5. FILE 5, ENERGY DISTRIBUTION OF SECONDARY PARTICLES

5.1. General Description

File 5 is used to describe the energy distributions of secondary particles expressed as normalized probability distributions. File 5 is for incident neutron reactions and spontaneous fission only, and should not be used for any other incident particle. Data will be given in File 5 for all reaction types that produce secondary neutrons, unless the secondary neutron energy distributions can be implicitly determined from data given in File 3 and/or File 4. No data will be given in File 5 for elastic scattering (MT=2), since the secondary energy distributions can be obtained from the angular distributions in File 4. No data will be given for neutrons that result from excitation of discrete inelastic levels when data for these reactions are given in both File 3 and File 4 (MT=51, 52, ..., 90).

Data should be given in File 5 for MT=91 (inelastic scattering to a continuum of levels), MT=18 (fission), MT=16 (n,2n), MT=17 (n,3n), MT=455 (delayed neutrons from fission), and certain other nonelastic reactions that produce secondary neutrons. The energy distribution for spontaneous fission is given in File 5 (in sub-library 4).

File 5 may also contain energy distributions of secondary charged particle for continuum reactions where only a single outgoing charged particle is possible (MT=649, 699, *etc.*). Continuum photon distributions should be described in File 15.

The use of File 6 to describe all particle energy distributions is preferred when several charged particles are emitted or the particle energy and angular distribution are strongly correlated. In these cases Files 5 and 15 should not be used.

Each section of the file gives the data for a particular reaction type (MT number). The sections are then ordered by increasing MT number. The energy distributions $p(E\rightarrow E')$, are normalized so that

$$\int_0^{E'_{\max}} p(E \rightarrow E')dE' = 1 \tag{5.1}$$

where E'_{max} is the maximum possible secondary particle energy and its value depends on the incoming particle energy E and the analytic representation of $p(E\rightarrow E')$. The secondary particle energy E' is always expressed in the *laboratory system*.

The differential cross section is obtained from

$$\frac{d\sigma(E \rightarrow E')}{dE'} = m\sigma(E)\, p(E \rightarrow E') \tag{5.2}$$

where $\sigma(E)$ is the cross section as given in File 3 for the same reaction type number (MT) and m is the neutron multiplicity for this reaction (m is implicit; *e.g.*, m=2 for n,2n reactions).

The energy distributions $p(E\rightarrow E')$ can be broken down into partial energy distributions, $f_k(E\rightarrow E')$, where each of the partial distributions can be described by different analytic representations;

$$p(E \rightarrow E') = \sum_{k=1}^{NK} p_k(E) f_k(E \rightarrow E') \tag{5.3}$$

and at a particular incident neutron energy E,

$$\sum_{k=1}^{NK} p_k(E) = 1$$

where $p_k(E)$ is the fractional probability that the distribution $f_k(E\rightarrow E')$ can be used at E.

The partial energy distributions $f_k(E \rightarrow E')$ are represented by various analytical formulations. Each formulation is called an energy distribution law and has an identification number associated with it (LF number). The allowed energy distribution laws are given below.

Secondary Energy Distribution Laws

LF = l, ***Arbitrary tabulated function***:

$$f(E \rightarrow E') = g(E \rightarrow E')$$

A set of incident energy points is given, E and $g(E \rightarrow E')$ is tabulated as a function of E'.

LF = 5, ***General evaporation spectrum***:

$$f(E \rightarrow E') = g\left(E'/\theta(E)\right)$$

$\theta(E)$ is tabulated as a function of incident neutron energy, E; $g(x)$ is tabulated as a function of x, $x = E'/\theta(E)$.

