obtained from the measured In(n,n') activity on the basis of a fluence transfer calibration carried out at the ^{252}Cf irradiation facility. In this procedure, the ratio of the In(n,n') 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 indium detector position in the source-detector capsule.

The expression which governs this flux transfer is

$$\Phi_{cvy} = \frac{G_{cf}}{G_{cvy}} \cdot \frac{\sigma_{cf}(>E_p)}{\sigma_{cvy}(>E_p)} \cdot \frac{\psi_{cf}(>E_p)}{\psi_{cvy}(>E_p)} \cdot \frac{D_{cvy}}{D_{cf}} \cdot \Phi_{cf} \qquad \text{(X-9)}$$

where Φ = fission neutron fluence and "cvy" denotes the ^{235}U cavity fission source and "cf" the ^{252}Cf fission neutron calibration flux.

The ratios on the right side of Eq. X-9 involve the decay correction factors (G), the In(n,n') truncated reaction cross sections (σ), the spectrum fractions (ψ), and the ratio of Indium detector activity in the cavity to that at the Cf irradiation facility. (See Sections 5.a. and 5.b. for integral detector reaction rate formulations.) For indium, the truncated cross section ratio is 1.023 ± 0.010 and the spectrum fraction ratio is 1.027 ± 0.013. Combined with a typical error of ~±0.5 for scattering corrections, ±0.6% for detector response ratios and decay correction factors, and ±1.6% for the ^{252}Cf fission neutron flux, the flux transfer procedure yields a free-field ^{235}U cavity fission neutron flux to an accuracy of ±2.5% (1σ).

Fission neutron fluences of ~1.5 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 that can be irradiated to a diameter of 12 mm and a thickness of a few millimeters.

3.c. SPECIFICATIONS FOR TRANSPORT CALCULATIONS

Neutron transport calculations are required for two types of corrections: (1) neutron scattering in the fission source disks and in the source and detector capsule; and (2) the neutron return flux from the graphite.

The return flux at the cavity center is a calculations problem of general interest and for this purpose the geometric and physical arrangements of the cavity will be described. The NBSR graphite thermal column is effectively infinite for fission neutrons originating in the cavity; its graphite density is 1.74 gm/cm^3, and the cavity is a well formed sphere 29.8 cm in diameter. The fission source disks may be considered as a point source at the cavity center.

Scattering in the source-detector capsule is more complex and has been established on the basis of Monte Carlo calculations (F12). The main scattering components in these calculations (see Fig. X-3) are the cadmium box (22.5 mm dia., 0.076 mm thick), the source disks (16 mm dia. x 0.13 mm thk), and the foil samples (typically less than 2 gm). Although the indium flux monitor foils are in the same geometry as the foil samples, they do not correctly monitor energy-angle effects for higher threshold detectors. The latter effect, along with energy transfer by inelastic scattering, are the largest uncertainties in detector response corrections for neutron scattering.

4. NEUTRON FIELD CHARACTERIZATION

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

Fission Spectrum 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

and a short multigroup-flux display, $\phi(E)\Delta E$, as follows:

^{235}U Fission Spectrum in Seven-Groups

Energy Bounds:	0	0.25	0.8	1.5	2.3	3.7	8	12 MeV
^{235}U	(0.054)	0.197	0.229	0.195	0.192	0.127	0.006	

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 ~0.05 MeV.

4.a. NEUTRON FLUX (see also SECTION 3a.)

The combined source strength for the two fission disks in the NBSR thermal column cavity is approximately 1×10^{11} n/s. Since this source strength is difficult to determine accurately, the ^{235}U fission neutron flux is established by means of flux transfer from the ^{252}Cf fission neutron irradiation facility as described in section 3.b. The neutron flux intensity at the midplane of

the source-detector capsule for a source disk separation of 11 mm is 0.13 n/cm^2s per source neutron corresponding to a flux of $\sim 2 \times 10^{10}$ n/cm^2s with the reactor at full power. The fission neutron flux is minimum at the midplane between the two source disks. At 1 mm above or below the midplane, the mean flux for a 12 mm diameter disk detector is ~4% higher than for the same disk at the mid plane. Radial variations of the flux are more severe: center-to-edge flux ratios are 1.2 for 12 mm diameter disks. This gradient does not require that detector disks be extremely uniform. For example, a radial mass nonuniformity of 20% distributed linearly in a detector changes the activation rate by less than 0.5% compared to a uniform mass distribution. Also, the indium flux monitor included in each irradiation package yields a flux averaged over the radial variation.

The fission neutron flux 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 (μ_{sc}) from its free-field activity attributable to both neutron scattering and cavity return:

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

This scattering correction depends markedly upon reaction threshold because of energy-angle correlation in the near by materials of the encapsulation. Components of μ_{sc}, obtained from detailed Monte-Carlo calculations of the source detector capsule which include a representative foil stack, are as follows:

	neutron flux	threshold reaction rates		
		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%
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 to 0.5)%	-(0.4 to 1.0)%	-(0.8 to 1.5)%
μ_{sc}(%):		+(0.7 to 1.2)%	-(0.6 to 1.2)%	-(2 to 3)%

The spread of the foil-stack component of μ_{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 error of ±0.6% is assigned to μ_{sc}.

It is the small source to detector distance which makes it possible to obtain pure fission neutron fluxes in small cavities. The cavity return flux at the cavity center is about 1.2% of the fission neutron flux. This return flux gives rise to threshold detector responses that are between 0.05% and 0.39% of the free-field fission neutron response depending upon the reaction threshold.

4.b. NEUTRON SPECTRA

4.b.1. Calculation. Fission neutron spectra are determined most accurately by laboratory measurement. There is no adequate nuclear theory for confidently calculating fission neutron spectra without reference to experiment. (See, however, Ref. F19 and earlier work cited therein.)

4.b.2. Measurements (F2, F1). See Section 4.b.2. in Part 1A of this Compendium, "^{252}Cf Spontaneous Fission Neutron Spectra."

4.b.3. Evaluated Spectrum. The NBS fission spectrum evaluation is chosen for this compendium because of the consistency of its procedures for both ^{235}U and ^{252}Cf fission spectra, and because it includes an articulated error estimate (F2, F1). Cross section comparisons with the ENDF/B-V ^{235}U fission spectrum shape will be given as a supplement.

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

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 X-4, Part IA. Group fluxes for other energy group structures may be derived from this tabulation with the interpolation recommended in the footnote to the table. Spectrum uncertainties, given in Table X-5 of Part IA, are based on departures of subsets of measured data from the final adjusted Maxwellian of the evaluation. This estimate, carried out in a seven-group structure, includes both 1σ and 2σ errors.

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

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

$$a = 0.988 \text{ MeV}^{-1} \quad b = 2.249 \text{ MeV}^{-1}$$

median energy: $E_p(p=0.5) = 1.64$ MeV

Contemporary time-of-flight data was emphasized in the ENDF/B-V evaluation, as compared with the NBS evaluation which considered all well-documented data. Differences between the two evaluated spectra, in terms of calculated spectral indexes, are shown in Table X-14a(85) and X-14a.1.

5. INTEGRAL DETECTOR RESPONSE

Performing neutron dosimetry measurements with integral detectors requires the use of benchmark neutron fields for calibrating measurement techniques and for referencing data interpretation methods. In order to carry out these measurement assurance procedures, and in particular for estimating uncertainties, it is necessary to distinguish between calculated and measured reaction probabilities and to identify parameters which characterize the energy response of each detector in the various spectra to which it is exposed. A formulation which meets these requirements is described in Section 5 of Part IA. In this section tables of response parameters including cross sections, spectral indexes, and energy response ranges are given for a number of integral detectors. Measured reaction probabilities are presented in Section 5.b. followed by comparisons of measured and calculated cross sections in Section 5.c.

5.a. CALCULATED REACTION PROBABILITIES

(See Section 5.a. in Part IA for reaction probability formulations, parameters, and error propagation.)

5.a.1. Spectrum Response Table Basic integral detector response parameters for ^{235}U fission neutrons are given in Table X-13(B5). The spectrum-averaged cross sections listed in column 2 are the full-spectrum averaged values above a cadmium cut-off energy of 0.4 eV and are followed in column 3 by cross sections truncated at p = 0.95. The energy dependent cross sections employed are those of the ENDF/B-V dosimetry file reduced to 620 energy groups; spectrum averaging is carried out with the NBS DETAN code. The spectrum fraction, [$\psi(>E_p)$, p = 0.95], follows in column 4. Energy response characteristics, given in the last three columns, are the median response energy and the energy response range, i.e. the lower- and upper-energy bounds that include 90% of the detector response. The low-energy bound, E_p(p = 0.95), is the truncation energy corresponding to the truncated cross section given in column 3.

5.a.2. Spectral Indexes (See Part IA, section 5.a.2 for formulations) A selected set of spectral indexes calculated for full-spectrum-averaged cross sections from Table X-13(B5) is given in Table X-14(B5); a corresponding set for truncated cross sections is in Table X-15. All of these spectral indexes are for the NBS evaluated fission spectrum shape described in Section 4.b.3. Tables 14a(B5) and 14.a.1 list spectral indexes for the ENDF/B-V fission spectrum and a comparison of the two evaluations.

