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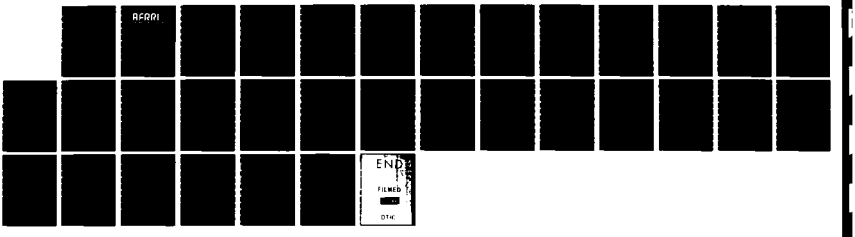

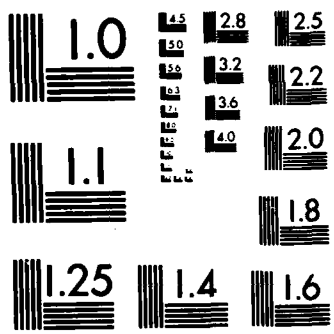

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AFRRI TECHNICAL REPORT

AFRRI TR83-2

Spectrum-averaged kerma factors for reactor dosimetry with paired ion chambers

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Neutron and gamma spectrum-averaged kerma factors (SAKF's) were calculated for four reference spectra and 18 configurations of the AFRRI TRIGA reactor for materials of interest in dosimetry applications. Reactor gamma SAKF ratios for tissue to carbon and for tissue to magnesium (ion chamber wall materials) gave average values ± 2 standard deviations of 1.13 ± 1% and 1.09 ± 3%, respectively. For some reactor configurations, the gamma SAKF ratios were as much as 7.6% different from corresponding values for

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20. ABSTRACT (continued)

cobalt-60 gamma rays. Reactor neutron SAKF ratios for ICRU (International Commission on Radiation Units and Measurements) muscle to ion chamber gas materials were as follows: TE (tissue-equivalent) gas, 0.983 ± 0.5% for all configurations; carbon dioxide, 9.8 + 10%; and argon, 71 ± 27%. At depth in a phantom, the neutron SAKF for ICRU muscle differed substantially from the free-in-air value for the same reactor configuration. This finding suggests the need for more thorough spectrum determinations in anatomical phantoms and for more sophisticated dosimetric tools to better quantitate the dose deposition process.

CONTENTS

INTRODUCTION

Kerma (kinetic energy released in material) is defined as the amount of energy per unit mass transferred by indirectly ionizing radiation to charged particles in a volume of material. Kerma is equal to radiation-absorbed dose under conditions of charged particle equilibrium, provided that energy loss due to bremsstrahlung is negligible. Kerma factors [expressed in units of rad per unit radiation fluence (rad · cm^2)] are functions of the radiation type (neutrons or gamma rays), radiation energy, and material irradiated. Kerma factors are used in neutron gamma dosimetry for a reactor to calculate radiation dose when the fluence is known. Ratios of kerma factors are used to calculate radiation dose in one material, such as tissue, when the radiation dose is known or has been measured in another material, such as an ion chamber (1).

Irradiation by neutrons or gamma rays of a spectrum of energies requires the use of spectrum-averaged kerma factors (SAKF's) to perform dosimetry calculations. Detailed energy spectra were recently determined (2,3) for several configurations of the AFRRI TRIGA reactor. These spectra were used in the present report to calculate SAKF's and ratios of SAKF's for the following materials of interest in reactor neutron gamma dosimetry with paired ion chambers:

$(K_T)_G$, $(K_T)_N$	Gamma and neutron SAKF's for tissue
$(K_T/K_C)_G$, $(K_T/K_{Mg})_G$	Gamma SAKF ratios for carbon and magnesium (ion chamber wall materials)
$(K_T/K_{Ar})_N$, $(K_T/K_{CO_2})_N$, and $(K_T/K_{TE\ gas})_N$	Neutron SAKF ratios for argon, carbon dioxide, and tissue-equivalent (TE) gas (ion chamber gases)

METHODS

Neutron and Gamma Spectra

Neutron and gamma spectra for 18 configurations of the AFRRI TRIGA reactor have been reported (2,3). Table 1 summarizes the configurations for which spectra are available, and Figure 1 shows two sample spectra. The spectra were reported in the format of Oak Ridge Data Library Collection DLC-31 (4), which consists of a 37-energy-group format for neutron spectra and a 21-energy-group format for gamma spectra. Angular distributions of the neutrons were reported in terms of "front" and "back" directions of incidence, but only the total ("front" plus "back") spectra were used in the calculations presented here.

Table 1. Reactor Configurations for Neutron and Gamma Spectra

Room	Distance to Core (cm)	Configuration	Average Energy (MeV) Neutron	Gamma	Neutron Spectra*
1	50	Unshielded	1.46	1.16	1-D
	100	Unshielded	1.49	1.15	1-D, 3-D, MEAS
	200	Unshielded	1.32	1.05	1-D
	300	Unshielded	1.16	0.91	1-D
	400	Unshielded	1.11	0.78	1-D
	500	Unshielded	0.89	0.67	1-D
2	50	Unshielded	1.35	1.18	1-D
	100	Unshielded	1.32	1.13	1-D, 3-D, MEAS
	200	Unshielded	1.12	0.95	1-D
	300	Unshielded	0.93	0.76	1-D
NA	NA	Pneumatic tubes	0.88	0.83	1-D
1	100	7.5 cm H_2O	1.99	1.04	1-D
	100	30 cm H_2O	3.71	0.95	1-D, 3-D
	100	5 cm Pb	1.58	1.65	1-D, 3-D
	100	5 cm Pb and exercise wheel	1.05	1.36	1-D, 3-D, MEAS
	100	15 cm Pb	0.96	0.92	1-D, 3-D
	100	15 cm Pb and cave	0.68	1.80	1-D, 3-D, MEAS
	100	15 cm Pb and cylindrical phantom	0.55	1.13	1-D, 3-D, MEAS

* 1-D, 3-D, and MEAS refer to the availability of neutron spectral data derived from one-dimensional ANISN calculations, three-dimensional Monte Carlo calculations, or activation foil measurements, respectively (2,3).

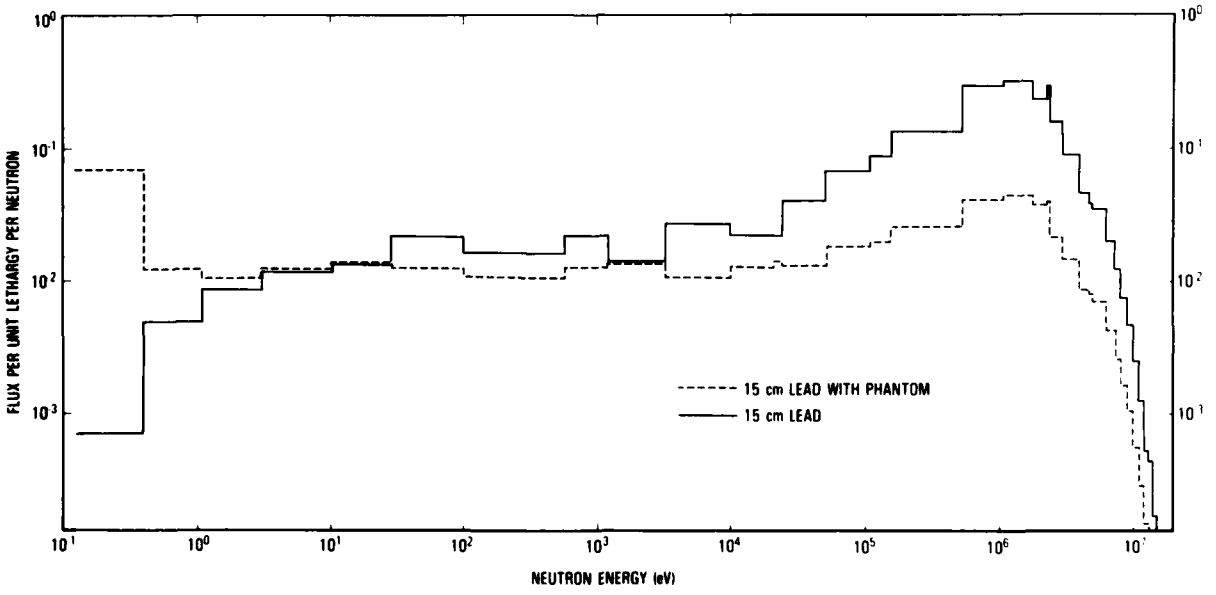

Figure 1. Neutron spectra at 100 cm from the AFRRI TRIGA reactor, shielded by 15-cm lead wall. Solid curve shows spectrum free in air; dashed curve shows spectrum at midline in an 18-cm-diameter cylindrical tissue-equivalent phantom.