LF = 7, ***Simple fission spectrum (Maxwellian)***:

$$f(E \rightarrow E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)}$$

I is the normalization constant,

$$I = \theta^{3/2}\left[\frac{\sqrt{\pi}}{2} erf\left(\sqrt{(E-U)/\theta}\right) - \sqrt{(E-U)/\theta} e^{-(E-U)/\theta}\right]$$

θ is tabulated as a function of energy, E;
U is a constant introduced to define the proper upper limit for the final particle energy such that $0 \le E' \le (E - U)$.

LF = 9, **Evaporation spectrum:**

$$f(E \rightarrow E') = \frac{E'}{I} e^{-E'/\theta(E)}$$

I is the normalization constant,

$$I = \theta^2\left[1 - e^{-(E-U)/\theta}\left(1 + \frac{E-U}{\theta}\right)\right]$$

θ is tabulated as a function of incident neutron energy, E;
U is a constant introduced to define the proper upper limit for the final particle energy such that $0 \le E' \le (E–U)$

LF = 11, Energy dependent Watt spectrum:

$$f(E \to E') = \frac{e^{-E'/a}}{I} \sinh\left(\sqrt{bE'}\right)$$

I is the normalization constant,

$$I = \frac{1}{2}\sqrt{\frac{\pi a^3 b}{4}} \exp\left(\frac{ab}{4}\right)\left[erf\left(\sqrt{\frac{E-U}{a}} - \sqrt{\frac{ab}{4}}\right) + erf\left(\sqrt{\frac{E-U}{a}} + \sqrt{\frac{ab}{4}}\right)\right]$$

$$- a \exp\left[-\left(\frac{E-U}{a}\right)\right] \sinh\sqrt{b(E-U)}$$

a and b are energy dependent;
U is a constant introduced to define the proper upper limit for the final particle energy such that $0 \le E' \le (E\text{-}U)$

LF = 12, Energy dependent fission neutron spectrum (Madland and Nix):

$$f(E \to E') = \frac{1}{2}\left[g(E', EFL) + g(E', EFH)\right]$$

where

$$g(E', EF) = \frac{1}{3(EF \times TM)^{1/2}}\left[u_2^{3/2} E_1(u_2) - u_1^{3/2} E_1(u_1) + \gamma(3/2, u_2) - \gamma(3/2, u_1)\right]$$

$$u_1 = \left(\sqrt{E'} - \sqrt{EF}\right)^2 \Big/ TM$$

$$u_2 = \left(\sqrt{E'} - \sqrt{EF}\right)^2 \Big/ TM$$

EFL and *EFH* are constants, which represent the average kinetic energy per nucleon of the average light and heavy fragments, respectively.
TM depends upon the incident neutron energy,
$E_1(x)$ is the exponential integral,
$\gamma(a,x)$ is the incomplete gamma function.

The integral of this spectrum between zero and infinity is one. The value of the integral for a finite integration range is given in Sec. 5.4.10.

The data are given in each section by specifying the number of partial energy distributions that will be used. The same energy mesh should be used for each one. The partial energy distributions may all use the same energy distribution law (LF number) or they may use different laws.

Note: Distribution laws are not presented for LF = 2, 3, 4, 6, 8, or 10. These laws are no longer used.

5.2. Formats

Each section of File 5 contains the data for a particular reaction type (MT number), starts with a HEAD record, and ends with a SEND record. Each subsection contains the data for one partial energy distribution. The structure of a subsection depends on the value of LF (the energy distribution law).

The following quantities are defined.