5.b. MEASURED REACTION PROBABILITIES (See Part IA, section 5.b)

The important observed quantities for reactor dosimetry are the ratios of reaction probabilities among detectors which, according to Eq. X-1 (Part IA), are proportional to the ratio of spectrum-averaged cross sections. These

cross section ratios, when they are sensitive to spectrum shape, are called spectral indexes as discussed in Section 5.a.2 of Part IA. As an observed quantity it is obtained by putting the measured reaction probability (Eq. X-10) in place of the calculated value in Eq. X-1 and dividing:

$$\left[S_{\alpha/\beta}\right]_{obs} = \left[R_\alpha/R_\beta\right]_{obs} \quad . \qquad (X\text{-}11)$$

The subscripts α and β refer to the two detector reactions involved.

5.c. MEASURED CROSS SECTIONS AND CALCULATED-TO-OBSERVED RATIOS

For this compendium, the primary integral detector quantities are cross sections and the spectral indexes taken with respect to the $^{238}U(n,f)$ reaction. This index will be denoted $S_{u8/\alpha}$ or $S_{\alpha/u8}$ depending upon whether the energy response range of a particular detector is higher or lower than that of $^{238}U(n,f)$. Available integral detector data for the ^{235}U fission spectrum are to be found in References F1, F10, F15, R24, F11, F13, R9, and R29. Collections of evaluated measured cross sections may be found in References R23, and R30, R31, R32.

A new and restricted evaluation of measurements reported before July, 1983, was performed at NBS for detector reactions considered important for reactor technology. The evaluation begins with the absolute $^{235}U(n,f)$ cross section measured for ^{252}Cf fission neutrons. Since the $^{235}U(n,f)$ cross section for ^{252}Cf and ^{235}U fission neutrons is the same to within a conservative upper bound of ±0.6%, cross section measurements relative to $^{235}U(n,f)$ in the ^{235}U fission spectrum can be scaled to the ^{252}Cf cross section for $^{235}U(n,f)$ with little additional error. Ratio measurements in the cavity fission source at SCK/CEN (F10 and R24) have established cross sections relative to $^{235}U(n,f)$ for three normalizing reactions: $^{238}U(n,f)$, $In(n,n')$, and $^{58}Ni(n,p)$. Other

measurements with cavity fission sources and with fission plates are evaluated with one of these three reactions taken as the flux monitor. Results are given in Table X-16 and X-18. The uncertainties are one standard deviation and include estimates of systematic errors.

Calculated-to-observed ratios of reaction cross sections are listed in Table X-17(B5) and include comparison with both the NBS evaluated and the ENDF/B-V fission spectrum shapes. The errors in calculated cross section errors in correspond to the uncertainties in the fission spectrum (Table X-5) propagated according to the second term of Eq. X-7 (Part IA, section 5.a). Only the ^{47}Ti(n,p) reaction shows a calculated-to-observed ratio in excess of 10%. The standard deviations of eleven C/E ratios in Table X-17(B5) for the NBS spectrum, excluding ^{47}Ti(n,p), are 0.029 about an average ratio of 1.048, and 0.031 about a mean C/E value of 0.955. The standard deviations for the same ten C/E ratios in the ENDV/B-V spectrum are 0.019 about an average ratio of 1.029, and 0.031 about a mean value of 0.984.

Calculated-to-observed ratios of spectral indexes, $C_{\alpha/U8}$, are presented in Table X-19(B5). The calculated indexes correspond to full-spectrum averaged cross sections ratios (NBS Evaluation) from Table X-14(B5). Non-overlapping energy response intervals for the two detectors involved are given in column 2, and contributing errors for observed and calculated indexes are listed separately.

Propagated errors for the calculated indexes assess only the contribution of spectrum uncertainties listed in Table X-5, Part IA. The propagation recipe for this partial error is that of Eq. X-8, (Part IA, sect. 5.a.3).

The $C_{\alpha/U8}$ ratios in Table-X19(B5) validate integral detector measurement techniques and detector cross sections. The difficulties with the titanium isotopes, which was so evident with the ENDF/B-IV dosimetry file, are improved except for ^{47}Ti where a 30% discrepancy remains. The discrepancy of $C_{U8/\beta}$ for U235 fission is in excess of 5% confirming the similar result obtained with the ^{252}Cf fission spectrum. Agreement otherwise is within ±5% for all detectors. The standard deviations of ten $C_{\alpha/U8}$ ratios (^{47}Ti excluded) in Table X-19(B5) are 0.016 about an average ratio of 1.034, and 0.038 about a mean C/E value of 0.996.

TABLE X-13(B5). INTEGRAL DETECTOR RESPONSE PARAMETERS

Spectrum: ^{235}U thermal Neutron-Induced Fission (NBS evaluation, F1, F2)

Designation: XU5-5-N1

Entry Date: June, 1978

Cross Sections: ENDF/B-V Dosimetry File (R25)

Revised: March, 1982

Detector Reaction	Cross Section[a] $\sigma(> E_p)$		Spectrum[b] Fraction	Median[c] Response Energy	Response Range[c]	
	E_p=0.4 eV	$E_p(p=0.95)$	$\psi(> E_p)$	$E_p(p=0.5)$	$E_p(p=0.95)$	$E_p(p=0.05)$
	(barns)	(barns)		(MeV)	(MeV)	(MeV)
Spectrum Check						
Constant	1.000	1.000	0.950	1.57	0.24	5.1
recip. vel., (1/v)[d]	0.1151	0.1103	0.991	0.921	0.076	4.0
Fissionable Mat'ls.						
$^{239}Pu(n,f)$	1.786	1.800	0.9424	1.65	0.26	5.2
$^{235}U(n,f)$	1.236	1.221	0.962	1.57	0.20	5.2
$^{233}U(n,f)$	1.912	1.895	0.959	1.51	0.21	5.0
$^{238}U(n,f)$	0.2947	0.532	0.527	2.67	1.48	6.7
$^{237}Np(n,f)$	1.322	1.578	0.7958	1.96	0.670	5.7
$^{232}Th(n,f)$	0.0724	0.1322	0.520	2.86	1.50	7.3
$^{240}Pu(n,f)$	1.326	1.585	0.795	1.97	0.67	5.6
$^{241}Pu(n,f)$	1.600	1.579	0.963	1.53	0.19	5.1
$^{238}U(n,\gamma)$	0.0721	0.0691	0.991	0.87	0.075	2.7
$^{232}Th(n,\gamma)$	0.0942	0.0906	0.988	0.88	0.094	2.8
Capture Reactions						
^{23}Na	2.817 E-4	2.69 E-4	0.995	0.845	0.053	4.3
^{58}Fe	1.712 E-3	1.634 E-3	0.995	0.948	0.049	4.5
^{59}Co	6.28 E-3	6.00 E-3	0.995	1.024	0.054	3.8
^{63}Cu	1.008 E-2	9.62 E-3	0.995	0.917	0.055	3.8
^{115}In	0.1266	0.1220	0.986	1.080	0.102	2.9
^{197}Au	0.0810	0.0774	0.994	0.695	0.058	2.8

TABLE X-13(B5). INTEGRAL DETECTOR RESPONSE PARAMETERS (continued)

Spectrum: ^{235}U Thermal Neutron-Induced Fission (NBS evaluation, F1, F2)

Designation: XU5-5-N1

Entry Date: June, 1978

Cross Sections: ENDF/B-V Dosimetry File (R25)

Revised: March, 1982

Detector Reaction	Cross Section[a] $\sigma(> E_p)$		Spectrum[b] Fraction	Median[c] Response Energy	Response Range[c]	
	E_p=0.4 eV	$E_p(p=0.95)$	$\psi(> E_p)$	$E_p(p=0.5)$	$E_p(p=0.95)$	$E_p(p=0.05)$
	(barns)	(barns)		(MeV)	(MeV)	(MeV)
Helium Production						
$^{10}B(n,\alpha)$	0.499	0.478	0.993	1.16	0.066	5.2
$^{6}Li(n,\alpha)$	0.465	0.455	0.971	1.25	0.167	5.7
Threshold Reactions						
$^{115}In(n,n')$	0.1734	0.264	0.624	2.6	1.16	5.9
$^{47}Ti(n,p)$	0.02159	0.0487	0.421	3.8	1.87	7.5
$^{32}S(n,p)$	0.0676	0.198	0.324	4.0	2.3	7.4
$^{58}Ni(n,p)$	0.1009	0.2645	0.36?	4.1	2.1	7.5
$^{54}Fe(n,p)$	0.0778	0.2334	0.317	4.2	2.3	7.6
$^{46}Ti(n,p)$	0.01081	0.0806	0.1274	5.7	3.8	9.3
$^{27}Al(n,p)$	4.121 E-3	2.522 E-2	0.1553	5.8	3.5	9.3
$^{56}Fe(n,p)$	1.006 E-3	2.446 E-2	0.0391	7.3	5.5	11.2
$^{63}Cu(n,\alpha)$	5.40 E-4	7.80 E-3	6.58 E-2	7.3	4.7	11.1
$^{27}Al(n,\alpha)$	6.93 E-4	3.54 E-2	1.861 E-2	8.4	6.5	11.9
$^{48}Ti(n,p)$	2.726 E-4	9.21 E-3	2.813 E-2	8.1	5.9	12.2
Additions[e]						
$^{103}Rh(n,n')$	0.682	0.835	0.776	2.3	0.72	5.7
$^{93}Nb(n,n')$	0.1543	0.2159	0.679	2.5	0.99	5.6
$^{60}Ni(n,p)$	2.528 E-3	3.99 E-2	0.0603	7.0	4.9	10.4
$^{55}Mn(n,2n)$	0.2016 E-3	0.366	0.523 E-3	12.7	11.1	16
dpa	854	973	0.83	2.5	0.57	6.3