The focus of the spectrum calculations of references 2 and 3 was on neutron spectra; thus more reliance can be placed on these than on the gamma spectra. In all cases, one dimensional (1-D) ANISN calculations of the spectra were performed. The gamma spectra excluded fission and fission-product gamma radiation since secondary gamma rays were presumed to dominate the spectrum (3). Therefore, the shapes of the gamma spectra (which depend on the materials irradiated) are probably realistic, but the absolute magnitudes (which depend on the calculational model) must be viewed with caution. For the neutron spectra, the calculations were refined with three-dimensional (3-D) MORSE calculations in nine cases, and verified by activation foil measurements in five of the more complex cases. Thus the neutron spectra represent the best state-of-the-art determinations now available.

To indicate the degree of moderation of each neutron and gamma spectrum, average spectrum energies were computed according to the following formula (and are listed in Table 1):

$$E_{avg} = \frac{\int_{0.4\,eV}^{\infty} E\,\Phi(E)\,dE}{\int_{0.4\,eV}^{\infty} \Phi(E)\,dE} = \frac{\sum_{i=1}^{21\text{ or }36} E_i\,\Phi_i\,\Delta E_i}{\sum_{i=1}^{21\text{ or }36} \Phi_i\,\Delta E_i} \qquad \text{Eq. 1}$$

where E and E_i represent neutron or gamma ray energy, and $\Phi(E)$ and Φ_i represent the number of neutrons or gamma rays with that energy. Note that this definition of average energy excludes thermal neutrons.

In addition to the spectra listed in Table 1, SAKF's were also calculated for the reference spectra given in DLC-31 (4). These spectra were prompt gamma-ray fission source, neutron spectra for fission and thermonuclear sources, and 14-MeV neutrons. A final reference spectrum was the ENDF uranium-235 fission spectrum, which was used as the source term for the AFRRI neutron spectrum calculations (2). While none of these spectra apply directly to the AFRRI reactor configurations, they were included in the present calculations for comparison purposes.

Gamma Kerma Factors

Gamma kerma factors for tissue in a 21-group format were taken directly from DLC-31 (4). Although DLC-31 tissue represents a standard man composed of 11 elements, the gamma kerma factors compared reasonably well with those calculated from mass energy transfer coefficients for muscle (5). Gamma kerma factors for the elements carbon and magnesium (ion chamber wall materials) were also taken directly from DLC-31 (4).

Neutron Kerma Factors

In addition to gamma kerma factors, selected neutron kerma factors were contained in DLC-31 (4), but some materials of particular interest for paired ion chamber constants were not covered, namely ICRU muscle, muscle-equivalent gas, and argon. Consequently, the more complete neutron kerma factor data of ICRU Report 26 (1) were the basis for the present calculations. The energy grouping of the ICRU-26 kerma factor data did not match the 37-group structure of the neutron spectra, so restructuring of the groups was required. The procedures used to convert the ICRU-26 kerma factor data to the 37-group format are described in reference 6.

Calculations

Spectrum-averaging calculations for neutron and gamma kerma factors were done with the AFRRI PDP11 computer. All kerma factor data and all neutron and gamma energy spectra were stored in formatted input files. A fortran computer program named SPT was written to complete the calculations and store the results in separate output files. SPT calculated the following quantities:

$$SAKF = \frac{\int_0^\infty K(E)\,\Phi(E)\,dE}{\int_0^\infty \Phi(E)\,dE} = \frac{\sum_{i=1}^{21\text{ or }37} K_i\,\Phi_i\,\Delta E_i}{\sum_{i=1}^{21\text{ or }37} \Phi_i\,\Delta E_i} \qquad \text{Eq. 2}$$

where SAKF is the spectrum averaged kerma factor, K(E) and K_i are the energy-dependent kerma factors, $\phi(E)$ and ϕ_i are the neutron or gamma fluence, and E is the energy. Note that the SAKF integrals include thermal neutrons in the averaging process. A listing of SPT is contained in Appendix A.

RESULTS

Reactor Gamma SAKF

Gamma SAKF's for tissue are shown in Table 2. The difference between the gamma SAKF's for tissue for the various reactor configurations can be interpreted as changes in the effective energy of the gamma spectra. Tissue-effective energies of the gamma spectra given in Table 2 represent the single photon energies that would give the same tissue kerma factors as the gamma spectra in question. Tissue-effective gamma energies ranged from 0.6 to 1.5 MeV for the reactor configurations, and the energies correlate reasonably

well with the average energies of the spectra listed in Table 1. The fact that tissue-effective gamma energies were 0.1 to 0.3 MeV lower than spectrum-averaged energies expresses the higher attenuation coefficients of lower energy gamma rays in tissue. For materials other than tissue, the effective gamma energies could vary because of differences in atomic number.

Table 2. Reactor Gamma Spectrum-Averaged Kerma Factors and Ratios*

Room	Distance to Core (cm)	Configuration	MeV_{eff} (tissue)	$(K_T)_G$	$(K_T/K_C)_G$	$(K_T/K_{Mg})_G$
1	50	Unshielded	1.0	4.92	1.128	1.108
	100	Unshielded	1.0	4.88	1.130	1.107
	200	Unshielded	0.9	4.47	1.132	1.101
	300	Unshielded	0.8	3.89	1.134	1.090
	400	Unshielded	0.7	3.35	1.139	1.073
	500	Unshielded	0.6	2.93	1.149	1.054
2	50	Unshielded	1.0	5.00	1.129	1.109
	100	Unshielded	1.0	4.89	1.129	1.106
	200	Unshielded	0.8	4.04	1.135	1.092
	300	Unshielded	0.6	3.29	1.142	1.068
NA	NA	Pneumatic tubes	0.7	3.54	1.138	1.089
1	100	7.5 cm H_2O	0.9	4.40	1.131	1.103
	100	30 cm H_2O	0.8	3.99	1.137	1.093
	100	5 cm Pb	1.5	6.77	1.128	1.110
	100	5 cm Pb and exercise wheel	1.2	5.63	1.131	1.108
	100	15 cm Pb	0.8	3.93	1.142	1.082
	100	15 cm Pb and cave	1.5	6.70	1.132	1.076
	100	15 cm Pb and cylindrical phantom	1.0	4.74	1.133	1.092
	Reference Spectra					
	DLC-31	Prompt gamma ray fission	0.8	4.23	1.146	1.065
	Cobalt-60		1.25	5.98	1.126	1.134

* $(K_T)_G$ units are 10^{-10} rad · cm^2. (K_T/K_C) and (K_T/K_{Mg}) are ratios of SAKF's.

Also shown in Table 2 are the ratios of SAKF's for tissue to those for carbon and magnesium. The C and Mg data are presented in this format to be directly useful in dosimetry applications. Average values (± 2 standard deviations) for C and Mg SAKF ratios for all reactor configurations are given below.

$$(K_T/K_C)_G = 1.133 \pm 1.0\%$$
$$(K_T/K_{Mg})_G = 1.09 \pm 3.1\% \qquad \text{Eq. 3}$$

Corresponding kerma factor ratios for cobalt-60 gamma rays (1.25 MeV effective energy) are also shown in Table 2. For C, cobalt-60 gamma rays give a kerma factor ratio lower than for any of the reactor configurations; for Mg, it is higher. This finding is consistent with the presence of low-energy photons (below 300 keV) in each of the gamma spectra of the reactor configurations. The average values (Equation 3) for C and Mg gamma SAKF ratios differ by less than 4% from the corresponding ratios for cobalt-60. However, for some configurations, the influence of low-energy gamma rays is more pronounced and causes gamma SAKF ratios to differ from cobalt-60 values by up to 7.6%. This is the case at great distances from the reactor, where low-energy gamma rays arising from the walls exert an increasing effect, and also behind the 15-cm lead shield, where lead X rays presumably contribute to the total gamma-ray spectrum. On the other hand, the 5-cm lead shield appears to have a hardening effect on the gamma spectrum, as far as C and Mg gamma SAKF ratios are concerned.