NK	Number of partial energy distributions. There will be one subsection for each partial distribution.
U	Constant that defines the upper energy limit for the secondary particle so that $0 \le E' \le E - U$ (given in the LAB system).
θ	Effective temperature used to describe the secondary energy distribution for LF = 5, 7, or 9.
LF	Flag specifying the energy distribution law used for a particular subsection (partial energy distribution). (The definitions for LF are given in Section 5.l.).
$p_k(E_N)$	Fractional part of the particular cross section which can be described by the k^{th} partial energy distribution at the Nth incident energy point. NOTE: $\sum_{k=1}^{NK} p_k(E_N) = 1.0$
$f_k(E \rightarrow E')$	k^{th} partial energy distribution. The definition depends on the value of LF.
NR	Number of interpolation ranges.
NP	Number of incident energy points at which $p_k(E)$ is given.
a,b	Parameters used in the energy dependent Watt spectrum, LF = 11.
EFL,EFH	Constants used in the energy-dependent fission neutron spectrum (Madland and Nix), LF = 12.
TM	Maximum temperature parameter, TM(E), of the energy-dependent fission neutron spectrum (Madland and Nix), LF =12.
NE	Number of incident energy points at which a tabulated distribution is given (NE≤200.)
NF	Number of secondary energy points in a tabulation. (NF≤l000.)

The structure of a section has the following form:

```
[MAT, 5, MT/  ZA,  AWR,  0,  0, NK,  0]HEAD
<subsection for k = 1>
<subsection for k = 2>
----------------------
<subsection for k = NK>
[MAT, 5, MT/ 0.0,  0.0,  0,  0,  0,  0]SEND
```

The structure of a subsection depends on the value of LF. The formats for the various values of LF are given following.

LF = 1, ***Arbitrary tabulated function***

[MAT, 5, MT/ 0.0, 0.0, 0, LF, NR, NP/ E_{int}/p(E)]TAB1 LF=1
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE/ E_{int}]TAB2
[MAT, 5, MT/ 0.0, E_1, 0, 0, NR, NF/ E'_{int} /
E'_1, 0.0, E'_2,g($E_1 \rightarrow E'_2$), E'_3,g($E_1 \rightarrow E'_3$),
---
---,E'_{NF-1},g($E_1 \rightarrow E'_{NF-1}$), E'_{NF} , 0.0]TAB1
[MAT, 5, MT/ 0.0, E_2 , 0, 0, NR, NF/ E'_{int} /
E'_1, 0.0, E'_2,g($E_2 \rightarrow E'_2$), E'_3 ,g($E_2 \rightarrow E'_3$),----
---
---, E'_{NF-1},g($E_2 \rightarrow E'_{NF-1}$), E'_{NF} , 0.0]TAB1
[MAT, 5, MT/ 0.0, E_{NE}, 0, 0, NR, NF/ E'_{int} /
E'_1, 0.0, E'_2,g($E_{NE} \rightarrow E'_2$), E'_3,g($E_{NE} \rightarrow E'_3$)----
---
--- ,E'_{NF-1},g($E_{NE} \rightarrow E'_{NF-1}$), E'_{NF}, 0.0]TAB1

Note that the incident energy mesh for $p_k(E)$ does not have to be the same as the E mesh used to specify the energy distributions. The interpolation scheme used between incident energy points, E, and between secondary energy points, E′, should be linear-linear.

LF = 5, ***General evaporation spectrum***

[MAT, 5, MT/ U, 0.0, 0, LF, NR, NP/ E_{int} / p(E)]TAB1 (LF=5)
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE/ E_{int} / θ(E)]TAB1
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NF/ x_{int} / g(x)]TAB1 (x=E′/θ(E))

LF = 7, ***Simple fission spectrum (Maxwellian)***

[MAT, 5, MT/ U, 0.0, 0, LF, NR, NP/ E_{int} / p(E)]TAB1 (LF=7)
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE/ E_{int} / θ (E)]TAB1

LF = 9, ***Evaporation spectrum***

[MAT, 5, MT/ U, 0.0, 0, LF, NR, NP/ E_{int} / p(E)]TAB1 (LF=9)
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE/ E_{int} / θ (E)]TAB1

LF = 11, ***Energy-dependent Watt spectrum***

[MAT, 5, MT/ U, 0.0, 0, LF, NR, NP / E_{int} / p(E)]TAB1 (LF=11)
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE / E_{int} / a(E)]TAB1
[MAT, 5, MT/ 0.0, 0.0, 0, 0, NR, NE / E_{int} / b(E)]TAB1