Footnotes for TABLE X-13(B5)

(a) The value given in column 2 is the spectrum averaged cross section above a cadmium cut-off of 0.4 eV. The truncated cross section in column 3 is for a truncation energy (column 6) above which 95% of the detector response occurs. A spectrum average cross section truncated at energy E_p is given by

$$\sigma(> E_p) = \int_{E_p}^{\infty} \sigma(E)\psi(E)dE \Big/ \int_{E_p}^{\infty} \psi(E)dE \quad .$$

(b) The fraction of the spectrum above E_p ($p = 0.95$): $\psi(> E_p) = \int_{E_p}^{\infty} \psi(E)dE$. The full-spectrum-averaged cross section $\sigma(> 0.4$ eV) for $p = 0.95$ is equal to $\sigma(> E_p) \cdot \psi(> E_p)/0.95$.

(c) The fractions $p = 0.95$, 0.5, and 0.05 define energies above which 95%, 50% (median), and 5% of the detector response occurs, respectively. E_p is defined by the relation

$$\int_{E_p}^{\infty} \sigma(E)\psi(E)dE = p \cdot [\sigma(> 0.4 \text{ eV})]$$

where $E_p(p = 1) = 0.4$ eV, and $E(p = 0) = 20$ MeV; and $\int_{0.4\text{eV}}^{\infty} \psi(E)dE = 1$.

(d) Normalization is $\int_{0.4\text{ eV}}^{18\text{ MeV}} \sigma_{1/v}(E)\psi(E)dE = 1$.

(e) Cross sections not on the ENDF/BV Dosimetry File:

- Nb(n,n') is from the IRDF Dosimetry File (R25a)
- Rh(n,n') is from data reported in Ref. (R26).
- dpa, the atom displacement cross section, is from ASTM Standard Practice E693-79. (R27)

TABLE X-14(B5). CALCULATED SPECTRAL INDEXES--NBS EVAL. FISS. SPEC.

Spectrum: ^{235}U Thermal-Neutron Induced Fission (NBS evaluation, F1, F2)

Designation: XU5-5-N1

Entry Date: June, 1978

Spectral Index: Full-Spectrum Cross Sections from Table X-13(B5) (ENDF/B-V Dosimetry File)

Revised: March, 1982

$$\left[S_{\alpha/\beta}\right]_{calc.} = \left[\bar{\sigma}_\alpha/\bar{\sigma}_\beta\right]_{calc.}$$

$\alpha \setminus \beta$	$^{235}U(n,f)$ ($\bar{\sigma}$=1.236b)	$^{238}U(n,f)$ ($\bar{\sigma}$=0.295b)
Fission		
^{239}Pu	1.445	6.06
^{235}U	1.000	4.19
^{233}U	1.547	6.49
^{238}U	0.2384	1.000
^{237}Np	1.070	4.49
^{240}Pu	1.073	4.50
He Prod.		
$^{10}B(n,\alpha)$	0.404	1.693
$^{6}Li(n,\alpha)$	0.376	1.578
Capture		
^{238}U	0.0583	0.245
^{232}Th	0.0762	0.320
^{197}Au	0.0656	0.2749
^{59}Co	5.08 E-3	0.0213
^{58}Fe	1.385 E-3	5.81 E-3
Threshold		
$^{115}In(n,n')$	0.1403	0.588
$^{47}Ti(n,p)$	0.01747	0.0733
$^{58}Ni(n,p)$	0.0816	0.342
$^{54}Fe(n,p)$	0.0630	0.2640
$^{46}Ti(n,p)$	8.75 E-3	0.0367
$^{56}Fe(n,p)$	8.14 E-4	3.41 E-3
$^{63}Cu(n,\alpha)$	4.37 E-4	1.832 E-3
$^{27}Al(n,\alpha)$	5.61 E-4	2.351 E-3
$^{48}Ti(n,p)$	2.21 E-4	9.25 E-4
Additions		
$^{103}Rh(n,n')$	0.552	2.314
$^{93}Nb(n,n')$	0.1248	0.523
$^{32}S(n,p)$	0.0547	0.229

TABLE X-14a(B5). CALCULATED SPECTRAL INDEXES--ENDF/BV FISS. SPEC.

Spectrum: ^{235}U Thermal-Neutron Induced Fission (ENDF/B-V Nuclear Data File)

Designation: XU5-9-E5

Data Date: March, 1982

Spectral Index: Full-Spectrum Cross Sections (ENDF/B-V Dosimetry File)

Revised:

$$\left[S_{\alpha/\beta}\right]_{calc.} = \left[\bar{\sigma}_\alpha/\bar{\sigma}_\beta\right]_{calc.}$$

$\alpha \setminus \beta$	$^{235}U(n,f)$ ($\bar{\sigma}$=1.236b)	$^{238}U(n,f)$ ($\bar{\sigma}$=0.3052b)
Fission		
^{239}Pu	1.449	5.87
^{235}U	1.000	4.05
^{233}U	1.543	6.25
^{238}U	0.2469	1.000
^{237}Np	1.090	4.41
He Prod.		
$^{10}B(n,\alpha)$	0.397	1.607
$^{6}Li(n,\alpha)$	0.368	1.489
Capture		
^{238}U	0.0568	0.2302
^{197}Au	0.0633	0.2565
^{59}Co	5.00 E-3	0.02023
^{58}Fe	1.367 E-3	5.53 E-3

$\alpha \setminus \beta$	$^{235}U(n,f)$ ($\bar{\sigma}$=1.236b)	$^{238}U(n,f)$ ($\bar{\sigma}$=0.3052b)
Threshold		
$^{115}In(n,n')$	0.1451	0.587
$^{47}Ti(n,p)$	0.01817	0.0736
$^{58}Ni(n,p)$	0.0850	0.344
$^{54}Fe(n,p)$	0.0656	0.2655
$^{46}Ti(n,p)$	9.04 E-3	0.0366
$^{56}Fe(n,p)$	8.38 E-4	3.39 E-3
$^{63}Cu(n,\alpha)$	4.51 E-4	1.829 E-3
$^{27}Al(n,\alpha)$	5.82 E-4	2.357 E-3
$^{48}Ti(n,p)$	2.280 E-4	9.23 E-4
Additions		
$^{103}Rh(n,n')$	0.564	2.284
$^{93}Nb(n,n')$	0.1292	0.523
$^{32}S(n,p)$	0.0570	0.231

TABLE X-14a.1. CALCULATED SPECTRAL INDEX RATIO: NBS EVALUATION VS ENDF/B-V

Spectrum: ^{235}U Thermal-Neutron-Induced Fission: (NBS Eval./ENDF/B-V)

Designation: XU5-5-N1 / XU5-9-E5

Entry Date: March, 1982

Spectral Index: Full-Spectrum Cross Sections (ENDF/B-V Dosimetry File)

Revised:

$$\left[S_{\alpha/\beta}\right]_{\chi NBS} \Big/ \left[S_{\alpha/\beta}\right]_{\chi BV}$$

$\alpha \setminus \beta$	$^{235}U(n,f)$	$^{238}U(n,f)$	$\alpha \setminus \beta$	$^{235}U(n,f)$	$^{238}U(n,f)$
Fission			Threshold		
^{239}Pu	0.997	1.032	$^{115}In(n,n')$	0.967	1.001
^{235}U	1.000	1.035	$^{47}Ti(n,p)$	0.962	0.996
^{233}U	1.003	1.038	$^{58}Ni(n,p)$	0.960	0.994
^{238}U	0.966	1.000	$^{54}Fe(n,p)$	0.960	0.994
^{237}Np	0.982	1.018	$^{46}Ti(n,p)$	0.971	1.003
He Prod.			$^{56}Fe(n,p)$	0.971	1.006
$^{10}B(n,\alpha)$	1.018	1.053	$^{63}Cu(n,\alpha)$	0.968	1.002
$^{6}Li(n,\alpha)$	1.022	1.060	$^{27}Al(n,\alpha)$	0.965	0.997
			$^{48}Ti(n,p)$	0.970	1.002
Capture					
^{238}U	1.026	1.064	Additions		
^{197}Au	1.036	1.072	$^{103}Rh(n,n')$	0.979	1.013
^{59}Co	1.018	1.053	$^{93}Nb(n,n')$	0.966	1.000
^{58}Fe	1.013	1.051	$^{32}S(n,p)$	0.960	0.991

