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 ± 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
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.
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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 \cdot 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

<table>
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<tr>
<th>Room Distance to Core (cm)</th>
<th>Configuration</th>
<th>Average Energy (MeV) Neutron</th>
<th>Gamma</th>
<th>Neutron Spectra*</th>
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<td>1.04</td>
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<td></td>
<td>100</td>
<td>5 cm Pb and exercise wheel</td>
<td>1.05</td>
<td>1.36</td>
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<td>100</td>
<td>15 cm Pb and cylindrical phantom</td>
<td>0.55</td>
<td>1.13</td>
</tr>
</tbody>
</table>

* 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 a 14-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):

\[
E_{\text{avg}} = \frac{\int_{0.4 \text{eV}}^{\infty} E \phi(E) \, dE}{\int_{0.4 \text{eV}}^{\infty} \phi(E) \, dE} = \frac{\sum_{i=1}^{21 \text{ or } 36} E_i \phi_i \Delta E_i}{\sum_{i=1}^{21 \text{ or } 36} \phi_i \Delta E_i}
\]

where \(E\) and \(E_i\) represent neutron or gamma ray energy, and \(\phi(E)\) and \(\phi_i\) 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:

\[
\text{SAKF} = \frac{\int_{0}^{\infty} K(E) \Phi(E) \, dE}{\int_{0}^{\infty} \Phi(E) \, dE} = \sum_{i=1}^{21 \text{ or } 37} K_i \phi_i \Delta E_i
\]

\[\text{Eq. 2}\]

where SAKF is the spectrum averaged kerma factor, \(K(E)\) and \(K_i\) are the energy-dependent kerma factors, \(\Phi(E)\) and \(\phi_i\) 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*

<table>
<thead>
<tr>
<th>Room</th>
<th>Distance to Core (cm)</th>
<th>Configuration</th>
<th>MeV_{eff} (tissue)</th>
<th>(K_T)^G</th>
<th>(K_T/K_C)^G</th>
<th>(K_T/K_{Mg})^G</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>50</td>
<td>Unshielded</td>
<td>1.0</td>
<td>4.92</td>
<td>1.128</td>
<td>1.108</td>
</tr>
<tr>
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<td>100</td>
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<td>1.0</td>
<td>4.88</td>
<td>1.130</td>
<td>1.107</td>
</tr>
<tr>
<td>1</td>
<td>200</td>
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<td>4.47</td>
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<td>1.139</td>
<td>1.073</td>
</tr>
<tr>
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<td>2.93</td>
<td>1.149</td>
<td>1.054</td>
</tr>
<tr>
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<td>5.00</td>
<td>1.129</td>
<td>1.109</td>
</tr>
<tr>
<td>2</td>
<td>100</td>
<td>Unshielded</td>
<td>1.0</td>
<td>4.89</td>
<td>1.129</td>
<td>1.106</td>
</tr>
<tr>
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<td>200</td>
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<td>NA</td>
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<td>3.54</td>
<td>1.138</td>
<td>1.089</td>
</tr>
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<td>30 cm H_2O</td>
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<td>3.99</td>
<td>1.137</td>
<td>1.093</td>
</tr>
<tr>
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<td>100</td>
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<td>1.110</td>
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<tr>
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<td>5 cm Pb and exercise wheel</td>
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Reference Spectra

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<td>Cobalt-60</td>
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</table>

* (K_T)^G units are 10^{-10} rad \cdot 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.

\[
(K_T/K_C)_G = 1.133 \pm 1.0\%
\]

\[
(K_T/K_{Mg})_G = 1.09 \pm 3.1\%
\]
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

<table>
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<tr>
<th>Distance to Core (cm)</th>
<th>Configuration</th>
<th>SAKF (10^{-10} \text{rad cm}^2)</th>
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<tr>
<td>1</td>
<td>30 cm H_2O</td>
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</tr>
<tr>
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<td>4.51</td>
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Table 4 presents the reactor neutron SAKF's for three tissue-like 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