Reactor Neutron SAKF for Tissue

Neutron spectra calculated from 3-D models are more reliable than those obtained from 1-D models. Consequently, the present calculations used 3-D neutron spectra for all reactor configurations for which they were available (see Table 1). Comparison of SAKF's for ICRU muscle obtained from 1-D and 3-D spectra is shown in Table 3 for those configurations for which both were available. For noncomplicated configurations such as unshielded at a distance of 100 cm, close agreement occurs between the 1-D and 3-D results. This agreement confirms the adequacy of the 1-D spectra for SAKF calculations at the other unshielded distances. Likewise, the reasonable agreement between the 1-D and 3-D results for the reactor shielded by 30 cm of water justifies the use of a 1-D spectrum for the 7.5-cm water shield. On the other hand, the lack of a 3-D spectrum for the pneumatic tubes leaves open the question of the adequacy of the 1-D neutron spectrum at that location.

Table 3 also presents SAKF values for ICRU muscle for the five measured neutron spectra. These values differ by 2%-58% from those determined from the 3-D calculated spectra. The closest agreement is for the lead cave configuration, for which the calculated and measured neutron spectra plotted in reference 3 are nearly coincident. For the worst case, the cylindrical phantom, the spectrum plots show a clear excess of neutrons below 300 keV in the calculated spectrum. Reference 3 explains this as being due to shortcomings in the calculated spectrum, on the assumption that the measured neutron spectrum is the more accurate. Thus, the SAKF differences between the 3-D and measured spectra can be interpreted as an estimate of the uncertainty in SAKF values due to uncertainties in the neutron spectra.

Table 3. ICRU Muscle SAKF for Calculated and Measured Reactor Neutron Spectra

Room	Distance to Core (cm)	Configuration	SAKF (10^{-10} rad cm^2) 1-D	3-D	MEAS
1	100	Unshielded	19.4	20.3	16.8
2	100	Unshielded	13.8	14.4	10.9
1	100	30 cm H_2O	27.3	34.4	
1	100	5 cm Pb	17.8	20.5	
1	100	5 cm Pb and exercise wheel	03.69	12.0	13.2
1	100	15 cm Pb	14.9	17.0	
1	100	15 cm Pb and cave	07.94	13.6	13.9
1	100	15 cm Pb and cylindrical phantom	02.11	02.85	4.51

Table 4 presents the reactor neutron SAKF's for three tissuelike materials. ICRU muscle, the generally accepted reference material for neutron dosimetry (7,8) is used as such here. ICRU tissue differs from ICRU muscle by a 2% lower hydrogen content (by weight), and the calculated kerma factors reflect this difference. DLC-31 tissue represents standard man composed of 11 elements, but the neutron SAKF values differ negligibly from those for ICRU muscle or ICRU tissue. A-150 is the tissue-equivalent plastic used for tissue-equivalent ion chamber construction.

Tissue-effective energies of the neutron spectra tabulated in Table 4 were determined in the same way as those of the gamma spectra. That is, tissue-effective spectrum energies were taken as the monoenergetic neutron energies giving the same tissue kerma factors as the spectra in question. The tissue-effective neutron energies display a decreasing trend with increasing free-in-air distance from the reactor (due to the increased importance of scattered neutrons), and an increasing trend for water-shielded configurations (due to a beam-hardening effect). Two phenomena are noteworthy: (a) The tissue-effective neutron energies are not equal to, nor do they correlate very well with, the average neutron energies given in Table 1. This finding emphasizes the fact that neutron spectra of complex shape cannot be represented well by any single energy descriptor. (b) The tissue-effective neutron energies are low at the two points with hydrogenous surrounds, i.e., pneumatic tubes and cylindrical phantom. Consideration of the shapes of the two neutron spectra plotted in Figure 1 clarifies this phenomenon. The spectra pertain to the reactor shielded with 15 cm of Pb, at a distance of 100 cm. One spectrum represents the case with no phantom present; the other spectrum is that at the center of the 18-cm-diameter cylindrical phantom. Within the phantom the total neutron flux below 0.01 MeV is 8.8 times the flux without the phantom, while the flux above 0.01 MeV is decreased by a factor of 5. It is the large low-energy component of the neutron spectrum within the phantom that causes such low tissue-effective neutron energy.

Table 4. Reactor Neutron Spectrum-Averaged Kerma Factors for Tissue

Room	Distance to Core (cm)	Configuration	MeV_{eff}	SAKF (10^{-10} rad cm^2) Muscle (ICRU)*	Tissue (ICRU)†	Tissue (DLC-31)‡	A-150 Plastic§
1	50	Unshielded	0.8	19.8	19.6	19.6	20.0
	100	Unshielded	0.7	20.3	20.2	20.0	20.6
	200	Unshielded	0.6	17.6	17.4	17.4	17.8
	300	Unshielded	0.4	15.4	15.2	15.2	15.6
	400	Unshielded	0.4	13.6	13.4	13.4	13.7
	500	Unshielded	0.3	12.1	11.9	11.9	12.2
2	50	Unshielded	0.4	14.2	14.0	14.0	14.4
	100	Unshielded	0.4	14.4	14.2	14.2	14.5
	200	Unshielded	0.3	12.1	11.9	11.9	12.2
	300	Unshielded	0.2	10.6	10.5	10.5	10.7
NA	NA	Pneumatic tubes	0.04	3.83	3.70	3.75	3.87
1	100	7.5 cm H_2O	1.0	23.7	23.4	23.4	24.1
	100	30 cm H_2O	2.7	34.4	33.9	34.0	35.1
	100	5 cm Pb	0.8	20.5	20.2	20.2	20.4
	100	5 cm Pb and exercise wheel	0.3	12.0	11.8	11.8	12.0
	100	15 cm Pb	0.6	17.0	16.8	16.8	16.8
	100	15 cm Pb and cave	0.4	13.6	13.4	13.4	13.5
	100	15 cm Pb and cylindrical phantom	0.03	2.85	2.81	2.77	2.83
	Reference Spectra						
		DLC-31 fission	0.8	20.4	20.1	20.1	20.4
		DLC-31 thermonuclear	0.6	18.1	17.8	17.8	18.3
		ENDF fission	1.16	28.3	27.9	27.9	28.4
		DLC-31 14 MeV	14.	64.6	63.9	63.2	67.4

* Muscle (ICRU): 10.2% H, 12.3% C, 3.5% N, 72.9% O, 1.1% (Na + Mg + P + S + K + Ca) (from reference 1)

† Tissue (ICRU): 10.0% H, 14.9% C, 3.5% N, 71.6% O (from reference 1)

‡ Tissue (DLC-31): 10% H, 24% C, 2.9% N, 60% O, 0.20% Na, 0.30% Mg, 1.1% P, 0.24% S, 0.20% K, 1.2% Ca, 0.20% Cl (from reference 5 of reference 4)

§ A-150 plastic: 10.1% H, 77.6% C, 3.5% N, 5.2% O, 1.8% Ca, 1.7% F (from reference 1)

Reactor Neutron SAKF Ratios

SAKF results for the three ion chamber gas materials are shown in Table 5. For TE gas, all the reactor configurations can be represented by a single $(K_T/K_{TE\ gas})_N$ value of 0.983 ± 0.5% (2 SD). For CO_2 and Ar, the SAKF ratios can be summarized as 9.78 ± 10% and 70.7 ± 27%, respectively. The variability among the CO_2 and Ar SAKF ratios indicates that individually tabulated values should be used for each reactor configuration.

Table 5. Reactor Neutron SAKF Ratios

Room	Distance to Core (cm)	Configuration	$(K_T/K_{TE\ gas})_N$*	(K_T/K_{CO_2})	(K_T/K_{Ar})
1	50	Unshielded	0.980	9.57	66.0
	100	Unshielded	0.979	9.50	64.3
	200	Unshielded	0.983	9.62	66.7
	300	Unshielded	0.981	9.68	67.5
	400	Unshielded	0.986	9.78	69.0
	500	Unshielded	0.984	9.84	70.3
2	50	Unshielded	0.979	9.59	66.0
	100	Unshielded	0.986	9.66	65.8
	200	Unshielded	0.984	9.68	67.2
	300	Unshielded	0.981	9.81	69.3
NA	NA	Pneumatic tubes	0.982	10.21	73.2
1	100	7.5 cm H_2O	0.979	9.26	61.2
	100	30 cm H_2O	0.983	8.49	55.5
	100	5 cm Pb	0.986	9.67	69.7
	100	5 cm Pb and exercise wheel	0.984	9.92	73.6
	100	15 cm Pb	0.983	10.18	83.7
	100	15 cm Pb and cave	0.986	10.38	92.5
	100	15 cm Pb and cylindrical phantom	0.986	11.00	91.1
	Reference Spectra				
	DLC-31 fission		0.986	9.86	76.7
	DLC-31 thermonuclear		0.989	6.24	50.7
	ENDF fission		0.983	9.53	65.2
	DLC-31 14 MeV		0.977	3.40	26.5

* TE gas: 10.2% H, 45.6% C, 3.5% N, 40.7% O

The constancy of $(K_T/K_{TE\ gas})_N$ among the reactor configurations was to be expected, because this ratio varied only from 0.95 to 1.01 over the entire range of 37-energy groups. For the other neutron SAKF ratios, uncertainties can be estimated by again comparing results from calculated and measured spectra, as shown in Table 6. Although individual SAKF values have uncertainties of 2%-58%, SAKF ratios determined from calculated and measured spectra differ by only 1%-8%.