TABLE X-15(B5). CALCULATED SPECTRAL INDEXES--TRUNCATED

Spectrum: ^{235}U Thermal-Neutron-Induced Fission Neutrons (NBS Eval., F1, F2)

Designation: XU5-5-N1

Entry Date: June, 1978

Revised: March, 1982

Spectral Index: Truncated Cross Sections from Table X-13(B5), Column 3 (ENDF/B-V Dosimetry File)

$$\left[S_{\alpha/\beta}\right]_{calc.} \equiv \left[\bar{\sigma}_{\alpha}(> E_p) \Big/ \bar{\sigma}_{\beta}(> E_p)\right]_{calc.}$$

$\alpha \backslash \beta$	$^{238}U(n,f)$ $\bar{\sigma}(> E_p) = 0.532$ b	$\alpha \backslash \beta$	$^{238}U(n,f)$ $\bar{\sigma}(> E_p) = 0.532$ b
Fission		Threshold	
^{239}Pu	3.38	$^{115}In(n,n')$	0.496
^{235}U	2.295	$^{47}Ti(n,p)$	0.0915
^{233}U	3.56	$^{32}S(n,p)$	0.372
^{238}U	1.000	$^{58}Ni(n,p)$	0.497
^{237}Np	2.97	$^{54}Fe(n,p)$	0.439
^{240}Pu	2.98	$^{46}Ti(n,p)$	0.1515
		$^{56}Fe(n,p)$	0.0461
Additions		$^{63}Cu(n,\alpha)$	0.01466
$^{130}Rh(n,n')$	1.570	$^{48}Ti(n,p)$	0.01731
$^{93}Nb(n,n')$	0.406	$^{27}Al(n,\alpha)$	0.0665

TABLE X-16. OBSERVED INTEGRAL CROSS SECTIONS

Spectrum: ^{235}U Thermal-Neutron-Induced Fission Neutrons Designation: XU5

Entry Date: March, 1982

Revised:

Reaction	Median Response Energy	Cross Section ($10^{-27}cm^2$)	Measurement Reference
	Threshold Reactions		
$^{238}u(n,f)^{\dagger}$	2.68	309 ± 2.6%	F10, R9, R17, R28
$^{237}Np(n,f)^{\dagger}$	1.96	1344 ± 4.0%	F10, R9, R17, R28
$^{93}Nb(n,n')$	2.4	164 ± 9 %	R28a, R29a
$^{115}In(n,n')^{\dagger}$	2.6	192 ± 3.7%	F10
$^{47}Ti(n,p)$	3.8	17.2 ± 6 %	F15
$^{58}Ni(n,p)$	4.1	109 ± 2.8%	R24, F11, R33
$^{54}Fe(n,p')$	4.2	80.5 ± 3.5%	R24, R33
$^{46}Ti(n,p)$	5.7	11.5 ± 7 %	F15, R24, R34
$^{63}Cu(n,\alpha)$	7.3	0.58 ± 7 %	R24, F15
$^{27}Al(n,\alpha)$	8.4	0.708 ± 4.0%	R24
$^{48}Ti(N,p)$	8.1	0.271 ± 7 %	F15
	Non-Threshold Reactions		
$^{239}Pu(n,f)^{\dagger}$	1.65	1832 ± 3.0%	F10, R9, R17, R28
$^{235}U(n,f)^{\dagger}$	1.57	[1216 ± 1.7%]	R9, R17, R28

†Ratio measurements with double fission chambers in the SCK/CEN cavity fission source (1 meter dia. cavity in graphite). The flux monitor was $^{235}U(n,f)$ for which the Cf fission spectrum observed value is (1216 ± 1.6%)mb--see Part IA. The ^{235}U fission cross section ratio between Cf and ^{235}U fission spectra is 1.000 based on spectrum evaluation (Section 4.b.3). A conservative upper bound for the error in this ratio is ±0.6%.

TABLE X-17(B5). CALCULATED-TO-OBSERVED RATIOS OF CROSS SECTIONS

Spectrum: ^{235}U Thermal-Neutron-Induced Fission (NBS-Evaluated, F1, F2)

Designation: XU5-5-N1

Entry Date: March, 1982

Cross Section: ENDF/BV Dosimetry File

Revised:

REACTION	Observed	Calculated		Calculated-to-Observed	
	(Table X-16)	NBS-eval.[(a)] fiss. spec. TABLE X-13(B5)	ENDF/B-V fiss. spec. TABLE X-14a(B5)	NBS-eval. fiss. spec.	ENDV/B-V fiss. spec.
	(mb)	(mb)	(mb)		
Fission					
$^{239}Pu(n,f)$	1832 ± 3.0%	1786 ± 0.2%	1791	0.975 ± 0.3%	0.978
$^{235}U(n,f)^{\dagger}$	[1216 ± 1.6%]	1236 ± 0.2%	1236	[1.017 ± 1.8%	1.017]
$^{238}U(n,f)$	309 ± 2.6%	294.7 ± 1.5%	305	0.953 ± 3.0%	0.988
$^{237}Np(n,f)$	1344 ± 4.0%	1322 ± 1.0%	1347	0.984 ± 4.1%	1.002
Threshold					
$^{93}Nb(n,n')$	164 ± 9 %	154.3 ± 1.4%	159.7	0.941 ± 9.1%	0.974
$^{115}In(n,n')$	192 ± 3.7%	173 ± 1.4%	179	0.90 ± 4.0%	0.93
$^{47}Ti(n,p)$	17.2 ± 6 %	21.6 ± 2.2%	22.5	1.26 ± 6.4%	1.31
$^{58}Ni(n,p)$	109. ± 2.8%	101 ± 2.6%	105	0.93 ± 3.8%	0.96
$^{54}Fe(n,p)$	80.5 3.5%	77.8 ± 2.8%	81	0.97 ± 4.5%	1.006
$^{46}Ti(n,p)$	11.5 ± 7 %	10.8 ± 3.8%	11.2	0.94 ± 8.0%	0.97
$^{63}Cu(n,\alpha)$	0.58 ± 7 %	0.540 ± 3.4%	0.558	0.93 ± 7.8%	0.96
$^{27}Al(n,\alpha)$	0.708 ± 4.0%	0.693 ± 3.6%	0.719	0.98 ± 5.4%	1.016
$^{48}Ti(n,p)$	0.271 ± 7 %	0.273 ± 3.3%	0.282	1.007 ± 7.7%	1.041

(a) Errors correspond to ^{235}U fission spectrum uncertainties given in Table X-5 and propagated according to the second term of Eq. X-7, Sect. 5.a.3., Part IA.

† See footnote, Table X-16.

TABLE X-18. OBSERVED SPECTRAL INDEXES

Spectrum: ^{235}U Thermal-Neutron-Induced Fission Neutrons

Designation: XU5

Entry Date: May, 1978

Revised: March, 1982

$$\left[S_{\alpha/U8}\right]_{obs} = \left[\bar{\sigma}_{\alpha}/\bar{\sigma}_{f}(U238)\right]_{obs}$$

Reaction[(a)]	Reaction[(b)] Non-overlap Interval (MeV)	Spectral[(c)] Index (1σ)	Measurement Reference
Threshold Reactions			
^{237}Np(n,f)/U8†	0.67 - 1.5	4.35 ± 3.0%	F10 (1975)
^{115}In(n,n')/U8†	1.2 - 1.5	0.620 ± 3.1%	F10 (1975)
^{47}Ti(n,p)/U8*	1.9 - 1.5	0.0557 ± 6 %	F15 (1976)
^{58}Ni(n,p)/U8	2.1 - 1.5	0.353 ± 3.8%	R33 (1982), R24
^{54}Fe(n,p)/U8	2.5 - 1.5	0.261 ± 4.4%	R33 (1982), R24
^{46}Ti(n,p)/U8*	3.8 - 1.5	0.0373 ± 7 %	F15 (1976)
^{63}Cu(n,α)/U8	4.7 - 1.5	1.88E-3 ± 8 %	R24 (1975), F15
^{48}Ti(n,p)/U8*	5.9 - 1.5	0.88E-3 ± 7 %	F15 (1976)
^{27}Aℓ(n,α)/U8	6.5 - 1.5	2.29E-3 ± 4.8%	R24 (1975), R22
Non-Threshold Reactions			
^{239}Pu(n,f)/U8†	0.26 - 1.5	5.93 ± 2.2%	F10 (1957)
^{235}U(n,f)/U8†	0.20 - 1.5	3.94 ± 2.0%	F10 (1975)

Footnotes for TABLE X-18

(a) All spectral indexes are taken relative to the U238(n,f) reaction labeled "U8".

(b) Non-overlapping energy response interval between the 95% response energy boundary for each detector. See Table X-13(B5) and footnote (b) on Table X-9.

(c) Indexes for independent cross section measurements are obtained from Table X-16.

†Ratio measurements obtained with double fission chamber at the SCK/CEN cavity fission source (1 m diameter cavity in graphite).

*Titanium cross sections obtained with a thermal-beam driven ^{235}U converter. Data renormalized to $\bar{\sigma}(In(n,n'))$ = 192±3.7%.

TABLE X-19(B5). CALCULATED-TO-OBSERVED RATIOS OF SPECTRAL INDEXES - NBS EVAL.