LF = 12, ***Energy-dependent fission neutron spectrum (Madland and Nix)***

[MAT, 5, MT/ 0.0, 0.0, 0, LF, NR, NP/ E_{int} / p(E)]TAB1 (LF=12)
[MAT, 5, MT/ EFL, EFH, 0, 0, NR, NE/ E_{int} /TM(E)]TAB1

5.3. Procedures

As many as three different energy meshes may be required to describe the data in a subsection (one partial distribution). These are the incident energy mesh for $p_k(E)$, the incident energy mesh at which the secondary neutrons are given, $f_k(E \rightarrow E')$, and the secondary energy mesh for $f_k(E \rightarrow E')$. It is recommended that a linear-linear or a linear-log interpolation scheme be used for the first two energy meshes, and a linear-linear interpolation for the last energy mesh.

Double energy points must be given in the incident energy mesh whenever there is a discontinuity in any of the $p_k(E)$'s (this situation occurs fairly frequently). This energy mesh must also include threshold energy values for all reactions being described by the $p_k(E)$'s. Zero values for p_k must be given for energies below the threshold (if applicable).

Two nuclear temperatures may be given for the (n,2n) reaction. Each temperature, θ, may be given as a function of incident neutron energy. In this case $p_1(E) = p_2(E) = 0.5$. A similar procedure may be followed for the (n,3n) and other reactions.

A constant, U, is given for certain distribution laws (LF = 5, 7, 9, or 11). The constant, U, is provided to define the proper upper limit for the secondary energy distribution so that $0 \leq E' \leq E - U$. The value of U depends on how the data are represented for a particular reaction type. Consider U for inelastic scattering.

Case A: The total inelastic scattering cross section is described as a continuum. U is the threshold energy for exciting the lowest level in the residual nucleus.

Case B: For the energy range considered, the first three levels are described explicitly (either in File 3, MT = 51, 52, and 53, or in File 5), and the rest of the inelastic cross section is treated as a continuum. U is the threshold energy (known or estimated) for the fourth level in the residual nucleus.

If the reaction being described is fission, then U should be a large negative value ($U = -20.0 \times 10^6$ eV to -30×10^6 eV). In this case neutrons can be born with energies much larger than the incident neutron energy. It is common practice to describe the inelastic cross section as the sum of excitation cross sections (for discrete levels) for neutron energies up to the point where level positions are no longer known. At this energy point, the total inelastic cross section is treated as a continuum. This practice can lead to erroneous secondary energy distributions for incident neutron energies just above the cutoff energy. It is recommended that the level excitation cross sections for the first several levels (*e.g.*, 4 or 5 levels) be estimated for several MeV above the cutoff energy. The continuum portion of the inelastic cross section will be zero at the cutoff energy, and it will not become the total inelastic cross section until several MeV above the cutoff energy.

It is recommended that the cross sections for excitation of discrete inelastic levels be described in File 3 (MT = 51, 52, ..., *etc.*). The angular distributions for the neutrons resulting from these levels should be given in File 4 (the same MT numbers). The secondary energy distributions for these neutrons can be obtained analytically from the data in Files 3 and 4. This procedure is the only way in which the energy distributions can be given for these neutrons. For inelastic scattering, the only data required in Files 5 are for MT = 91 (continuum part).