<table>
<thead>
<tr>
<th>Distance to Core (cm)</th>
<th>Configuration</th>
<th>MeV&lt;sub&gt;eff&lt;/sub&gt;</th>
<th>Muscle&lt;sup&gt;△&lt;/sup&gt; (ICRU)</th>
<th>Tissue&lt;sup&gt;△&lt;/sup&gt; (ICRU)</th>
<th>Tissue&lt;sup&gt;△&lt;/sup&gt; (DLC-31)</th>
<th>A-150 Plastic&lt;sup&gt;△&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>100</td>
<td>Unshielded</td>
<td>0.8</td>
<td>19.8</td>
<td>19.6</td>
<td>19.6</td>
<td>20.0</td>
</tr>
<tr>
<td>200</td>
<td>Unshielded</td>
<td>0.7</td>
<td>20.3</td>
<td>20.2</td>
<td>20.0</td>
<td>20.6</td>
</tr>
<tr>
<td>300</td>
<td>Unshielded</td>
<td>0.6</td>
<td>17.6</td>
<td>17.4</td>
<td>17.4</td>
<td>17.8</td>
</tr>
<tr>
<td>400</td>
<td>Unshielded</td>
<td>0.6</td>
<td>13.6</td>
<td>13.4</td>
<td>13.4</td>
<td>13.7</td>
</tr>
<tr>
<td>500</td>
<td>Unshielded</td>
<td>0.3</td>
<td>12.1</td>
<td>11.9</td>
<td>11.9</td>
<td>12.2</td>
</tr>
<tr>
<td>100</td>
<td>Unshielded</td>
<td>0.4</td>
<td>14.2</td>
<td>14.0</td>
<td>14.0</td>
<td>14.4</td>
</tr>
<tr>
<td>100</td>
<td>Unshielded</td>
<td>0.4</td>
<td>14.4</td>
<td>14.2</td>
<td>14.2</td>
<td>14.5</td>
</tr>
<tr>
<td>100</td>
<td>Unshielded</td>
<td>0.3</td>
<td>12.1</td>
<td>11.9</td>
<td>11.9</td>
<td>12.2</td>
</tr>
<tr>
<td>100</td>
<td>Unshielded</td>
<td>0.2</td>
<td>10.6</td>
<td>10.5</td>
<td>10.5</td>
<td>10.7</td>
</tr>
<tr>
<td>NA</td>
<td>NA</td>
<td>0.2</td>
<td>3.83</td>
<td>3.70</td>
<td>3.75</td>
<td>3.87</td>
</tr>
</tbody>
</table>

Reference Spectra

- DLC-31 fission: 0.8 20.4 20.1 19.4 29.4
- 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: 14.1 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): 10.0% H, 24.9% C, 2.9% N, 40.0% O, 0.20% Na, 0.20% 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 CO₂ and Ar, the SAKF ratios can be summarized as 9.78 ± 10% and 70.7 ± 27%, respectively. The variability among the CO₂ and Ar SAKF ratios indicates that individually tabulated values should be used for each reactor configuration.
Table 5. Reactor Neutron SAKF Ratios

<table>
<thead>
<tr>
<th>Distance to Core Room (cm)</th>
<th>Configuration</th>
<th>((K_T/K_{TE \text{ gas}})_N^*)</th>
<th>((K_T/K_{CO_2}))</th>
<th>((K_T/K_{Ar}))</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Unshielded</td>
<td>0.980</td>
<td>9.57</td>
<td>66.0</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>0.979</td>
<td>9.50</td>
<td>64.3</td>
</tr>
<tr>
<td></td>
<td>200</td>
<td>0.983</td>
<td>9.62</td>
<td>66.7</td>
</tr>
<tr>
<td></td>
<td>300</td>
<td>0.981</td>
<td>9.68</td>
<td>67.5</td>
</tr>
<tr>
<td></td>
<td>400</td>
<td>0.986</td>
<td>9.78</td>
<td>69.0</td>
</tr>
<tr>
<td></td>
<td>500</td>
<td>0.984</td>
<td>9.84</td>
<td>70.3</td>
</tr>
<tr>
<td>2</td>
<td>Unshielded</td>
<td>0.979</td>
<td>9.59</td>
<td>66.0</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>0.986</td>
<td>9.66</td>
<td>65.8</td>
</tr>
<tr>
<td></td>
<td>200</td>
<td>0.984</td>
<td>9.68</td>
<td>67.2</td>
</tr>
<tr>
<td></td>
<td>300</td>
<td>0.981</td>
<td>9.81</td>
<td>69.3</td>
</tr>
<tr>
<td>NA</td>
<td>Pneumatic tubes</td>
<td>0.982</td>
<td>10.21</td>
<td>73.2</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>7.5 cm (H_2O)</td>
<td>0.979</td>
<td>9.26</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>30 cm (H_2O)</td>
<td>0.983</td>
<td>8.49</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>5 cm Pb</td>
<td>0.986</td>
<td>9.67</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>5 cm Pb and exercise wheel</td>
<td>0.984</td>
<td>9.92</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>15 cm Pb</td>
<td>0.983</td>
<td>10.18</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>15 cm Pb and cave</td>
<td>0.986</td>
<td>10.38</td>
</tr>
<tr>
<td></td>
<td>100</td>
<td>15 cm Pb and cylindrical phantom</td>
<td>0.986</td>
<td>11.00</td>
</tr>
</tbody>
</table>

Reference Spectra

- DLC-31 fission: 0.986, 9.66, 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 \((K_T/K_{TE \text{ gas}})_N^*\) 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