Spectrum: ^{235}U Thermal-Neutron-Induced Fission (NBS evaluation, F1, F2)

Designation: XU5-5N1

Entry Date: May, 1978

Spectral Index: Calculated: Table X-14(B5)
Observed: Table X-18

Revised: March, 1982

$$C_{\alpha/U8} = \left[S_{\alpha/U8}\right]_{calc.} \Big/ \left[S_{\alpha/U8}\right]_{obs.} \quad \text{or} \quad C_{U8/\alpha}$$

Reaction[a]	Reaction[b] Non-overlap Interval (MeV)	Calculated-To-Observed Values	E R R O R S (1σ) Observed Value	Calculated[c] Value	Total
Threshold Reactions					
U8/Np(n,f)	0.7 - 1.5	0.969	± 3.0%	± 1.0%	± 3.2%
U8/In(n,n')	1.2 - 1.5	1.054	± 3.1%	± 0.3%	± 3.1%
$^{47}Ti(n,p)/U8$	1.5 - 1.9	1.32	± 6 %	± 1.1%	± 6.1%
$^{58}Ni(n,p)/U8$	1.5 - 2.1	0.969	± 3.8%	± 1.7%	± 4.2%
$^{54}Fe(n,p)/U8$	1.5 - 2.5	1.011	± 4.4%	± 1.9%	± 4.8%
$^{46}Ti(n,p)/U8$	1.5 - 3.8	0.984	± 7 %	-	-
$^{63}Cu(n,\alpha)/U8$	1.5 - 4.7	0.974	± 8 %	-	-
$^{48}Ti(n,p)/U8$	1.5 - 5.9	1.051	± 7 %	± 2.8%	± 7.5%
$^{27}Al(n,\alpha)/U8$	1.5 - 6.5	1.027	± 4.8%	± 2.8%	± 5.5%
Non-Threshold Reactions					
$U8/^{239}Pu(n,f)$	0.3 - 1.5 low 5.2 - 6.7 high	0.979	± 2.2%	± 1.4%	± 2.6%
$U8/^{235}U(n,f)$	0.2 - 1.5 low 5.2 - 6.7 high	0.940	± 2.0%	± 1.7%	± 2.6%

(a) Spectral indexes are taken with respect to the $^{238}U(n,f)$ reaction labeled "U8" with the lower energy response detector in the denominator. The calculated-to-observed ratio for σ_f(U238) is 0.953 (NBS-eval.) from Table X-17(B5).

(b) Non-overlapping energy response interval between the 95% response energy boundary for each detector. For non-threshold reactions, two non-overlapping energy intervals are listed: The 95% exclusion interval is labeled "low" and the 5% exclusion interval "high". See Table X-13.

(c) Errors in the calculated spectral indexes are for ^{235}U fission spectrum uncertainties only as given in Table X-5 and propagated according to the second term of Eq. X-8, Part IA, sect. 5.a.3. The error in σ_f(U238) calculated is ±1.5% from Table X-17(B5).

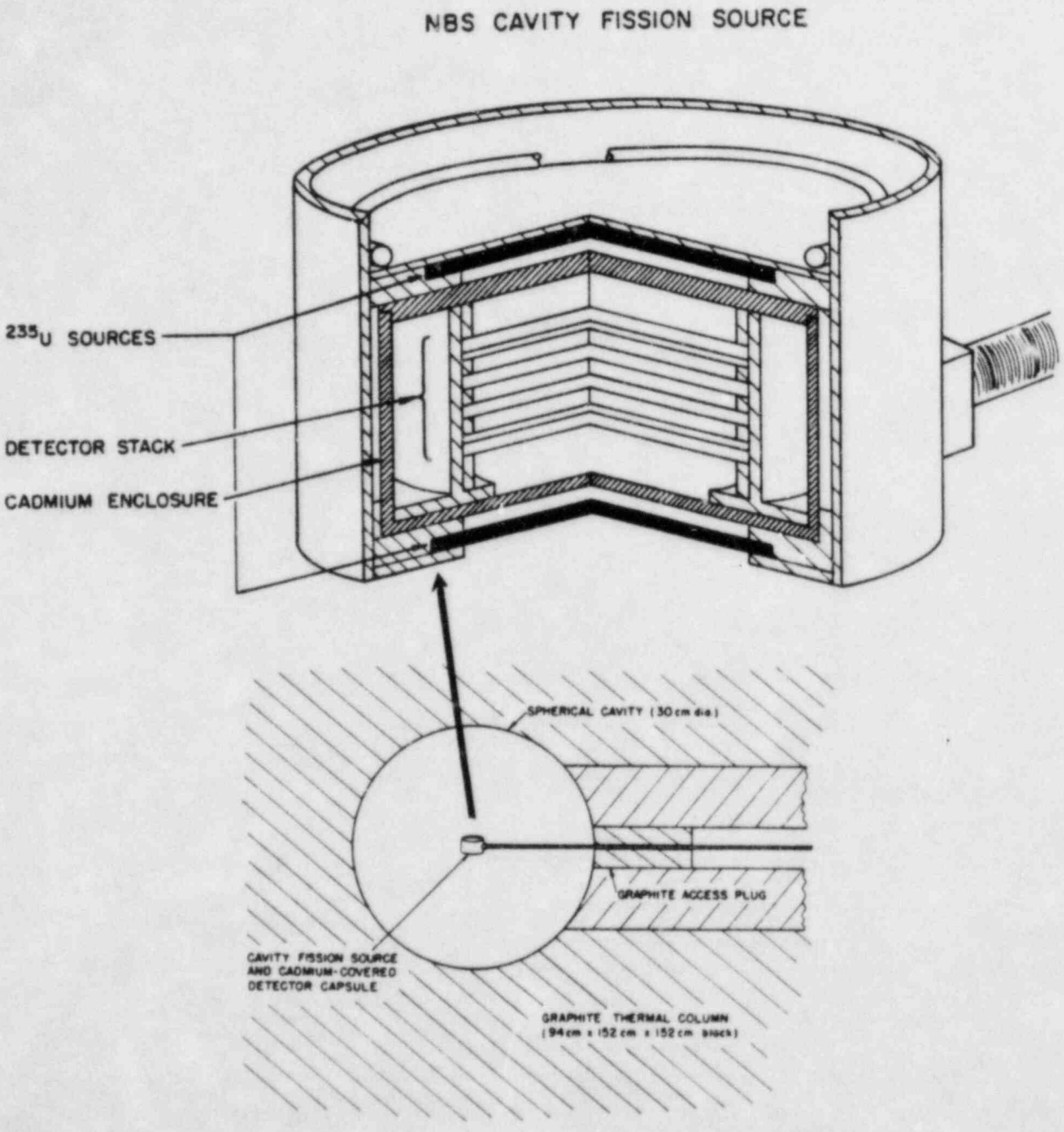

FIGURE X-3. NBS cavity fission source and detector assembly for fission neutron irradiations in the NBS reactor thermal column. Two source disks of 16 mm diameter are mounted in a capsule at a separation distance of 11 mm.

COMPENDIUM REFERENCES

SECTION 1. INTEGRAL DETECTOR RESPONSE

(R1) Mannhart, W. and Alberts, W. G., "Measurement and Calculation of Average Activation Cross Section in the Spontaneous Fission Neutron Field of ^{252}Cf," Nucl. Sci. and Eng., 69, 333 (1979).

(R2) Alberts, W. G., Bortfeldt, J., Günther, E., Knauf, K., Matzke, M., Rassl, G., Siegel, V., and Walz, K. F., "Measurement of Cross Sections for Threshold Reaction Induced by ^{252}Cf Spontaneous Fission Neutrons," Proceedings Nuclear Cross Sections and Technology Conference, NBS Special Publication 425, Vol. I, p. 273 (1975).

(R3) Alberts, W. G., Günther, E., Matzke, M., Rassl, G., "Measurements of Integral Cross Sections in the ^{252}Cf Fission Spectrum," Proc. of First ASTM-EURATOM Symposium on Reactor Dosimetry, EUR 5667 e/f, Part II, p. 131, Petten, (September 22-26, 1975).

(R4) Green, L., "Absorption Cross Section Measurements for ^{252}Cf Spontaneous Fission Neutron," Nucl. Sci. Eng. 58, 361 (1975).

(R5) Brandon, W. E., Cogburn, C. O., Culp, R. R., Sallee, W. W., and Williams, J. G., "Wall Return of ^{252}Cf Neutrons at the SEFOR Calibration Center," Trans. Am. Nucl. Soc. 38, 579 (June, 1981).

(R6) Brandon, W. E., Cogburn, C. O., Culp, R. R., Sallee, W. W., and Williams, J. G., "Calibration of Neutron Dosimetry Techniques For Use in Reactor Buildings," Proc. Eighth DOE Workshop on Personnel Neutron Dosimetry, PNL-SA-9950, Louisville, Kentucky (June, 1981).

(R7) Grundl, J. A., Eisenhauer, C. M., and McGarry, E. D. "Neutron Flux Measurements in the Pressure Vessel Cavity of an Operating U.S. Power Reactor," LWR Pressure Vessel Irradiation Surveillance Dosimetry Progress Report, HEDL-Time 8218, NUREG/CR-2805 Vol. I. (March, 1982).