5.4. Additional Procedures

5.4.l. General Comments

1. Do not give File 5 data for the discrete level excitation data given in File 3 as MT = 51, 52, ..., 90. If MT=91 is given in File 3, a section for MT=91 must be given in File 5 or File 6. A section must also be given in File 5 or File 6 for all other neutron-producing reactions. Continuum energy distributions for emitted protons, deuterons *etc*., may be given in MT=649 *etc*., and for photons, in File 6 or File 15. When more than one particle type is emitted, File 6 should be used to assure energy conservation.
2. Care must be used in selecting the distribution law number (LF) to represent the data. As a rule, use the simplest law that will accurately represent the data.
3. A section in File 5 must cover the same incident energy range as was used for the same MT number in File 3. The sum of the probabilities for all laws used must be equal to unity for all incident energy points.
4. If the incident neutron energy exceeds several MeV, pre-equilibrium neutron emission can be important, as illustrated from high-resolution neutron and proton spectra measurements and analysis of pulsed sphere experiments. In these cases either tabulated spectra or "mocked-up" levels can be constructed to supplement or replace simple evaporation spectra.
5. Note that prompt fission spectra are given under MT=18, 19, 20, 21, and 38. The delayed fission spectra are given under MT=455. The energy distribution for prompt spontaneous fission is given in File 5 for MT=18, but in sublibrary 4. It is used with $\bar{\nu}_p$ from File 1 (MT=456) to determine the prompt spontaneous fission spectrum. The delayed spontaneous fission spectrum is determined from $\bar{\nu}_d$ from File 1 (MT=455) and the delayed energy spectrum in File 5 MT=455. Note that for the specification of spontaneous spectra no cross sections from File 3 are required.

5.4.2. LF = 1 (Tabulated Distributions)

Use only tabulated distributions to represent complicated energy distributions. Use the minimum number of incident energy points and secondary neutron energy points to accurately represent the data. The integral over secondary neutron energies for each incident energy point must be unity to within four significant figures. All interpolation schemes must be with linear-linear or linear-log (INT=1,2, or 3) to preserve probabilities upon interpolation. All secondary energy distributions must start and end with zero values for the distribution function $g(E \rightarrow E')$.

5.4.3. LF = 7 (Maxwellian Spectrum)

A linear-linear interpolation scheme is preferred for specifying the nuclear temperature as a function of energy.

5.4.4. LF = 9 (Evaporation Spectrum)

An evaporation spectrum is preferred for most reactions. Care must be taken in describing the nuclear temperature near the threshold of a reaction. Nuclear temperatures that are too large can violate conservation of energy.

5.4.5. LF = 11 (Watt Spectrum)

A linear-linear interpolation scheme is preferred for specifying the parameters a and b as a function of energy.

5.4.6. LF = 12 (Madland-Nix Spectrum)

A log-log interpolation scheme may be used for specifying the parameter TM as a function of incident neutron energy.

5.4.7. Selection of the Integration Constant, U

1. When LF = 5, 7, 9, or 11 is used, an integration constant U is required. This constant is used in defining the upper energy limit of secondary neutrons; *i.e.*, $E_{max} = E_n - U$, where E_n is the incident neutron energy. U is a constant for the complete energy range covered by a subsection in File 5 and is given in the LAB system.
2. U is negative for fission reactions. The preferred value is -20 MeV.
3. In practice, U can be taken to be the absolute value of Q for the lowest level (known or estimated) that can be excited by the particular reaction within the incident energy range covered by the subsection. U is actually a function of the incident neutron energy, but it can be shown that it is always greater than the absolute value of Q and less than the threshold energy of the reaction. At large AWR, since Eth and |Q| are approximately equal, either could be used but the absolute value of Q is preferred. At small AWR, using |Q| for U is the best approximation and must be used.
4. The following four cases commonly occur in data files; procedures are given for obtaining U values.

   Case A: The complete reaction is treated as a continuum.

   $U = -Q$, where Q is the reaction Q-value.

   Case B: The reaction is described by excitation of three levels (in File 3 as MT = 51, 52, 53) and a continuum part where Q_4 is the known or estimated Q-value for the fourth level.

   $U = -Q_4$.

   Case C: The reaction is described by excitation of three levels (in File 3 as MT=51, 52, and 53) and a continuum part which extends below the threshold for MT=51. If, for example, the reaction is a 3-body breakup reaction, use

   $U = -Q$, where Q is the energy required for 3-body breakup.