<table>
<thead>
<tr>
<th>Room</th>
<th>Distance to Core (cm)</th>
<th>Configuration</th>
<th>Neutron Spectrum</th>
<th>Difference (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td>3-D</td>
<td>MEAS</td>
</tr>
<tr>
<td>$K_{T}/K_{\text{Ar}}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>Unshielded</td>
<td>64.3</td>
<td>67.5</td>
</tr>
<tr>
<td>2</td>
<td>100</td>
<td>Unshielded</td>
<td>65.8</td>
<td>67.7</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>5 cm Pb and exercise wheel</td>
<td>73.6</td>
<td>74.6</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>15 cm Pb and cave</td>
<td>92.5</td>
<td>91.4</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>15 cm Pb and cylindrical phantom</td>
<td>91.1</td>
<td>84.3</td>
</tr>
<tr>
<td>$K_{T}/K_{\text{CO}_2}$</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>Unshielded</td>
<td>9.50</td>
<td>9.66</td>
</tr>
<tr>
<td>2</td>
<td>100</td>
<td>Unshielded</td>
<td>9.66</td>
<td>9.73</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>5 cm Pb and exercise wheel</td>
<td>9.92</td>
<td>10.00</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>15 cm Pb and cave</td>
<td>10.38</td>
<td>10.30</td>
</tr>
<tr>
<td>1</td>
<td>100</td>
<td>15 cm Pb and cylindrical phantom</td>
<td>11.0</td>
<td>10.56</td>
</tr>
</tbody>
</table>

### 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 $\text{CO}_2$ 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 CO₂ 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 AFQRI 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
THIS PROGRAM WILL COMPUTE TWO SPECTRUM DEPENDENT FUNCTIONS:
A SPECTRUM WEIGHTED FUNCTION AND A SPECTRUM FOLDED FUNCTION,
AND THE ESTIMATED VARIATION OF EACH

AUTHOR: KEF FERLIC CIRCA 1981 OR 82

THIS PROGRAM IS FOR EVALUATING THE INTEGRALS (BY SUMMATION)
NECESSARY FOR SPECTRUM AVERAGE QUANTITIES

NOTES
1. SPECTRUM INPUT DATA CAN BE A FROM AND BACK SPECTRUM
2. SPECTRUM INPUT DATA CAN BE LETHARGY, PHI(E)DE OR PHI(E)
3. THE DELTA SPECTRUM IS THE FRACTIONAL CHANGE EXPECTED IN NEUTRON
   POPULATION FOR A PARTICULAR ENERGY GROUP
4. THE DELTA FUNCTION IS THE FRACTIONAL CHANGE EXPECTED IN THE
   FUNCTION PER ENERGY GROUP
5. ENERGY GROUP BOUNDS ARE THE UPPER BOUNDS
6. ENERGY GROUP 1 IS THE MAXIMUM ENERGY GROUP, I.E. GROUP 1 UPPFR
   BOUND IS 19.6 MEV

HIGHEST CONTROL STATEMENT IS 238
HIGHEST FORMAT STATEMENT IS 1217

BYTE FILE1(3C) 'FILE1-FLUX SPECTRUM FILE
BYTE FILE2(3C) 'FILE2-ENERGY GROUP FILE, UPPER BOUNDS
BYTE FILE3(3C) 'FILE3-FUNCTION FILE BY GROUP
BYTE FILE4(3C) 'FILE4-OUTPUT FILE
BYTE FILE5(3C) 'FILE5-DELTA SPECTRUM FILE
BYTE FILE6(3C) '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 NUMBERS
INTEGER NROW 'NUMBER OF GROUPS NUMBER OF INPUT RCWC
INTEGER ICOL 'NUMBER OF COLUMNS IN INPUT DATA
INTEGER ICNT 'COUNTER CF FOR FILE NAMES
INTEGER NSPT 'TYPE OF FLUX INPUT IE ENERGY,LETHARGY
INTEGER NTYPE 'TYPE OF INPUT SPECTRUM TOTAL VS FRONT * PACK
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 DUBBI 'DESIRED UPPER BOUND NORMALIZING INTEGRAL
REAL DLBNI 'DESIRED LOWER BOUND NORMALIZING INTEGRAL
INTEGER IUBGW 'UPPER BOUND GROUP WEIGHTING INTEGRAL
REAL UBFWI 'UPPER BOUND FRACTION WEIGHTING INTEGRAL
INTEGER IUBFGW 'UPPER BOUND FRACTIONAL GROUP WEIGHTING INTEGRAL
INTEGER IUBGN1 'UPPER BOUND FRACTIONAL GROUP NORMALIZING INTEGRAL
INTEGER IUBFGN 'UPPER BOUND FRACTIONAL GROUP NORMALIZING INT
INTEGER ILBGW 'LOWER BOUND GROUP WEIGHTING INTEGRAL
REAL LBFWI 'LOWER BOUND FRACTION WEIGHTING INTEGRAL
INTEGER ILBFGW 'LOWER BOUND FRACTIONAL GROUP WEIGHTING INTEGRAL
INTEGER ILBGN1 'LOWER BOUND FRACTIONAL GROUP NORMALIZING INTEGRAL
REAL LBFNI 'LOWER BOUND FRACTIONAL GROUP NORMALIZING INTEGRAL