(R8) Gilliam, D. M., "Integral Measurement Results in Standard Fields," Proceedings International Symposium on Neutron Standards and Applications, NBS Special Publ. 493, U.S. Dept. of Commerce, Washington, D.C. (March, 1977).

(R9) Heaton II, H. T., Gilliam, D. M., Spiegel, V., Eisenhauer, C., and Grundl, J. A., "Fission Cross Sections of ^{235}U, ^{238}U, and ^{239}Pu Averaged Over the ^{252}Cf Fission Neutron Spectron," Proceedings of the NEANDC/NEACRP Specialist Meeting on Fast Neutron Fission Cross Sections of U-233, U-235, U-238, and Pu-239, Argonne, Illinois. ANL-76-90, ERDA-NDC-5/L, NEANDC(US)-199/L. p. 333, June 1976.

(R10) Gilliam, D. M., Eisenhauer, C., Heaton II, H. T., and Grundl, J. A., "Fission Cross Section Ratios in the ^{252}Cf Neutron Spectrum (^{235}U, ^{238}U, ^{239}Pu, ^{237}Np)," Proceedings of a Conference on Nuclear Cross Sections and Technology, Washington, D.C. NBS Special Publication 425, Vol. I, March 1975. p. 270.

(R11) Heaton II, H. T., Grundl, J. A., Spiegel Jr., V., Gilliam, D. M., and Eisenhauer, C., "Absolute ^{235}U Fission Cross Section for ^{252}Cf Spontaneous Fission Neutrons," Proceedings of a Conference on Nuclear Cross Sections and Technology, March 1975, Washington, D.C., NBS Special Publication 425.

(R12) Spiegel, V., Eisenhauer, C. M., and Grundl, J. A. (NBS), and Martin Jr., G. C. (GE), "Reaction Rate Measurements and Integral Cross Sections Using the NBS ^{252}Cf Fission Neutron Indoor Irradiation Facility," Proc. Second ASTM-EURATOM Symposium on Reactor Dosimetry, Palo Alto, CA (October, 1977).

(R13) Fleming, R., and Spiegel, V., "Measurement of Threshold Reaction Cross Section Ratios in Fission Neutron Fields," Proc. Second ASTM-EURATOM Symposium on Reactor Dosimetry, Palo Alto, CA (October, 1977).

(R14) Mannhart, W., "Integral Cross Section Measurement With Regard to the Low and High Energy Part of the ^{252}Cf Neutron Spectrum," IAEA Consultants Meeting on Neutron Cross Sections for Reactor Dosimetry, Vienna (November 1976), IAEA Proceedings (1978).

(R15) Wagschal, J. J.. Maerker, R. E., ORNL; and Gilliam, D. M., NBS; "Detailed Error Analysis of Average Fission Cross-Section Measurements in NBS Standard Neutron Fields," Trans. Am. Nucl. Soc. 33, 823 (November, 1979).

(R16) Mannhart, W., "Progress In Integral Data and Their Accuracy: Average Neutron Cross Sections in the Californium-252 Benchmark Field," Nucl. Sci. and Eng., 77, 40 (1981).

(R17) Grundl, J., Gilliam, D., McGarry, E. D., Eisenhauer, C., Soran, P., "Revised Value for the ^{252}Cf Fission-Spectrum-Averaged Cross Section for ^{235}U Fission," private communication to CSEWG (May, 1982).

(R18) Winkler, C., Spiegel, V., Eisenhauer, C. M., Smith, D. L., "Measurement of the Average Activation Cross Section for the Reaction $^{63}Cu(n,\alpha)^{60}Co$ in the Spontaneous Fission Neutron Field of Californium-252," Nucl. Sci. and Eng., 78, (1981).

(R19) Mannhart, W., "Average Neutron Cross Sections in the Cf-252 Benchmark Field," Proc. Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Washington, D.C. (March, 1982).

(R20) Grundl, J., and Gilliam, D. "Fission Cross Section Ratio Measurements in Reactor Physics and Dosimetry Benchmarks," Trans. Am. Nucl. Soc., 44, June (1983).

(R21) Adamov, V. M., Alexandrov, B. M., Alkhazov, I. D., Drapchinsky, L. V., Kovalenko, S. S., Kostochkin, O. I., Kudriavzev, G. Yu., Malkin, L. Z., Petrzhak, K. A., Pleskachevsky, L. A., Fomichev, A. V., and Shapakov, V. I., "Absolute ^{235}U, ^{238}U, ^{237}Np Fast Neutron Fission Cross-Section Measurements, Proc. Symp. on Neutron Standards and Applications, March 1977, Washington, D.C., NBS Special Publication 493.

(R22) Davis, M. C., and Knoll, G. F., "Fission Cross Sections of ^{235}U and ^{239}Pu Averaged Over ^{252}Cf Neutron Spectrum," Ann. Nucl. Energy 5, 583 (1978).

(R23) Fabry, A., CEN/SCK, Mol, Belgium; McElroy, W. N., Kellogg, L. S., Lippincott, E. P., HEDL; Grundl, J. A., Gilliam, D. M., NBS; Hansen, G. E., LASL; "Review of Microscopic Intergral Cross Section Data in Fundamental Reactor Dosimetry Benchmark Neutron Fields," Consultants' Meeting on Integral Cross-Section Measurements in Standard Neutron Fields; International Atomic Energy Agency, Vienna, November 15-19, 1976. IAEA-208, Neutron Cross Sections for Reactor Dosimetry (1978).

(R24) Fabry, A., Ceulemans, H., Vandeplas, P., McElroy, W. N., and Lippincott, E. P., "Reactor Dosimetry Integral Reaction Rate Data," Proceedings of the First International ASTM-EURATOM Symposium on Reactor Dosimetry, Proc. EUR 5667 e/f Supplement, p. 23, (September 22-26, 1975).

(R25) ENDF/B-V Dosimetry File, National Neutron Cross Section Center; Brookhaven National Laboratory. See also Magurno, B. A., "ENDF/B-IV Dosimetry File," Report BNL-NCS-50446.

(R25a) Cullen, D. E., N. Kucherov, P. M. McLaughlin, "International Reactor Dosimetry File-82," IAEA-NDS-41/R. See also B. Strohmaier, S. Tagesen, H. Vonach, Physics Datem-Physics Data 13 (1980).

(R26) Paulsen, A., Widera, R., Vaninbrouckx, R., and Liskien, H., "Cross-Section Measurement for the Reaction $^{103}Rh(n,n')$," Nucl. Sci. and Eng., 76, 331-335 (1980).

(R27) 1983 Annual Book of ASTM Standards, Part 45 Nuclear Standards; American Society for Testing and Materials, Philadelphia, PA (1981).

(R28) Spiegel, V., Eisenhauer, C. M., Gilliam, D. M., Grundl, J. A., McGarry, E. D., Schröder, I., Slater, W. E., and Schwartz, R. S., "^{235}U Cavity Fission Neutron Field Calibration via the ^{252}Cf Spontaneous Fission Neutron Field," IAEA Consultant's Mtg. on Neutron Source Properties, Debrecen, Hungary (March, 1980).

(R28a) Alberts, W. G., Hollnagel, R., Knauf, K., Matzke, M. Pebara, W., "Measurements with the Niobium Neutron Fluence Detector at the PTB," Proc. Fourth ASTM-EURATOM Symp. on Reactor Dosimetry, CONF-820321/V1 (1982).

(R29) Acquah, I., Glumac, B., Remec, I., Najzer, M., "Spectral Indices of Some Threshold Reactions Measured in Uranium-235 Fission Spectrum," Proc. Advisory Group Meeting on Nuclear Data for Radiation Damage Assessment and Related Safety Aspects, IAEA-TECDOC-263, Vienna (Oct. 1981).

(R29a) Kobayashi, K., Kimura, I., "Fission Spectrum Averaged Cross Sections with Standard Neutron Fields," Proc. Third ASTM-EURATOM Symp. on Reactor Dosimetry (Ispra, 1979), ISBN 92-825-1903-1; Commission of European Communities, EUR6813, Luxembourg (1980).

(R30) Hannan, A., Williams, J. G., "Activation Foil Data for NISUS, Mol-ΣΣ, and ^{235}U Fission Spectra," IAEA Consultants Meeting on Neutron Cross Sections for Reactor Dosimetry, Vienna (Nov. 1976), IAEA Proceedings (1978).

(R31) Fabry, A., "Evaluation of Microscopic Integral Cross Section Averaged in a ^{235}U Fission Neutron Spectrum," Report Blg-465, CEN/SCK, Mol (1972).

(R32) Grundl, J. A., "Fission-Neutron Spectra: Macroscopic and Integral Results," Proc. Symp. on Neutron Standards and Flux Normalization, ANL, CONF-701002 (October, 1970).

(R33) Lamaze, G. P., McGarry, E.D., and Schima, F. J., "Integral Reaction Rate Measurements in ^{252}Cf and ^{235}U Fission Spectra," Proc. Int.Conf. on Nuclear Data for Science and Technology, Antwerp, Belgium (September, 1982).