Table 6. SAKF Ratios for Calculated and Measured Neutron Spectra

Room	Distance to Core (cm)	Configuration	Neutron Spectrum 3-D	Neutron Spectrum MEAS	Difference (%)
K_T/K_{Ar}					
1	100	Unshielded	64.3	67.5	-5
2	100	Unshielded	65.8	67.7	-3
1	100	5 cm Pb and exercise wheel	73.6	74.6	-1
1	100	15 cm Pb and cave	92.5	91.4	+1
1	100	15 cm Pb and cylindrical phantom	91.1	84.3	+8
K_T/K_{CO_2}					
1	100	Unshielded	9.50	9.66	-2
2	100	Unshielded	9.66	9.73	-1
1	100	5 cm Pb and exercise wheel	9.92	10.00	-1
1	100	15 cm Pb and cave	10.38	10.30	+1
1	100	15 cm Pb and cylindrical phantom	11.0	10.56	+4

DISCUSSION

The gamma SAKF ratios for C and Mg (ion chamber wall materials) vary only slightly among the reactor configurations, and average values represent these quantities with fairly high precision. Historically it has been questioned (9,10) whether the AFRRI reactor gamma spectra can be approximated by cobalt-60 radiation (1.25 MeV). The present calculations confirm that this is nearly the case, since the average C and Mg SAKF ratios differ less than 4% from the values applicable to cobalt-60. However, in some configurations, a low-energy photon component in the reactor gamma spectra does cause C and Mg SAKF ratios to differ by up to 7.6% from corresponding values for cobalt-60.

The constancy of the neutron SAKF ratio for TE gas between the various reactor configurations was to be expected, because the compositions of TE gas and ICRU muscle match closely. For CO_2 and Ar, the neutron SAKF ratios are nearly constant for all free-in-air unshielded reactor configurations, but the ratios differ considerably (5%-40%) for more complex reactor configurations.

The differences between SAKF values derived from 3-D calculated neutron spectra and those derived from measured neutron spectra allow estimation of the uncertainties in the reported SAKF values arising from spectrum uncertainties. For ICRU muscle, the observed 2%-58% differences place a limit on dosimetric accuracy that can be obtained using a direct fluence-to-dose method. This emphasizes the importance of using ionization chambers for dose determination since the overall uncertainty of those chambers is about 10% (8). In this regard, the SAKF ratios for CO_2 and Ar differ by only 1%-8% between the 3-D calculated and measured neutron spectra. The improved precision for SAKF ratios over individual SAKF values is due to the similarity in energy response of the kerma factors for the different materials. This improved precision is indeed fortunate, because the SAKF ratios are required to evaluate paired ionization chamber response constants for reactor neutron and gamma dosimetry.

Considering the SAKF results for the reference spectra, it was apparent that for ICRU muscle and TE gas, the prompt gamma-ray fission source as well as the fission and thermonuclear sources fell within the range of values obtained for AFRRI reactor configurations. In particular, the 15-cm lead shield gave the same tissue-effective neutron energy as the thermonuclear source, and the 5-cm lead shield matched the fission source. However, these apparent similarities should be viewed with caution because, as pointed out above, no single energy descriptor can adequately represent a complex energy spectrum.

The results of this study demonstrate the need for further neutron spectrum calculations or measurements for AFRRI reactor configurations at depth in tissue-equivalent or anatomical phantoms. The large change in the neutron spectrum within the 18-cm-diameter phantom shows the importance of also studying phantoms of different sizes and shapes. The very low tissue-effective energy of the neutron spectrum within the phantom also suggests the need for more sophisticated dosimetric tools (such as tissue-equivalent proportional counters) to supplement the paired ionization chambers and to better quantitate the dose deposition process.

APPENDIX A.