15
INTEGER ILBGNI  !LOWER BOUND GROUP NORMALIZING INTEGRAL
REAL WINT  !INTEGRAL OF THE WEIGHTING INTEGRAL
REAL MINT  !INTEGRAL OF THE NORMALIZING INTEGRAL
REAL PCEFF  !EFFECTIVE VALUE OF FUNCTION
INTEGER NROW1  !ROWS IN INPUT SPECTRUM FILE
INTEGER NROW2  !ROWS IN INPUT ENERGY 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 FRCTN(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 CCNTRL
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)  !DELTA E FOR NOR INT IN VARIATION CALCULATION
DIMENSION PMPHI(38)  !PARTIAL WGT WITH RESPECT TO PHI
DIMENSION PWFNT(38)  !PARTIAL WGT WITH RESPECT TO THE FUNCTION
DIMENSION PWPHI(38)  !PARTIAL OF WGT FUNCT WITH RESPECT TO PHI
DIMENSION PWFFNT(38)  !PARTIAL OF WGT FUNCT WITH RESPECT TO THE FUNCTION
REAL VWFNT  !VARIATION OF WEIGHTED FUNCTION

COMMON STATEMENTS FOR SUBROUTINES UPBND AND LWBND
COMMON IXGP, XDUB, XE(38), IXUBG, IXUBFG, XUBF
COMMON XDLB, IXLBG, IXLBFG, XLBFG, XUBFE

DETERMINATION OF CALCULATIONS TO BE PERFORMED

TYPE 120C
120C FORMAT(' DO YOU WANT A SPECTRUM AVERAGED FUNCTION? Y=1, N=0, CR?')
      ACCEPT 1201, SAF
1201 FORMAT(1X)
      IF(SAF .EQ. 0) GO TO 26C
      TYPE 120C2
120C2 FORMAT(' DO YOU WANT THE ESTIMATED VARIATION? Y=1, N=0, CR?')
      ACCEPT 120C1, VSAFE
26C CONTINUE
TYPE 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
FORMAT( 'DO YOU WANT THE ESTIMATED VARIATION? YES=1, NO=0 OR CR')
ACCEPT 1201,VSSF
201 CONTINUE
IF(SAF .NE. SFF)GO TO 203
IF(SAF .EQ. 0)GO TO 202
GO TO 203
202 TYPE 1205
FORMAT( 'WHAT DO YOU WANT?? BYE, BYE')
GO TO 140
203 CONTINUE

PROGRAM CONTROL FOR CALCULATIONS, DETERMINATION OF ACTUAL INPUT DATA

INPUT1 DELTA FILES INPUT,
INPUT2 BOUNDS ON NORMALIZING INTEGRAL INPUT

IF(SAF .EQ. C)GO TO 2C4
INPUT1=0
INPUT2=1
IF(VSAF .EQ.C)GO TO 2C4
INPUT1=1
2C4 CONTINUE
IF(SFF .EQ. C)GO TO 2C5
INPUT1=C
INPUT2=1
IF(VSFF .EQ.O)GO TO 205
INPUT1=1
2C5 CONTINUE
IF(INPUT2 .EQ. INPUT3)GO TO 229
INPUT2=1
229 CONTINUE

IDENTIFY NUMBER OF GROUPS

TYPE 1116
FORMAT( 'WHAT IS THE NUMBER OF GROUPS IN SPECTRUM<13>')
ACCEPT 1006,NROW

SET LOW BOUND ON LOWEST ENERGY GROUP

IF(NROW .EQ. 37)GO TO 119
IF(NROW .EQ. 21)GO TO 12C
TYPE 112C
1120 FORMAT( 'WHAT IS LOWER BOUND (IN EV) ON THE LOWEST ENERGY GROUP?<1.4>')
ACCEPT 1121,ENGP(NROW-1)
1121 ENGP(NROW-1)=1.E-5 'IN EV
GO TO 121
119
FILE1 NAME: SPECTRUM FILE

TYPE 1000
1000 FORMAT( 'ENERGY SPECTRUM FILE NAME <FILENAME.TYP>:' )
ACCEPT 1001, ICNT, FILE1
1001 FORMAT(Q,3OA1)
FILE1(ICNT+1)=0
TYPE 1005
1005 FORMAT( 'SPECTRUM INPUT: FRONT = BACK = O, TOTAL = 1' )
ACCEPT 1006, NTYP
1006 FORMAT(13)
TYPE 1115
1115 FORMAT( 'IS SPECTRUM INPUT: PHI(U)*1, PHI(E)DE=0, PHI(E) = -1' )
ACCEPT 1116, NSPT