(R34) Bresesti, A., Bresesti, M., Rota, A., Rydin, R., Threshold Reaction Excitation Functions Intercalibrated in a Pure Fission Spectrum," Nucl. Sci. Eng., 40, p. 331 (1970).

SECTION 2. FACILITY DESCRIPTION AND FIELD CHARACTERIZATION

(F1) Grundl, J. and Eisenhauer, C., "Fission Rate Measurements for Materials Neutron Dosimetry in Reactor Environments," Proceedings First ASTM-EURATOM Symposium on Reactor Dosimetry, EUR5667 e/f, Commission of the European Committees, Petten (September, 1975).

(F2) Grundl, J. A. and Eisenhauer, C. M., "Fission Spectrum Neutrons for Cross Section Validation and Neutron Flux Transfer," Proceedings of a Conference on Nuclear Cross Sections and Technology, NBS Special Publication 425, U.S. Dept. of Commerce, Washington, D.C. (March, 1975).

(F3) Grundl, J. A., Spiegel, V., Eisenhauer, C. M., Heaton II, H. T., Gilliam, D. M. NBS; and Bigelow, J., ORNL; "A Californium-252 Fission Spectrum Irradiation Facility for Neutron Reaction Rate Measurements," Nucl. Tech. 32, 315 (March, 1977).

(F4) Wiedling, T., Proc. Symp. on Neutron Standards and Flux Normalization, ANL (1970). Smith, A. B., Proc. Consultants Meeting on Prompt Fission Neutron Spectra, Vienna (1971). Koster, A., Ibid., Vienna (1971).

(F5) Green, L., et al., Nucl. Sci. Eng., 50, 257 (1973). Knitter, H., et al., Atomkernenergie, Bd. 22, p. 84 (1973). Meyer, W., et al., Trans. Am. Nucl. Soc., 19, 480 (1974). Jeki, L., et al., Proc. Meeting on Prompt Fission Neutron Spectra, Vienna (1971).

(F6) Werle, H. and Bluhn, H., J. Nucl. Energy 26, 165 (1972). Meadows, J., Phys. Rev. 157, 1076 (1967). Condé, H., and During, G., Arkiv for Fysik, 29, 313 (1965). Yu. S. Zamyatin, et al., Nucl. Data for Reactors (Proc. Conf.), Helsinki (1970).

(F7) Cranberg, L., Frye, G., Nereson, N., and Rosen, L., Phys. Rev. 103, 662 (1956). Barnard, E., et al., Nucl. Phys. 71, 228 (1965). Watt, R., Phys. Rev. 87, 1037 (1952).

(F8) Islam, N. and Knitter, H., Nucl. Sci. Eng. 50, 108 (1973). Johansson, P., et al., Nucl. Data for Reactors (Proc. Conf.), Helsinki (1970); see also Ref. (F2).

(F9) A. Fabry, G. Minsart, F. Cops, and S. DeLeeuw, "The Mol Cavity Fission Spectrum Standard Neutron Field and Its Applications," Proc. Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Washington, D.C. (March, 1982).

(F10) Fabry, A., (CEN/SCK, Mol, Belgium); Grundl, J. A., and Eisenhauer, C., (NBS, USA); "Fundamental Integral Cross Section Measurements in the Thermal-Neutron-Induced Uranium-235 Fission Neutron Spectrum," Proceedings of a Conference on Nuclear Cross Sections and Technology, NBS Special Publications 425, U.S. Dept. of Commerce, Washington, D.C., p. 254 (March, 1975).

(F11) Fabry, A., and Czock, "Measurements of Integral Cross Section Ratios in Two Dosimetry Benchmark Neutron Fields," INCD(IAEA)-005/G, IAEA/RL/27, IAEA Publ. (1974).

(F12) E. D. McGarry, G. P. Lamaze, C. M. Eisenhauer, D. M. Gilliam, and F. J. Schima, "NBS ISNF and Cavity Fission U-235 Standard Neutron Fields," Proc. Fourth ASTM-EURATOM Symposium on Reactor Dosimetry, Washington, D.C. (March, 1982).

(F13) Grundl, J. A., "Measurement of the Average Fission Cross-Section $\bar{\sigma}_f$ (^{238}U) for ^{235}U and ^{239}Pu Fission Neutrons," IAEA Consultants Meetings on Prompt Fission Neutron Spectra, Vienna, Austria (August 1971). IAEA Proceedings (1972).

(F14) Fabry, A. and Jenkins, J. D., "Wall Return Neutron Fluxes for High and Intermediate-Energy Cavity Neutron Sources," Trans. Am. Nucl. Soc. 15, 2 (1972).

(F15) Kimura, I., Kobayashi, K., Hayashi, Shu A., Yamamoto, S., Gotoh, H., and Yagi, H., "Measurement and Evaluation of Threshold Reaction Cross Section in Standard Neutron Fields," IAEA Consultants Meeting on Neutron Cross Sections for Reactor Dosimetry, Vienna (Nov. 1976) IAEA Proceedings, Vol. II, p. 265 (1978).

(F16) Eisenhauer, C. M. and Grundl, J. A., "Neutron Transport Calculations for the Intermediate-Energy Standard Neutron Field (ISNF) at the National Bureau of Standards," Proceedings International Symposium on Neutron Standards and Applications, NBS Special Publication 493, U. S. Dept. of Commerce, Washington, D.C. (March, 1977).

(F17) Fabry, A., DeLeeuw, G., and DeLeeuw, S., CEN/SCK, Mol, Belgium, "The Secondary Intermediate-Energy Standard Neutron Field at the Mol-ΣΣ Facility," Nucl. Tech. 25, 349 (February, 1975).

(F18) Grundl, J., "Cavity Fission Spectra," Trans. Am. Nucl. Soc. 5, 381 (1962).

(F19) D. G. Madland and J. R. Nix, "New Calculation of Prompt Fission Neutron Spectra and Average Prompt Neutron Multiplicities," Nucl. Sci. and Eng. 81, 213-271 (1982).

DISTRIBUTION

R5

DOE-HQ/Office of Converter
Reactor Deployment
Nuclear Regulation & Safety Division
NE-12
Washington, DC 20545

JD Griffith, Deputy Director

DOE-HQ/Office of Breeder
Technology Projects (8)
NE-53
Washington, DC 20545

H. Alter, Asst Director, Safety
PB Hemmig, Asst Director,
Reactor Physics Technology
JW Lewellen, Manager,
Core Analysis Technology
DK Magnus, Director,
Fuels & Core Materials
WA Nelson, Director, Office of
Breeder Technology Projects
RJ Neuhold, Director,
Safety & Physics
CM Purdy, Asst Director,
Materials & Structures Tech
A. Van Echo, Manager, Metallurgy,
Absorbers & Standards

DOE-RL/AME
Breeder Technology Division
Technology Development Branch
P.O. Box 550, FED/242
Richland, WA 99352

KR Absher, Chief

Arizona State University (2)
College of Eng & Appl Sciences
Tempe, AZ 85287

B. Stewart
JW McKlveen

Argonne National Laboratory (2)
9700 South Cass Avenue
Argonne, IL 60439

RJ Armani
RR Heinrich, Bldg 316

Babcock & Wilcox Co.
Lynchburg Research Center (4)
P.O. Box 1260
Lynchburg, VA 24505

RH Lewis AA Lowe Jr
LB Gross CL Whitmarsh

Battelle
Pacific Northwest Laboratory
P.O. Box 999
Richland, WA 99352

EP Simonen

Battelle Memorial Institute
505 King Avenue
Columbus, OH 43201

MP Manahan

Bechtel Power Corporation
15740 Shady Grove Road
Gaithersburg, MD 20760

WC Hopkins

Brookhaven National Laboratory (3)
National Neutron Cross-Section Center
Upton, Long Island, NY 11973

JF Carew BA Magurno
S. Pearlstein, Bldg T-197

Burns & Roe, Inc.
633 Industrial Avenue
Paramus, NJ 07672

C. Celnik

DISTRIBUTION (Cont'd)

Carolina Power & Light Co.
P.O. Box 1551
Raleigh, SC 27602

S. Grant

Centre d'Etude de l'Energie Nucléaire
Studiecentrum voor Kernenergie (6)
Boeretang 200
B-2400 Mol, Belgium

J. Debrue
G. DeLeeuw
S. DeLeeuw
A. Fabry
G. Minsart
PH Van Asbroeck

Combustion Engineering Inc. (5)
1000 Prospect Hill Road
Windsor, CT 06095

S. Byrne
G. Cavanaugh
JJ Koziol
RG Shimko
D. Stephen

Comitato Nazional per Energia Nucleare
Centro di Studi Nucleari della Casaccia
Casella Postale 2400
Santa Maria di Galeria
I-00060 Rome, Italy

U. Farinelli

Commissariat a l'Energie Atomique
Centre d'Etudes Nucleaires de Saclay (6)
Boite Postale 2
91190 Gif-sur-Yvette, France

AA Alberman
C. Buchalet (Framatome)
JM Cerles
JP Genthon
P. Mas (Grenoble)
P. Petrequin