Fortran Computer Program SPT

```
C** SPT
C       LATEST CORRECTION 17 AUG 82
C+
C       THIS PROGRAM WILL COMPUTE TWO SPECTRUM DEPENDENT FUNCTIONS:
C       A SPECTRUM WEIGHTED FUNCTION AND A SPECTRUM FOLDED FUNCTION,
C       AND THE ESTIMATED VARIATION OF EACH
C-
C       AUTHOR: KEN FERLIC  CIRCA 1981 OR 82
C
C       THIS PROGRAM IS FOR EVALUATING THE INTEGRALS (BY SUMMATION)
C       NECESSARY FOR SPECTRUM AVERAGED QUANTITIES
C
C       NOTES
C       1. SPECTRUM INPUT DATA CAN BE A FROM AND BACK SPECTRUM
C       2. SPECTRUM INPUT DATA CAN BE LETHARGY, PHI(E)DE OR PHI(E)
C       3. THE DELTA SPECTRUM IS THE FACTIONAL CHANGE EXPECTED IN NEUTRON
C          POPULATION FOR A PARTICULAR ENERGY GROUP
C       4. THE DELTA FUNCTION IS THE FRACTIONAL CHANGE EXPECTED IN THE
C          FUNCTION PER ENERGY GROUP
C       5. ENERGY GROUP BOUNDS ARE THE UPPER BOUNDS
C       6. ENERGY GROUP 1 IS THE MAXINUM ENERGY GROUP. I.E. GROUP 1 UPPER
C          BOUND IS 19.6 MEV

C
C       HIGHEST CONTROL STATEMENT IS 238
C       HIGHEST FORMAT STATEMENT IS 1217
C

        BYTE FILE1(30)  !FILE1=FLUX SPECTRUM FILE
        BYTE FILE2(30)  !FILE2=ENERGY GROUP FILE, UPPER BOUNDS
        BYTE FILE3(30)  !FILE3=FUNCTION FILE BY GROUP
        BYTE FILE4(30)  !FILE4=OUTPUT FILE
        BYTE FILE5(30)  !FILE5=DELTA SPECTRUM FILE
        BYTE FILE6(30)  !FILE6=DELTA FUNCTION FILE
        DIMENSION PHIT(38)      !TOTAL INPUT FLUX
        DIMENSION PHIF(38)      !TOTAL FRONT FLUX
        DIMENSION PHIB(38)      !TOTAL BACK FLUX
        DIMENSION IGROUP(38)    !GROUP NUMBER
        INTEGER NROW            !NUMBER OF GROUPS NUMBER OF INPUT ROWS
        INTEGER ICOL            !NUMBER OF COLUMNS IN INPUT DATA
        INTEGER ICNT            !COUNTER OF FOR FILE NAMES
        INTEGER NSPT            !TYPE OF FLUX INPUT IE ENERGY,LETHARGY
        INTEGER NTYPE           !TYPE OF INPUT SPECTRUM TOTAL VS FRONT + BACK
        DIMENSION ENGP(38)      !ENERGY GROUP UPPER BOUNDS
        DIMENSION FNCT(38)      !FUNCTION GROUPS
        DIMENSION DE(38)        !CHANGE IN ENERGY
        DIMENSION DU(38)        !CHANGE IN LETHARGY
        REAL DUBWI              !DESIRED UPPER BOUND WEIGHTING INTEGRAL
        REAL DLBWI              !DESIRES LOWER BOUND WEIGHTING INTEGRAL
        REAL DUBNI              !DESIRED UPPER BOUND NORMALIZING INTEGRAL
        REAL DLBNI              !DESIRED LOWER BOUND NORMALIZING INTEGRAL
        INTEGER IUBGWI          !UPPER BOUND GROUP WEIGHTING INTEGRAL
        REAL UBFWI              !UPPER BOUND FRACTION WEIGHTING INTEGRAL
        INTEGER IUBFGW          !UPPER BOUND FRACTIONAL GROUP WEIGHTING INTEG
        INTEGER IUBGNI          !UPPER BOUND GROUP NORMALIZING INTEGRAL
        REAL UBFNI              !UPPER BOUND FRACTION NORMALIZING INTEGRAL
        INTEGER IUBFGN          !UPPER BOUND FRACTIONAL GROUP NORMALIZING INT
        INTEGER ILBGWI          !LOWER BOUND GROUP WEIGHTING INTEGRAL
        REAL LBFWI              !LOWER BOUND FRACTION WEIGHTING INTEGRAL
        INTEGER ILBFGW          !LOWER BOUND FRACTIONAL GROUP WEIGHTING INTEG
        INTEGER ILBFGN          !LOWER BOUND FRACTIONAL GROUP NORMALIZING INT
        REAL LBFNI              !LOWER BOUND FRACTION NORMALIZING INTEGRAL
```

```
      INTEGER ILBGNI          !LOWER BOUND GROUP NORMALIZING INTEGRAL
      REAL WINT               !INTEGRAL OF THE WEIGHTING INTERGRAL
      REAL NINT               !INTEGRAL OF THE NORMALIZING INTEGRAL
      REAL FNCEFF             !EFFECTIVE VALUE OF FUNCTION
      INTEGER NROW1           !ROWS IN INPUT SPECTRUM FILE
      INTEGER NROW2           !ROWS IN INPUT ENEGRY GROUP FILE
      INTEGER NROW3           !ROWS IN INPUT FUNCTION GROUP FILE
      INTEGER NROW5           !ROWS IN INPUT DELTA SPECTRUM FILE
      INTEGER NROW6           !ROWS IN INPUT DELTA FUNCTION FILE
      DIMENSION FRCTW(38)     !FRACTION OF WEIGHTING INTEGRAL FOR GROUP I
      DIMENSION FRCTN(38)     !FRACTION OF NORMALIZING INTEGRAL FOR GROUP I
      DIMENSION DELTA(38)     !DE OR DU IN INTEGRAL FOR GROUP I
      REAL UTAIL              !UPPER TAIL OF INTEGRAL BETWEEN GROUPS
      REAL LTAIL              !LOWER TAIL OF INTEGRAL BETWEEN GROUPS
      INTEGER SAF             !CONTROL FOR SPECTRUM AVERAGED FUNCTIONS
      INTEGER VSAF            !CONTROL FOR VARIATION OF SPECTRUM AVERAGED
      INTEGER SFF             !CONTROL FOR SPECTRUM FOLDED FUNCTION
      INTEGER VSFF            !CONTROL FOR VARIATION OF SPECTRUM FOLDED
      INTEGER INPUT1          !DELTA (VARIATION) FILES CONTROL CHARACTER
      INTEGER INPUT2          !BOUNDS ON NORMALIZING INTEGRAL CONTROL
      INTEGER INPUT3          !TEMPORARY NORMALIZING INTEGRAL CONTROL
      DIMENSION DFF(38)       !DELTA FUNCTION ENTRY
      DIMENSION DSF(38)       !DELTA SPECTRUM ENTRY
      DIMENSION PHIENR(38)    !PHI(E) FOR EACH GROUP
      REAL UBFEW              !FRACTION OF ENERGY UPPER END WGT INT
      REAL UBFEN              !FRACTION OF ENERGY UPPER END NOR INT
      REAL LBFEW              !FRACTION OF ENERGY LOWER END WGT INT
      REAL LBFEN              !FRACTION OF ENERGY LOWER END NOR INT
      DIMENSION DEW(38)       !DELTA E FOR WGT INT IN VARIATION CALCULATION
      DIMENSION DEN(38)       !DELAT E FOR NOR INT IN VARIATION CALCULATION
      DIMENSION PWPHI(38)     !PARTIAL WGT WITH RESPECT TO PHI
      DIMENSION PWFNT(38)     !PARTIAL WGT WITH RESPECT TO THE FUNCTION
      DIMENSION PWSUM(38)     !SUM OF PARTIALS WITH RESP TO PHI AND FUNCT
      REAL VFFNCT             !VARIATION OF FOLDED FUNCTION
      DIMENSION TERM1(38)     !FIRST TERM IN CALCULATION OF VAR OF WGT FNCT
      DIMENSION TERM2(38)     !SECOND TERM IN CALCUL OF VAR OF WGT FNCT
      DIMENSION TERM3(38)     !THIRD TERM IN CALCUL OF VAR OF WGT FNCT
      DIMENSION PWFPHI(38)    !PARTIAL OF WGT FUNCT WITH RESPECT TO PHI SQRT
      DIMENSION PWFFNT(38)    !PARTIAL OF WGT FNCT WITH RESPECT TO FNT SQRT
      REAL VWFNT              !VARIATION OF WEIGHTED FUNCTION

C
C     COMMON STATEMENTS FOR SUBROUTINES UPBND AND LWBND
C

      COMMON IXGP,XDUB,XE(38),IXUBG,IXUBFG,XUBF
      COMMON XDLB,IXLBG,IXLBFG,XLBF,XUBFE,XLBFE

C