FILE2 NAME: ENERGY GROUP UPPER BOUND FILE

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

FILE3 NAME: FUNCTION GROUP FILE

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

FILE4 NAME: OUTPUT FILE NAME

TYPE 1196
1196 FORMAT( 'ENTER OUTPUT FILE NAME:<FILENAME.TYP>' )
ACCEPT 1001, ICNT, FILE4
FILE4(ICNT+1)=C

FILE5 NAME: DELTA SPECTRUM FILE

IF( INPUT1 .EQ. C ) GO TO 206
TYPE 1266
1266 FORMAT( 'ENTER DELTA SPECTRUM FILE NAME: <FILENAME.TYP>: ' )
ACCEPT 1261, ICNT, FILE5
FILE5(ICNT+1)=C

FILE6 NAME: DELTA FUNCTION FILE

TYPE 1267
1207 FORMAT(' ENTER DELTA FUNCTION FILE NAME <FILENAME.XY>&: ')
ACCEPT INCF INF FILE6
FILE6(INCF+1)=0
206 CONTINUE

C
C EXTRACT DATA FROM SPECTRUM FILE: FILE1

OPEN(UNIT=1,NAME=FILE1,TYPE='OLD')
READ(1,1007) ICOL,NROW
1007 FORMAT(1I4)
IF(NROW .LT. NROW)GO TO 112
IF(NTYPE .LT. ICOL)GO TO 100
DO 101 I=1,NROW
READ(1,1008) IGROUP(I),PHIT(I)
1008 FORMAT(13.E12.4)
101 CONTINUE
DO 102
102 CONTINUE
CLOSE(UNIT=1)

C
C EXTRACT DATA FROM ENERGY GROUP FILE

OPEN(UNIT=1,NAME=FILE2,TYPE='OLD')
READ(1,1007) ICOL,NROW2
IF(NROW2 .LT. NROW)GO TO 114
DO 113 I=1,NROW2
READ(1,1009) IGROUP(I),ENGP(I)
1009 FORMAT(13.E12.4)
113 CONTINUE
CLOSE(UNIT=1)

C
C EXTRACT DATA FROM FUNCTION FILE: FILE3

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

C
C EXTRACT DATA FROM DELTA SPECTRUM FILE: FILE6

IF(INCF .LE. ICOL)GO TO 147
OPEN(UNIT=1,NAME=FILE5,TYPE='OLD')
READ(1,1007) ICOL,NROW5
IF(NROW5 .LT. NROW5)GO TO 146
DO 149 I=1,NROW5
READ(1,1008), IGROUP(I),DFP(I)
149 CONTINUE
CLOSE(UNIT=1);

19
EXTRACT DATA FROM DELTA FUNCTION FILE: FILE6

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),DFP(I)
CONTINUE
CLOSE(UNIT=1)
GO TO 141

CHECK FOR PROPER NUMBER OF GROUPS IN EACH INPUT FILE: NEGATIVE ANSWF

112 TYPE 1117,NROW1
GO TO 113
114 TYPE 1117,NROW2
GO TO 113
115 TYPE 1117,NROW3
GO TO 113
216 TYPE 1117,NROW5
GO TO 113
21C TYPE 1117,NROW6
1117 FORMAT(' ACTUAL GROUPS IN FILE',I3)
113 TYPE 1199,NROW
1159 FORMAT(' DOES NOT EQUAL IDENTIFIED SPECTRUM GROUPS:',I3)
CLOSE(UNIT=1)
GO TO 14C

CALCULATE TOTAL FLUX IF FRONT AND BACK

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

CALCULATE DU OR DE FOR THE SPECTRUM

OLD LINE IF(NSPT) 116,117,11E

CHANGE IN LETHARGY/ENERGY

116 DC 1C6 I=1,NROW
DU(I)-LOG(ENGF(I)/ENGF(I+1))
CONTINUE
OLD LINE GO TO 117
116 DC 1C6 I=1,NROW
DE(I)=ENGF(I)-ENGF(I+1)
CONTINUE
117 CONTINUE
ACCEPT BOUNDS ON WEIGHTING AND NORMALIZING INTEGRAL

TYPE 1010
FORMAT(' WHAT IS UPPER BOUND OF WEIGHTING INTEGRAL IN EV:
1 '<E10.4>'

TYPE 1122
FORMAT(' NEUTRON 37 GROUP MAX IS 19.6E+6 EV; PHOTON 21 GROUP MAX
1 'IS 14.0E+6 EV')
ACCEPT 1011,DUBWI

TYPE 1011
FORMAT(E10.4)

TYPE 1012
FORMAT(' WHAT IS LOWER BOUND OF WEIGHTING INTEGRAL IN EV:
1 '<E10.4>'

TYPE 1123
FORMAT(' NEUTRON 37 GROUP MIN IS 1.0E-5 EV; PHOTON 21 GROUP MAX
1 'IS 1.0E-4 EV')
ACCEPT 1011,DLBWI

ACCEPT BOUNDS ON NORMALIZING INTEGRAL

IF(INPUT2 .EQ. C)GO TO 212

TYPE 1C13
FORMAT(' WHAT IS UPPER BOUND ON NORMALIZING INTEGRAL IN EV:
1 '<E10.4>'