Commonwealth Edison
P.O. Box 767
Chicago, IL 60690

E. Steeve

EG&G Idaho, Inc. (3)
P.O. Box 1625
Idaho Falls, ID 83415

R. Greenwood
Y. Harker
JW Rogers

EG&G Ortec, Inc.
100 Midland Road
Oak Ridge, TN 37830

W. Zimmer

Electric Power Research Institute (7)
3412 Hillview Avenue
P.O. Box 10412
Palo Alto, CA 94304

TU Marston
O. Ozer
Y. Passell
AD Rossin
R. Shaw
K. Stahlkopf
J. Taylor

Energieonderzoek Centrum Nederland
Westerdijnweg 3
NL-1755 ZG, Petten, Netherlands

WL Zijp

Engineering Services Associates
3320 Bailey
Buffalo, NY 14215

M. Haas

EURATOM
Joint Research Center Ispra (2)
I-21020 Ispra, Varese, Italy

R. Dierckx
H. Rief

DISTRIBUTION (Cont'd)

Fracture Control Corporation
5951 Encina Road, No. 105
Goleta, CA 93117

P. McConnell

Fracture Control Corporation
2041 Willowick Drive
Columbus, OH 43229

JS Perrin

GE/Vallecitos Nuclear Center
P.O. Box 460
Pleasanton, CA 94566

GC Martin

IKE-Stuttgart (2)
Pfaffenwaldring 31
Postfach 801140
D-7000, Stuttgart 80 (Vaihingen)
Federal Republic of Germany

G. Hehn
G. Prillinger

International Atomic Energy Agency (2)
Wagramerstrasse 5
Postfach 100
A-1400 Vienna, Austria

A. Sinev
JJ Schmidt

IRT Corporation (3)
P.O. Box 80817
San Diego, CA 92183

NA Lurie WE Selph
C. Preskitt

Italian Atomic Power Authority
National Electric Energy Agency (2)
Viale Regina Margherita 137
Rome, Italy

M. Galliani
F. Remondino

Japan Atomic Energy Research Institute (2)
Tokai-mura, Naka-gun
Ibaraki-ken, Japan

S. Mizazono
K. Sakurai

Kernforschungsanlage Jülich GmbH (3)
Postfach 1913
D-517 Jülich 1,
Federal Republic of Germany

D. Pachur L. Weise
W. Schneider

Kraftwerk Union Aktiengesellschaft (5)
Postfach 3220
D-8520 Erlangen,
Federal Republic of Germany

A. Gerscha J. Koban
U. Groschel C. Leitz
W. Hofmann

Los Alamos National Laboratory (2)
P.O. Box 1663
Los Alamos, NM 87545

GE Hansen, Group N-2
L. Stewart

Maine Yankee Atomic Power Co.
Edison Drive
Augusta, MA 04336

HF Jones Jr

DISTRIBUTION (Cont'd)

Materials Engineering Associates
111 Mel-Mara Drive
Oxen Hill, MD 20021

JR Hawthorne

Max-Planck-Institüt
für Plasma Physik
The NET Team
D-8046 Garching bei Müchen,
Federal Republic of Germany

DR Harries, Technology

National Bureau of Standards
Center of Radiation Research (6)
Washington, DC 20234

RS Caswell — JA Grundl
CM Eisenhauer — G. Lamaze
DM Gilliam — ED McGarry

Naval Research Laboratory
Engineering Materials Division
Thermostructural Materials Branch
Code 6390
Washington, DC 20375

LE Steele

Nuclear Regulatory Commission (17)
Office of Nuclear Regulatory Research
Division of Engineering Technology
Materials Engineering Branch
NL-5650
Washington, DC 20555

Chief — L. Lois
Public Doc Rm (3) — S. Pawlicki
M. Bolotski — PN Randall
M. Dunenseld — CZ Serpan
R. Gamble — D. Sieno
W. Hazelton — A. Taboada
KG Hoge — M. Vagin
RE Johnson

Oak Ridge National Laboratory (8)
P.O. Box X
Oak Ridge, TN 37830

CA Baldwin — RE Maerker
RG Berggren — LS Miller
FBK Kam — R. Nanstad
AL Lotts — FW Stallmann

Radiation Research Associates (2)
3550 Hulen Street
Fort Worth, TX 76107

RM Rubin
MB Wells

Rockwell International
Energy Systems Group (2)
P.O. Box 309
Canoga Park, CA 91304

H. Farrar IV
B. Oliver

Rolls-Royce & Associates Ltd. (4)
P.O. Box 31
Derby, UK

M. Austin — AF Thomas
P. Burch — TJ Williams

S.A. Cockerill-Ougree
Recherches et Developments
Division de la Construction Mecanique
B-4100 Seraing, Belgium

J. Widart

Science Applications Inc. (3)
P.O. Box 2325
La Jolla, CA 92037

W. Hagan — VV Verbinski
GL Simmons

DISTRIBUTION (Cont'd)

Ship Research Institute
Tokai Branch Office
Tokai-mura, Naka-gun
Ibaraki-ken, Japan

K. Takeuchi

Southwest Research Institute
8500 Calebra Road
P.O. Box 28510
San Antonio, TX 78284

EB Norris

Swiss Federal Institute
for Reactor Research
CH-5303 Würenlingen, Switzerland

F. Hegedus

United Kingdom Atomic Energy Authority
Atomic Energy Research Establishment (2)
Harwell, Oxon, UK

LM Davies
AJ Fudge

United Kingdom Atomic Energy Authority
Atomic Energy Establishment (4)
Winfrith, Dorchester, Dorset, UK

J. Butler
CG Campbell
AK McCracken
J. Sanders

University of Arkansas (2)
Dept of Mechanical Engineering
Fayetteville, AR 72701

CO Cogburn
L. West

University of California
at Santa Barbara (2)
Dept of Chem & Nucl Engineering
Santa Barbara, CA 93106

G. Lucas
GR Odette

Univ of London Reactor Center
Silwood Park, Sunnyhill
Ascot, Berkshire, SL57PY, UK

JA Mason

University of Tokyo (2)
Dept of Nuclear Engineering
7-3-1, Hongo
Bunkyo-ku, Tokyo, 113 Japan

M. Nakazawa
J. Sekiguchi

Westinghouse
Nuclear Energy Systems (4)
P.O. Box 355
Pittsburgh, PA 15230

SL Anderson
TR Mager
RC Shank
SE Yanichko

Westinghouse
Research and Development Center
1310 Beulah Road
Pittsburgh, PA 15235

JA Spitznagel

University of Missouri (2)
at Rolla
Dept of Nucl Engineering
Building C
Rolla, MO 65401

DR Edwards
N. Tsoulfanidis

DISTRIBUTION (Cont'd)

HEDL (46)
c/o Document Processing Supervisor
P.O. Box 1970, W/C-123
Richland, WA 99352

HJ Anderson	W/C-28	JP McNeece	W/A-56
RA Bennett	W/D-3	JE Nolan	W/B-65
LD Blackburn	W/A-40	RE Peterson	W/B-66
DG Doran	W/A-57	CC Preston	W/C-39
EA Evans	W/C-23	JH Roberts	W/C-39
DS Gelles	W/A-64	FH Ruddy	W/C-39
R. Gold	W/C-39	JM Ruggles	W/C-33
GL Guthrie	W/A-40	RE Schenter	W/A-58
BR Hayward	W/C-44	FA Schmittroth	W/A-58
LA James	W/A-40	WF Sheely	W/C-442
LS Kellogg	W/C-39	FR Shober	W/E-3
NE Kenny	W/C-115	RL Simons	W/A-57
RL Knecht	W/A-40	HH Yoshikawa	W/C-44
JJ Laidler	W/B-107	Program Files (10)	W/C-39
EP Lippincott (2)	W/C-39	Central Files (3)	W/C-110
WY Matsumoto	W/C-33	Publ Services	W/C-115
WN McElroy (2)	W/C-39		

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-3391, Vol. 4

2. TITLE AND SUBTITLE
LWR Pressure Vessel Surveillance Dosimetry Improvement Program
Quarterly Progress Report
October 1983 - December 1983

3. LEAVE BLANK

4. DATE REPORT COMPLETED
MONTH: March YEAR: 1984

5. AUTHOR(S)
E.P. Lippincott, W.N. McElroy

6. DATE REPORT ISSUED
MONTH: May YEAR: 1984

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Hanford Engineering Development Laboratory
Richland, WA 99352

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
B5988

10. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Engineering Technology
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

11a. TYPE OF REPORT
Quarterly

b. PERIOD COVERED (Inclusive dates)
October - December 1983

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

The Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program (LWR-PV-SDIP) has been established by NRC to improve, test, verify, and standardize the physics-dosimetry-metallurgy, damage correlation, and the associated reactor analysis methods, procedures and data used to predict the integrated effect of neutron exposure to LWR pressure vessels and their support structures. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the US NRC-supported activities at HEDL, ORNL, NBS, and MEA-ENSA and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany), and several UK laboratories have been extended to a number of other countries and laboratories. These cooperative links are strengthened by the active membership of the scientific staff from many participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR surveillance standards.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS
light water reactor (LWR)
pressure vessel

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page) Unclassified
(This report) Unclassified

17. NUMBER OF PAGES

18. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555