C     DETERMINATION OF CALCULATIONS TO BE PERFORMED
C
      TYPE 1200
1200  FORMAT(' DO YOU WANT A SPECTRUM AVERAGED FUNCTION? YES=1,NO=0
     1 OR CR')
      ACCEPT 1201,SAF
1201  FORMAT(I3)
      IF(SAF .EQ. 0)GO TO 200
      TYPE 1202
1202  FORMAT(' DO YOU WANT THE ESTIMATED VARIATION? YES=1, NO=0 OR CR')
      ACCEPT 1201,VSAF
200   CONTINUE
```

```
      TYPE 1203
1203  FORMAT(' DO YOU WANT A SPECTRUM FOLDED FUNCTION? YES=1,NO=0 OR CR')
      ACCEPT 1201,SFF
      IF(SFF .EQ. 0)GO TO 201
      TYPE 1204
1204  FORMAT(' DO YOU WANT THE ESTIMATED VARIATION? YES=1, NO=0 OR CR')
      ACCEPT 1201,VSFF
201   CONTINUE
      IF(SAF .NE. SFF)GO TO 203
      IF(SAF .EQ. 0)GO TO 202
      GO TO 203
202   TYPE 1205
1205  FORMAT(' WHAT DO YOU WANT???  BYE, BYE')
      GO TO 140
203   CONTINUE

C
C     PROGRAM CONTROL FOR CALCULATIONS, DETERMINATION OF ACTUAL INPUT DATA
C
C
C     INPUT1  DELTA FILES INPUT,
C     INPUT2  BOUNDS ON NORMALIZING INTEGRAL  INPUT
C

      IF(SAF .EQ. 0)GO TO 204
      INPUT1=0
      INPUT2=1
      IF(VSAF .EQ.0)GO TO 204
      INPUT1=1
204   CONTINUE
      IF(SFF .EQ. 0)GO TO 205
      INPUT1=0
      INPUT3=0
      IF(VSFF .EQ.0)GO TO 205
      INPUT1=1
205   CONTINUE
      IF(INPUT2 .EQ. INPUT3)GO TO 229
      INPUT2=1
229   CONTINUE

C
C     IDENTIFY NUMBER OF GROUPS
C

      TYPE 1116
1116  FORMAT(' WHAT IS THE NUMBER OF GROUPS IN SPECTRUM:<I3>')
      ACCEPT 1006,NROW

C
C     SET LOW BOUND ON LOWEST ENERGY GROUP
C

      IF(NROW .EQ. 37)GO TO 119
      IF(NROW .EQ. 21)GO TO 120
      TYPE 1120
1120  FORMAT(' WHAT IS LOWER BOUND (IN EV) ON THE LOWEST ENERGY'
     1' GROUP?<E10.4>')
      ACCEPT 1121, ENGP(NROW+1)
1121  FORMAT(E10.8)
      GO TO 121
119   ENGP(38)=1.0E-5          'IN EV
      GO TO 121
```

```
120     ENGP(22)=1.0E+4          !IN EV
121     CONTINUE

C
C       FILE1 NAME: SPECTRUM FILE
C

        TYPE 1000
1000    FORMAT(' ENERGY SPECTRUM FILE NAME <FILENAMEX.TYP>:')
        ACCEPT 1001,ICNT,FILE1
1001    FORMAT(Q,30A1)
        FILE1(ICNT+1)=0
        TYPE 1005
1005    FORMAT(' SPECTRUM INPUT: FRONT + BACK = 0, TOTAL = 1')
        ACCEPT 1006, NTYPE
1006    FORMAT(I3)
        TYPE 1115
1115    FORMAT(' IS SEPCTRUM INPUT: PHI(U)=1,PHI(E)DE=0,PHI(E)=-1')
        ACCEPT 1006,NSPT

C
C       FILE2 NAME: ENERGY GROUP UPPER BOUND FILE
C

        TYPE 1003
1003    FORMAT(' ENTER ENERGY GROUP (UPPER BOUND) FILE: <FILENAMEX.TYP>')
        ACCEPT 1001,ICNT,FILE2
        FILE2(ICNT+1)=0

C
C       FILE3 NAME: FUNCTION GROUP FILE
C

        TYPE 1004
1004    FORMAT(' ENTER FUNCTION GROUP FILE: <FILENAMEX.TYP>')
        ACCEPT 1001,ICNT,FILE3
        FILE3(ICNT+1)=0

C
C       FILE4 NAME: OUTPUT FILE NAME
C

        TYPE 1198
1198    FORMAT(' ENTER OUTPUT FILE NAME:<FILENAMEX.TYP>')
        ACCEPT 1001,INCT,FILE4
        FILE4(INCT+1)=0

C
C       FILE5 NAME: DELTA SPECTRUM FILE
C

        IF(INPUT1 .EQ. 0)GO TO 206
        TYPE 1206
1206    FORMAT(' ENTER DELTA SPECTRUM FILE NAME <FILENAMEX.TYP>:')
        ACCEPT 1001,INCT,FILE5
        FILE5(INCT+1)=0

C
C       FILE6 NAME: DELTA FUNCTION FILE
C

        TYPE 1207
```

```
1207    FORMAT(' ENTER DELTA FUNCTION FILE NAME <FILENAMEX.TYP>:')
        ACCEPT 1001,INCT,FILE6
        FILE6(INCT+1)=0
206     CONTINUE

C
C       EXTRACT DATA FROM SPECTRUM FILE: FILE1
C

        OPEN(UNIT=1,NAME=FILE1,TYPE='OLD')
        READ(1,1007) ICOL,NROW1
1007    FORMAT(2I4)
        IF(NROW1 .NE. NROW)GO TO 112
        IF(NTYPE .EQ. 0)GO TO 100
        DO 101 I=1,NROW
        READ(1,1008) IGROUP(I),PHIT(I)
1008    FORMAT(I3,E12.4)
101     CONTINUE
        GO TO 102
100     DO 103 I=1,NROW
        READ(1,1009)IGROUP(I),PHIF(I),PHIB(I)
1009    FORMAT(I3,2E12.4)
103     CONTINUE
102     CONTINUE
        CLOSE(UNIT=1)

C
C       EXTRACT DATA FROM ENERGY GROUP FILE
C

        OPEN(UNIT=1,NAME=FILE2,TYPE='OLD')
        READ(1,1007)ICOL,NROW2
        IF(NROW2 .NE. NROW)GO TO 114
        DO 104 I=1,NROW
        READ(1,1008) IGROUP(I),ENGP(I)
104     CONTINUE
        CLOSE(UNIT=1)

C
C       EXTRACT DATA FROM FUNCTION FILE:FILE3
C

        OPEN(UNIT=1,NAME=FILE3,TYPE='OLD')
        READ(1,1007),ICOL,NROW3
        IF(NROW3 .NE. NROW)GO TO 115
        DO 105 I=1,NROW
        READ(1,1008), IGROUP(I),FNCT(I)
105     CONTINUE
        CLOSE(UNIT=1)

C
C       EXTRACT DATA FROM DELTA SPECTRUM FILE: FILE5
C

        IF(INPUT1 .EQ. 0)GO TO 207
        OPEN(UNIT=1,NAME=FILE5,TYPE='OLD')
        READ(1,1007),ICOL,NROW5
        IF(NROW5 .NE. NROW)GO TO 208
        DO 209 I=1,NROW
        READ(1,1008),IGROUP(I),DSF(I)
209     CONTINUE
        CLOSE(UNIT=1)
```

```
C
C       EXTRACT DATA FROM DELTA FUNCTION FILE:FILE6
C
        OPEN(UNIT=1,NAME=FILE6,TYPE='OLD')
        READ(1,1007),ICOL,NROW6
        IF(NROW6 .NE. NROW)GO TO 210
        DO 211 I=1,NROW
        READ(1,1008),IGROUP(I),DFF(I)
211     CONTINUE
        CLOSE(UNIT=1)
207     CONTINUE
        GO TO 141

C
C       CHECK FOR PROPER NUMBER OF GROUPS IN EACH INPUT FILE; NEGATIVE ANSWER
C

112     TYPE 1117,NROW1
        GO TO 113
114     TYPE 1117,NROW2
        GO TO 113
115     TYPE 1117,NROW3
        GO TO 113
208     TYPE 1117,NROW5
        GO TO 113
210     TYPE 1117,NROW6
1117    FORMAT(' ACTUAL GROUPS IN FILE',I3)
113     TYPE 1199,NROW
1199    FORMAT(' DOES NOT EQUAL IDENTIFIED SPECTRUM GROUPS:',I3)
        CLOSE(UNIT=1)
        GO TO 140

C
C       CALCULATE TOTAL FLUX IF FRONT AND BACK
C

141     IF(NTYPE .EQ. 1)GO TO 106
        DO 107 I=1, NROW
        PHIT(I)=PHIF(I)+PHIB(I)
107     CONTINUE
106     CONTINUE

C
C       CALCULATE DU OR DE FOR THE SPECTRUM
C

C       OLD LINE        IF(NSPT) 116,117,118

C
C       CHANGE IN LETHARGY/ENERGY
C

118     DO 108 I=1,NROW
        DU(I)=LOG(ENGP(I)/ENGP(I+1))
108     CONTINUE
C       OLD LINE        GO TO 117
116     DO 109 I=1,NROW
        DE(I)=ENGP(I)-ENGP(I+1)
109     CONTINUE
117     CONTINUE
```

```
C
C       ACCEPT BOUNDS ON WEIGHTING AND NORMALIZING INTEGRAL
C

        TYPE 1010
1010    FORMAT(' WHAT IS UPPER BOUND OF WEIGHTING INTEGRAL IN EV:'
        1 '<E10.4>')
        TYPE 1122
1122    FORMAT(' NEUTRON 37 GROUP MAX IS 19.6E+6 EV; PHOTON 21 GROUP MAX'
        1 'IS 14.0E+6 EV')