TYPE 1122
ACCEPT 1011,DUBNI

TYPE 1014
FORMAT(' WHAT IS LOWER BOUND ON NORMALIZING INTEGRAL IN EV:
1 '<E10.4>'

TYPE 1123
ACCEPT 1011,DLBNI

212 CONTINUE

SET UPPER BOUND ON WEIGHTING INTEGRAL

IXGP=NROW+1
XIUB=DUBNI
DO 151 I=1, NROW+1
XE(I)=ENGP(I)
151 CONTINUE
CALL UPBND
IXUBG=IXUBG
UXFW=UXUF
UXFCW=UXUBFG
UXFEW=UXUFE

SET UPPER BOUND ON NORMALIZING INTEGRAL

IF(INPUT2 .EQ. C)GO TO 213
XUB=DUBNI
CALL UBNI
IUBFNC=IXUBFG
UXFNI=UXUF
IUBNI=IXUBG
UXFEN=UXUFE

21
CONTINUE

SET LOWER BOUND WEIGHTING INTEGRAL

XDLB=DLBWI
CALL LOWEND
ILBGI=1XLBG
LBFW=XLBF
ILBFOW=1XLBFG
LBFEW=XLBFEL
LBFEW=XLBFNL

SET LOWER BOUND ON NORMALIZING INTEGRAL

IF(INPUT2 .EQ. 0) GO TO 214
XDLB=DLBWI
CALL LOWEND
ILBFON=1XLBFG
LBFI=XLBF
ILBE=1XLBG
LBFC=XLBF
LBFE=XLBF

CONTINUE

EVALUATION OF WEIGHTING INTEGRAL

DO 110 I=1,NROW
FRCWI(I)=-C.C
FRCWI(I)=-C.C
CONTINUE

DETERMINE DELTA TERMS FOR INTEGRALS I.E. LE OR DU
PLACE ALL DATA IN THE FORM OF PHI/E, IRREGARDLESS OF INPUT

IF(NSPT) 122,123,124
122 DO 125 I=1,NRCW
DELTA(I)=DE(I)
PHI=PHIT(I)
CONTINUE
GO TO 126
123 DO 127 I=1,NROW
DELTA(I)=1.0
PHIENR=PHIT(I)/DE(I)
CONTINUE
GO TO 126
124 DO 126 I=1,NRCW
DELTA(I)=EU(I)
PHIENR(I)*PHIT(I)*EU(I)/DE(I)
126 CONTINUE

EVALUATION OF WEIGHTING/FOLLOWING INTEGRAL

SUM=0.0
DO 150 I=1LBGW1,ILBOW1

22
IF(CTW(I)) = PHIT(I) * DELTA(I) * FNCT(I)
SUM = SUM + CTW(I)
CONTINUE

UTAIL = PHIT(IUBFGW) * DELTA(IUBFGW) * FNCT(IUBFGW) * UBFW
LTAIL = PHIT(ILBFGN) * DELTA(ILBFGN) * FNCT(ILBFGN) * LBFW
IF(IUBGWI .EQ. IUBFGW) GO TO 230
FRCTW(IUBFGW) = UTAIL
230 IF(IULBFGW .EQ. IUBFGW) GO TO 231
FRCTW(IUBFGW) = LTAIL
231 WINT = SUM + UTAIL + LTAIL

C EVALUATION OF NORMALIZING INTEGRAL
C
IF(INPUT2 .EQ. 0) GO TO 215
SUM = 0.0
DO 111 I = IUPGNI, ILBGNI
FRCTN(I) = PHIT(I) * DELTA(I)
SUM = SUM + FRCTN(I)
111 CONTINUE
UTAIL = PHIT(IUBFGN) * DELTA(IUBFGN) * UBFN
LTAIL = PHIT(ILBFGN) * DELTA(ILBFGN) * LBFW
IF(IUBGNI .EQ. IUBFGN) GO TO 232
FRCTN(IUBFGN) = UTAIL
232 IF(IULBGNI .EQ. ILEFGN) GO TO 237
FRCTN(IUBFGN) = LTAIL
233 NINT = SUM + UTAIL + LTAIL
234
C EVALUATE EFFECTIVE FUNCTION
C
FKCEFF = WINT / NINT
CONTINUE

215
C SET OF DE FOR DELTA E IN THE POLLED FUNCTION VARIATION EQUATION
C
IF(INPUT1 .EQ. 0) GO TO 216
IC = IUBFGW - 1
DG = 217, 1 = 1, IC
DEW(I) = 0.0
217 CONTINUE
IC = (NROW - 1) - ILBFGW
DG = 218 I = 1, IC
L = (NROW - 1) - I
DEW(I) = 0.0
216 CONTINUE
DG = 219 I = IUBFGW, ILBFGW
DEW(I) = DE(I)
219 CONTINUE
IF(IUBFGW .EQ. IUBGWI) GO TO 276
LEW(IUBFGW) = DEW(IUBFGW) * UBFW
236 IF(IUBPGW .EQ. IUBGFNI) GO TO 237
DEW(IUBFGW) = DEW(IUBFGW) * LBFW
237 CONTINUE