        ACCEPT 1011,DUBWI
1011    FORMAT(E10.4)
        TYPE 1012
1012    FORMAT(' WHAT IS LOWER BOUND OF WEIGHTING INTEGRAL IN EV:'
        1 '<E10.4>')
        TYPE 1123
1123    FORMAT(' NEUTRON 37 GROUP MIN IS 1.0E-5 EV; PHOTON 21 GROUP MAX'
        1 'IS 1.0E+4 EV')
        ACCEPT 1011,DLBWI

C
C       ACCEPT BOUNDS ON NORMALIZING INTEGRAL
C
        IF(INPUT2 .EQ. 0)GO TO 212
        TYPE 1013
1013    FORMAT(' WHAT IS UPPER BOUND ON NORMALIZING INTEGRAL IN EV:'
        1 '<E10.4>')
        TYPE 1122
        ACCEPT 1011,DUBNI
        TYPE 1014
1014    FORMAT(' WHAT IS LOWER BOUND ON NORMALIZING INTEGRAL IN EV:'
        1 '<E10.4>')
        TYPE 1123
        ACCEPT 1011,DLBNI
212     CONTINUE

C
C       SET UPPER BOUND ON WEIGHTING INTEGRAL
C

        IXGP=NROW+1
        XLUB=DUBWI
        DO 151 I=1, NROW+1
        XE(I)=ENGP(I)
151     CONTINUE
        CALL UPBND
        IUBGWI=IXUBG
        UBFWI=XUBF
        IUBFGW=IXUBFG
        UBFEW=XUBFE

C
C       SET UPPER BOUND ON NORMALIZING INTEGRAL
C

        IF(INPUT2 .EQ. 0)GO TO 213
        XDUB=DUBNI
        CALL UPBNI
        IUBFGN=IXUBFG
        UBFNI=XUBF
        IUBGNI=IXUBG
        UBFEN=XUBFE
```

```
213     CONTINUE

C
C       SET LOWER BOUND WEIGHTING INTEGRAL
C

        XDLB=DLBWI
        CALL LOWBND
        ILBGWI=IXLBG
        LBFWI=XLBF
        ILBFGW=IXLBFG
        LBFEW=XLBFE
        LBFEW=XLBFE

C
C       SET LOWER BOUND ON NORMALIZING INTEGRAL
C

        IF(INPUT2 .EQ. 0)GO TO 214
        XDLB=DLBNI
        CALL LOWBND
        ILBFGN=IXLBFG
        LBFNI=XLBF
        ILBGNI=IXLBG
        LBFEN=XLBFE
214     CONTINUE

C
C       EVALUATION OF WEIGHTING INTEGRAL
C

        DO 110 I=1,NROW
        FRCTW(I)=0.0
        FRCTN(I)=0.0
110     CONTINUE

C
C       DETERMINE DELTA TERMS FOR INTEGRALS I.E. DE OR DU
C       PLACE ALL DATA IN THE FORM OF PHI(E, IRREGARDLESS OF INPUT
C
        IF(NSPT) 122,123,124
122     DO 125 I=1,NROW
        DELTA(I)=DE(I)
        PHIENR(I)=PHIT(I)
125     CONTINUE
        GO TO 126
123     DO 127 I=1,NROW
        DELTA(I)=1.0
        PHIENR(I)=PHIT(I)/DE(I)
127     CONTINUE
        GO TO 126
124     DO 128 I=1,NROW
        DELTA(I)=DU(I)
        PHIENR(I)=PHIT(I)*DU(I)/DE(I)
128     CONTINUE

C
C       EVALUATION OF WEIGHTING/FOLDING INTEGRAL
C

126     SUM=0.0
        DO 150 I=IUBGWI,ILBGWI
```

```
      FRCTW(I)=PHIT(I)*DELTA(I)*FNCT(I)
      SUM=SUM+FRCTW(I)
150   CONTINUE
      UTAIL=PHIT(IUBFGW)*DELTA(IUBFGW)*FNCT(IUBFGW)*UBFWI
      LTAIL=PHIT(ILBFGW)*DELTA(ILBFGW)*FNCT(ILBFGW)*LBFWI
      IF(IUBGWI .EQ. IUBFGW)GO TO 230
      FRCTW(IUBFGW)=UTAIL
230   IF(ILBGWI .EQ. ILBFGW)GO TO 231
      FRCTW(ILBFGW)=LTAIL
231   WINT=SUM+UTAIL+LTAIL

C
C     EVALUATION OF NORMALIZING INTEGRAL
C

      IF(INPUT2 .EQ. 0)GO TO 215
      SUM=0.0
      DO 111 I=IUBGNI,ILBGNI
      FRCTN(I)=PHIT(I)*DELTA(I)
      SUM=SUM+FRCTN(I)
111   CONTINUE
      UTAIL=PHIT(IUBFGN)*DELTA(IUBFGN)*UBFNI
      LTAIL=PHIT(ILBFGN)*DELTA(ILBFGN)*LBFNI
      IF(IUBGNI .EQ. IUBFGN)GO TO 232
      FRCTN(IUBFGN)=UTAIL
232   IF(ILBGNI .EQ. ILBFGN)GO TO 233
      FRCTN(ILBFGN)=LTAIL
233   NINT=SUM+UTAIL+LTAIL

C
C     EVALUATE EFFECTIVE FUNCTION
C

      FNCEFF=WINT/NINT
215   CONTINUE

C
C     SET OF DE FOR DELTA E IN THE FOLDED FUNCTION VARIATION EQUATION
C

      IF(INPUT1 .EQ. 0)GO TO 216
      IC=IUBFGW-1
      DO 217,I=1,IC
      DEW(I)=0.0
217   CONTINUE
      IC=(NROW+1)-ILBFGW
      DO 218 I=1,IC
      L=(NROW+1)-I
      DEW(L)=0.0
218   CONTINUE
      DO 219 I=IUBFGW,ILBFGW
      DEW(I)=DE(I)
219   CONTINUE
      IF(IUBFGW .EQ. IUBGWI)GO TO 236
      DEW(IUBFGW)=DEW(IUBFGW)*UBFEW
236   IF(ILBFGW .EQ. ILBGWI)GO TO 237
      DEW(ILBFGW)=DEW(ILBFGW)*LBFEW
237   CONTINUE

C
C     COMPUTE ESTIMATED VARIATION IN THE FOLDED FUNCTION
C
```

```
      SUM=0.0
      DO 220 I=1,NROW
      PWPHI(I)=(FNCT(I)*DSF(I)*PHIENR(I)*DEW(I))**2
      PWFNT(I)=(PHIENR(I)*DFF(I)*FNCT(I)*DEW(I))**2
      PWSUM(I)=PWPHI(I)+PWFNT(I)
      SUM=PWSUM(I)+SUM
220   CONTINUE
      VFFNCT=SUM**0.5

C
C     SET UP OF DE FOR DELTA E IN THE NORMALIZING INTEGRAL PORTION
C     OF THE EXPECTED VARIATION IN THE WEIGHTED FUNCTION
C

      IF(INPUT2 .EQ. 0)GO TO 216
      IC=IUBFGN-1
      DO 228 I=1,IC
      DEN(I)=0.0
228   CONTINUE
      IC=NROW-ILBFGN
      DO 221 I=1,IC
      L=(NROW+1)-I
      DEN(L)=0.0
221   CONTINUE
      DO 222 I=IUBFGN,ILBFGN
      DEN(I)=DE(I)
222   CONTINUE
      IF(IUBFGN .EQ. IUBGNI)GO TO 234
      DEN(IUBFGN)=DEN(IUBFGN)*UBFEN
234   IF(ILBFGN .EQ. ILBGNI)GO TO 235
      DEN(ILBFGN)=DEN(ILBFGNI)*LBFEN
235   CONTINUE

C
C     COMPUTE EXPECTED VARIATION IN WEIGHTED FUNCTION
C

      SUM=0.0
      DO 223 I=1, NROW
      TERM1(I)=FNCT(I)*DEW(I)/NINT
      TERM2(I)=WINT*DEN(I)/(NINT**2)
      XXX=(TERM1(I)-TERM2(I))
      PWFPHI(I)=(XXX*PHIENR(I)*DSF(I))**2
      TERM3(I)=PHIENR(I)*DEW(I)/NINT
      PWFFNT(I)=(TERM3(I)*FNCT(I)*DFF(I))**2
      SUM=PWFPHI(I)+PWFFNT(I)+SUM
223   CONTINUE
      VWFNT=SUM**0.5
216   CONTINUE

C
C     OPEN OUTPUT FILE
C

      OPEN(UNIT=1,NAME=FILE4,TYPE='NEW')

C
C     OUTPUT ALL DATA
C

      WRITE(1,1124)
```

```
      TYPE 1124
1124  FORMAT(' GROUP  ENERGY      PHI        DELETA      FUNCTION ')
      DO 129 I=1,NROW
      WRITE(1,1125)I,ENGP(I),PHIT(I),DELTA(I),FNCT(I)
      TYPE 1125, I,ENGP(I),PHIT(I),DELTA(I),FNCT(I)
1125  FORMAT(I5,4E11.4)
129   CONTINUE
      WRITE(1,1126)
      TYPE 1126
1126  FORMAT(' GROUP  ENERGY  FRACTION NUMERATOR    FRACTION DENOMINAT'
     1 'OR')
      DO 130 I=1,NROW
      WRITE(1,1127) IGROUP(I),ENGP(I),FRCTW(I),FRCTN(I)
      TYPE 1127,IGROUP(I),ENGP(I),FRCTW(I),FRCTN(I)
1127  FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
130   CONTINUE
      TYPE 1128
      WRITE(1,1128)
1128  FORMAT(20X,' NUMERATOR INTEGRAL  DENOMINATOR INTEGRAL')
      WRITE(1,1129) IUBGWI,IUBGNI
      TYPE 1129, IUBGWI,IUBGNI
1129  FORMAT(' UPPER BOUND GROUP  ',10X,I3,18X,I3)
      WRITE(1,1130) ILBGWI,ILBGNI
      TYPE 1130,ILBGWI,ILBGNI
1130  FORMAT(' LOWER BOUND GROUP  ',10X,I3,18X,I3)
      WRITE(1,1131)IUBFGW,IUBFGN
      TYPE 1131,IUBFGW,IUBFGN
1131  FORMAT(' FRACTION GROUP UB  ',10X,I3,18X,I3)