C COMPUTE ESTIMATED VARIATION IN THE POLLED FUNCTION

23
SUM=0.0
DO 220 I=1,NROW
PWPHI(I)=(FNCT(I)*DSF(I)*PHIENR(I)*DEW(I))**2
PWFFNT(I)=(PHIENR(I)*DFF(I)*FNCT(I)*DEW(I))**2
PWSUM(I)=PWPHI(I)+PWFFNT(I)
SUM=PWSUM(I)+SUM
220 CONTINUE
VFNFCT=SUM**0.5

C 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)=C.0
226 CONTINUE
IC=NROW-1LBFGN
DO 221 I=1,IC
L=(NROW+1)-I
DEN(I)=C.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(ILBFGN)*LBFEN
235 CONTINUE
C C
C COMPUTE EXPECTED VARIATION IN WEIGHTED FUNCTION
C
SUM=C.C
DO 223 I=1, NROW
TERM1(I)=FNCT(I)*DEW(I)/NINT
TERM2(I)=WINT*DEN(I)/(NINT**2)
XXX=(TERM1(I)-TERM2(I))
PWPHI(I)=(XXX*PHIENR(I)*DSF(I))**2
TERM3(I)=PHIENR(I)*DEN(I)/NINT
PWFFNT(I)=(TERM3(I)*FNCT(I)*DFF(I))**2
SUM=PWPHI(I)+PWFFNT(I)+SUM
223 CONTINUE
VFNFNT=SUM**0.5
216 CONTINUE
C C
C OPEN OUTPUT FILE
C
OPEN(UNIT=1,NAME=FILE4,TYPE='NEW')
C C
C OUTPUT ALL DATA
C
WRITE(1,1144)

1144
TYPE 1124
FORMAT(' GROUP ENERGY PHI DELTA FUNCTION ')
DO 129 I=1,NROW
WRITE(1,1125)IGROUP(I),ENG(I),PHI(I),DELTA(I),FNCT(I)
TYPE 1125, I,ENG(I),PHI(I),DELTA(I),FNCT(I)
1126 FORMAT(I5,4E11.4)
129 CONTINUE
WRITE(1,1126)
TYPE 1126
1127 FORMAT(' GROUP ENERGY FRACTION NUMERATOR FRACTION DENOMINATOR ')
'GR')
DO 130 I=1,NROW
WRITE(1,1127)IGROUP(I),ENG(I),FRCTW(I),FRCTN(I)
TYPE 1127,IGROUP(I),ENG(I),FRCTW(I),FRCTN(I)
1128 FORMAT(I5,E11.4,E14,12X,E11.4)
130 CONTINUE
TYPE 1128
WRITE(1,1128)
1129 FORMAT(20X,' NUMERATOR INTEGRAL DENOMINATOR INTEGRAL'
'GR')
WRITE(1,1129)UBFNI,UBFWI
TYPE 1129, UBFNI,UBFWI
1130 FORMAT(' UPPER BOUND GROUP ',10X,I3,16X,I3)
WRITE(1,1130)ILBFWI,ILBFWI
TYPE 1130,ILBFWI,ILBFWI
1131 FORMAT(' LOWER BOUND GROUP ',10X,I3,16X,I3)
WRITE(1,1131)ILBFWI,ILBFWI
TYPE 1131,ILBFWI,ILBFWI
1132 FORMAT(' FRACTION GROUP UB ',10X,I3,16X,I3)
WRITE(1,1132)ILBFWI,ILBFWI
TYPE 1132, ILBFWI,ILBFWI
1133 FORMAT(' FRACTION GROUP LB ',10X,I3,16X,I3)
WRITE(1,1133)UBFNI,UBFNI
TYPE 1133,UBFNI,UBFNI
1134 FORMAT(' UPPER BOUND FRACTION',9X,E11.3,11X,E11.3)
TYPE 1134, LBFNI, LBFWI
WRITE(1,1134)LBFNI, LBFWI
1135 FORMAT(' LOWER BOUND FRACTION',9X,E11.3,11X,E11.3)
TYPE 1135, LBFWI, LBFNI
WRITE(1,1135)LBFWI, LBFNI
1136 FORMAT(' UPPER BOUND ',16X,E11.3,11X,E11.3)
TYPE 1136, LBLNI, LBLNI
WRITE(1,1136)LBLNI, LBLNI
1137 FORMAT(' LOWER BOUND ',16X,E11.3,11X,E11.3)
TYPE 1137, LBLNI, LBLNI
WRITE(1,1137)LBLNI, LBLNI
1138 FORMAT(' INTEGRAL ',20X,E11.3,11X,E11.3)
TYPE 1138, PNCNI, PNCNI
WRITE(1,1138)PNCNI, PNCNI
1139 FORMAT(' WEIGHTED FUNCTION ',17X,E11.3)
TYPE 1208,VFPNI, VFPNI
WRITE(1,1208), VFPNI
1208 FORMAT(' VARIATION ON FOLLOWED FUNCTION ',9X,E11.3)
TYPE 1208, VFPNI
WRITE(1,1208), VFPNI
1209 FORMAT(' VARIATION OF WEIGHTED FUNCTION ',7X,E11.3)
TYPE 1210, VFPNI
WRITE(1,1210)
1210 FORMAT(' GROUP DELTA FNCT DELTA SPECT FHIENR ')
DO 224 I=1,NROW
WRITE(1,1211),IGROUP(I),DFP(I),DSP(I),FHNP(I),FHNSP(I)
TYPE 1211, IGROUP(I),DFP(I),DSP(I),FHNP(I),FHNSP(I)
1211 FORMAT(I5,E11.4,E6X,E11.4,E12X,E11.4)
CONTINUE
TYPE 1212
WRITE(1,1212)

1212 FORMAT(' GROUP UGW PWFHI PWFNT'*)
DO 225 I=1,NROW
 TYPE 1213,IGROUP(I),DEW(I),PWFHI(I),PWFNT(I)
 WRITE(1,1213),IGROUP(I),DEW(I),PWFHI(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 PWFNT'*)
DO 227 I=1,NROW
 TYPE 1217,IGROUP(I),DEN(I),PWFPHI(I),PWFNT(I)
 WRITE(1,1217),IGROUP(I),DEN(I),PWFPHI(I),PWFNT(I)
1217 FORMAT(I5,E11.4,8X,E11.4,12X,E11.4)
227 CONTINUE

CLOSE(UNIT=1)
14L CONTINUE
STOP
END

C SUBROUTINE UPBND
SUBROUTINE UPBND

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

COMMON IGP,DUB,E(38),IUBG,IUBFG,UBF
COMMON DLB,ILBG,ILBFG,UBF,UXEF

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 IUBFG = UPPER BOUND FRACTIONAL GROUP
C UBF = UPPER BOUND FRACTION OF GROUP FOR PHI*LE OR PHI*DU
C UBFE= UPPER BOUND FRACTION OF ENERGY
C DLB = DESIRED LOWER BOUND
C ILB = LOWER BOUND GROUP
C ILBF = LOWER BOUND FRACTIONAL GROUP
C XLF = LOWER BOUND FRACTION OF GROUP FOR PHI*LE OR PHI*DU
C XLEFE = LOWER BOUND FRACTION OF ENERGY

C LOOK TO SEE IF UPPER BOUND LIES ON A BOUNDARY

26
!SUBROUTINE LOWBND

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

COMMON IGP,LUB,E(38),IUBG,IUBFG,UBF
COMMON DLB,ILBC,ILBFG,XMLBF,UBFE,XLPEF

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

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 NOTE: E(ISET-1) IS THE HIGHER ENERGY

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 NOTE: IF DUB.E(ISET) THEN DUB-E(ISET)+C AND UBF=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
ICP = NUMBER OF GROUPS
DUB = DESIRED UPPER BOUND
E(38) = ENERGY GROUP SPECTRUM (UPPER ENERGY BOUNDS)
IUBG = UPPER BOUND GROUP
IUFBG = UPPER BOUND FRACTIONAL GROUP
UBF = UPPER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
UBFE=UPPER BOUND FRACTION OF ENERGY
DLB = DESIRED LOWER BOUND
ILBG = LOWER BOUND GROUP
ILBFG = LOWER BOUND FRACTIONAL GROUP
XLBF = LOWER BOUND FRACTION OF GROUP FOR PHI*DE OR PHI*DU
XLBFE=LOWER BOUND FRACTION OF ENERGY

CHECK TO SEE IF THE LOWER BOUNDARY LIES ON A BOUNDARY

DO 1 I=1,ICP
ISET=ISET+1
IF(DLB .EQ. E(I))GO TO 100
CONTINUE
GO TO 104
100 ILBG=ISET-1
ILBFG=ISET-1
XLBF=0.0
GO TO 105

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

DO 103 I=1,ICP
IF(DLB .GT. E(I))GO TO 104
ISET=ISET+1
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

TYPE 1002
1002 FORMAT( ' TEMP FROM SUBROUTINE LOWBND FORMAT 1002' )
TYPE 1000
1000 FORMAT( 'ILBG,ILBFG,XLBF,DELTAE,PARTE' )
TYPE 1001, 1002, 1003, 1004, 1005, 1006
1001 FORMAT(2I5,3E12.4)
RETURN
END
REFERENCES