      WRITE(1,1132)ILBFGW,ILBFGN
      TYPE 1132, ILBFGW,ILBFGN
1132  FORMAT(' FRACTION GROUP LB  ',10X,I3,18X,I3)
      WRITE(1,1133)UBFWI,UBFNI
      TYPE  1133,UBFWI,UBFNI
1133  FORMAT(' UPPER BOUND FRACTION',9X,E11.3,11X,E11.3)
      TYPE 1134, LBFWI,LBFNI
      WRITE(1,1134) LBFWI,LBFNI
1134  FORMAT(' LOWER BOUND FRACTION',9X,E11.3,11X,E11.3)
      TYPE 1135, DUBWI,DUBNI
      WRITE(1,1135) DUBWI,DUBNI
1135  FORMAT(' UPPER BOUND ',16X,E11.3,11X,E11.3)
      TYPE 1136, DLBWI,DLBNI
      WRITE(1,1136)DLBWI,DLBNI
1136  FORMAT(' LOWER BOUND',16X,E11.3,11X,E11.3)
      TYPE 1137, WINT,NINT
      WRITE(1,1137) WINT,NINT
1137  FORMAT(' INTEGRAL ',20X,E11.3,11X,E11.3)
      TYPE 1138, FNCEFF
      WRITE(1,1138) FNCEFF
1138  FORMAT(' WEIGHTED FUNCTION ',17X,E11.3)
      TYPE 1208,VFFNCT
      WRITE(1,1208),VFFNCT
1208  FORMAT(' VARIATION ON FOLDED FUNCTION ',9X,E11.3)
      TYPE 1209, VWFNT
      WRITE(1,1209),VWFNT
1209  FORMAT(' VARIATION OF WEIGHTED FUNCTION ',7X,E11.3)
      TYPE 1210
      WRITE(1,1210)
1210  FORMAT(' GROUP    DELTA FNCT    DELTA SPECT      PHIENR')
      DO 224 I=1,NROW
      TYPE 1211,IGROUP(I),DFF(I),DSF(I),PHIENR(I)
      WRITE(1,1211),IGROUP(I),DFF(I),DSF(I),PHIENR(I)
1211  FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
```

```
224     CONTINUE
        TYPE 1212
        WRITE(1,1212)
1212    FORMAT(' GROUP        DEW          PWPHI         PWFNT')
        DO 225 I=1,NROW
        TYPE 1213,IGROUP(I),DEW(I),PWPHI(I),PWFNT(I)
        WRITE(1,1213),IGROUP(I),DEW(I),PWPHI(I),PWFNT(I)
1213    FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
225     CONTINUE
        TYPE 1214
        WRITE(1,1214)
1214    FORMAT(' GROUP        TERM1         TERM2         TERM3')
        DO 226 I=1,NROW
        TYPE 1215,IGROUP(I),TERM1(I),TERM2(I),TERM3(I)
        WRITE(1,1215),IGROUP(I),TERM1(I),TERM2(I),TERM3(I)
1215    FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
226     CONTINUE
        TYPE 1216
        WRITE(1,1216)
1216    FORMAT(' GROUP        DEN          PWFPHI        PWFFNT')
        DO 227 I=1,NROW
        TYPE 1217,IGROUP(I),DEN(I),PWFPHI(I),PWFFNT(I)
        WRITE(1,1217),IGROUP(I),DEN(I),PWFPHI(I),PWFFNT(I)
1217    FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
227     CONTINUE
        CLOSE(UNIT=1)
140     CONTINUE
        STOP
        END

C
C       SUBROUTINE UPBND
C

        SUBROUTINE UPBNL

C
C       THIS SUBROUTINE CALCULATES THE UPPER BOUND ON INTEGRALS IN WHICH
C       THE INPUT DATA  IS IN GROUPS BUT THE LIMIT IS BETWEEN THE
C       BOUNDS OF A GROUP
C

        COMMON IGP,DUB,E(38),IUBG,IUBFG,UBF
        COMMON DLB,ILBG,ILBFG,XLBF,UBFE,XLBFE

C
C       IGP = NUMBER OF GROUPS
C       DUB = DESIRED UPPER BOUND
C       E(38) = ENERGY GROUP SPECTRUM (UPPER ENERGY BOUNDS)
C       IUBG = UPPER BOUND GROUP
C       IUFBG = UPPER BOUND FRACTIONAL GROUP
C       UBF = UPPER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
C       UBFE= UPPER BOUND FRACTION OF ENERGY
C       DLB = DESIRED LOWER BOUND
C       ILBG = LOWER BOUND GROUP
C       ILBFG = LOWER BOUND FRACTIONAL GROUP
C       XLBF = LOWER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
C       XLBFE = LOWER BOUND FRACTION OF ENERGY
C
C
C       LOOK TO SEE IF UPPER BOUND LIES ON A BOUNDARY
C
```

```
      ISET=0
      DO 1 I=1,IGP
      ISET=ISET+1
      IF(DUB .EQ. E(I)) GO TO 100
1     CONTINUE
      GO TO 101
100   IUBG=ISET
      IUBFG=ISET
      UBF=0.0
      GO TO 105

C
C     IF UPPER BOUND LIES BETWEEN GROUPS, FIND FRACTION OF GROUP
C


101   ISET=0
      DO 103 I=1,IGP+1
      ISET=ISET+1
      IF(DUB .GT. E(I)) GO TO 104
103   CONTINUE
104   IUBG=ISET
      IUBFG=ISET-1
      DELTAE=E(ISET-1)-E(ISET)
C
C     NOTE: E(ISET-1) IS THE HIGHER ENERGY
C
      PARTE=DUB-E(ISET)
      X1=E(ISET-1)
      X2=E(ISET)
      XDELTA=ALOG(X1)-ALOG(X2)
      XPART=ALOG(DUB)-ALOG(X2)
      UBF=SQRT((XPART/XDELTA)**2)
      UBFE=PARTE/DELTAE
C
C     NOTE: IF DUB=E(ISET) THEN DUB-E(ISET)=C AND UBF=0
C
105   TYPE 1002
1002  FORMAT(' TEMP AT SUBROUTINE FORMAT 1002')
      TYPE 1000
1000  FORMAT(' IUBG,IUBFG,UBF,DELTAE,PARTE')
      TYPE 1001,IUBG,IUBFG,UBF,DELTAE,PARTE
1001  FORMAT(2I5,3E12.4)
      RETURN
      END

C
C     SUBROUTINE LOWBND
C

      SUBROUTINE LOWBND

C
C     THIS SUBROUTINE IS FOR CALCULATING THE LOWER BOUND ON INTEGRALS
C     WHERE THE INPUT DATA IS IN GROUPS AND THE CHOSEN BOUND IS BETWEEN
C     THE BOUNDS OF A GROUP
C

      COMMON IGP,DUB,E(38),IUBG,IUBFG,UBF
      COMMON DLB,ILBG,ILBFG,XLBF,UBFE,XLBFE
```

```
C
C       IGP = NUMBER OF GROUPS
C       DUB = DESIRED UPPER BOUND
C       E(38) = ENERGY GROUP SPECTRUM (UPPER ENERGY BOUNDS)
C       IUBG = UPPER BOUND GROUP
C       IUFBG = UPPER BOUND FRACTIONAL GROUP
C       UBF = UPPER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
C       UBFE=UPPER BOUND FRACTION OF ENERGY
C       DLB = DESIRED LOWER BOUND
C       ILBG = LOWER BOUND GROUP
C       ILBFG = LOWER BOUND FRACTIONAL GROUP
C       XLBF = LOWER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
C       XLBFE=LOWER BOUND FRACTION OF ENERGY
C
C
C       CHECK TO SEE IF THE LOWER BOUNDRY LIES ON A BOUNDRY
C

        ISET=0
        DO 1 I=1,IGP+1
        ISET=ISET+1
        IF(DLB .EQ. E(I))GO TO 100
1       CONTINUE
        GO TO 101
100     ILBG=ISET-1
        ILBFG=ISET-1
        XLBF=0.0
        GO TO 105

C
C       IF LOWER BOUND LIES IN A BOUNDRY OF A GROUP FIND THE FRACTION OF
C       OF THAT GROUP
C

101     ISET=0
        DO 103 I=1, IGP
        IF(DLB .GT. E(I))GO TO 104
        ISET=ISET+1
103     CONTINUE
104     ILBG=ISET-1
        ILBFG=ISET
        DELTAE=E(ISET-1)-E(ISET)
        PARTE=E(ISET-1)-DLB
        X1=E(ISET-1)
        X2=E(ISET)
        XDELTA=ALOG(X1)-ALOG(X2)
        XPART=ALOG(X1)-ALOG(DLB)
        XLBF=SQRT((XPART/XDELTA)**2)
        XLBFE=PARTE/DELTAE
105     TYPE 1002
1002    FORMAT(' TEMP FROM SUBROUTINE LOWBND FORMAT 1002')
        TYPE 1000
1000    FORMAT(' ILBG,ILBFG,XLBF,DELTAE,PARTE')
        TYPE 1001, ILBG,ILBFG,XLBF,DELTAE,PARTE
1001    FORMAT(2I5,3E12.4)
        RETURN
        END
```

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