   Case D: The reaction is described by excitation of the first three levels (in File 3 as MT=51, 52, 53) for neutron energies from the level thresholds up to 20 MeV, excitation of the next five levels (in File 3 as MT=54, ..., 58) from their thresholds up to 8 MeV, and by a continuum part that starts at 5 MeV.

   In this case two subsections should be used, one to describe the energy range from 5 to 8 MeV and another to describe the energy region from 8 to 20 MeV. In the first subsection (5 - 8 MeV),

   $U = -Q_9$,

   and the second (8 - 20 MeV),

   $U = -Q_4$.

5.4.8. Multiple Nuclear Temperatures

Certain reactions, such as (n,2n), may require specification of more than one nuclear temperature. θ(E) should be given for each neutron in the exit channels; this is done by using more than one subsection for a reaction. The U value is the same for all subsections. The upper energy limit is determined by the threshold energy and not by level densities in the residual nuclei.

5.4.9. Average Energy for a Distribution

The average energy of a secondary neutron distribution must be less than the available energy for the reaction:

$$E_{avail} = E + \frac{1 + AWR}{AWR} Q,$$

where E_{avail} is greater than the neutron multiplicity times the average energy of all the emitted neutron $\nu\ \bar{E}'$, where ν is the multiplicity. The mean energy should be calculated from the distribution at each value of E. This mean is analytic in the four cases given below.

LF	$\overline{E'}$
7	$\frac{3}{2}\theta - \frac{\theta^{5/2}}{I}\left(\frac{E-U}{\theta}\right)^{3/2} e^{-(E-U)/\theta}$
9	$2\theta - \frac{\theta^3}{I}\left(\frac{E-U}{\theta}\right)^2 e^{-(E-U)/\theta}$
11	$\frac{1}{2I}\left\{ a\exp\left(\frac{ab}{4}\right)\sqrt{\pi\frac{a^3 b}{4}}\left(3+\frac{ab}{2}\right)\left[\frac{erf\left(\sqrt{\frac{E_r}{a}}-\sqrt{\frac{ab}{4}}\right)+erf\left(\sqrt{\frac{E_r}{4}}+\sqrt{\frac{ab}{4}}\right)}{2}\right] - 3a^2\sqrt{ab}\exp\left(-\frac{E_r}{a}\right)\left[\sqrt{\frac{E_r}{a}}\cosh\sqrt{bE_r}-\sqrt{\frac{ab}{4}}\sinh\sqrt{bE_r}\right] - 2a^2\exp\left(-\frac{E_r}{a}\right)\left[\left(\frac{E_r}{a}+\frac{ab}{4}\right)\sinh\sqrt{bE_r}-\sqrt{bE_r}\cosh\sqrt{bE_r}\right]\right\}$ where $E_r = E-U$
12	$\frac{1}{2}(EFL+EFH)+\frac{4}{3}TM$

U is described in Section 5.3. The analytic functions for I are given in Section 5.1 for LF = 7, 9, 11. For LF = 12, Section 5.4.10 gives the method for obtaining the integral of the distribution function.

5.4.10. Additional procedures for LF = 12, Energy-Dependent Fission Neutron Spectrum (Madland and Nix)

Integral over finite energy range [a,b].

Set:

$$\alpha = \sqrt{TM}\,,\ \beta = \sqrt{EF}$$

$$A = \left(\sqrt{a} + \beta\right)^2 / \alpha^2$$

$$B = \left(\sqrt{b} + \beta\right)^2 / \alpha^2$$

$$A' = \left(\sqrt{a} - \beta\right)^2 / \alpha^2$$

$$B' = \left(\sqrt{b} - \beta\right)^2 / \alpha^2$$

Then,

the integral is given by one of the following three expressions depending on the region of integration in which a and b lie.

<u>Region I</u> (a > EF, b > EF)

$$3(ET\ TM)^{1/2}\int_a^b g(E',EF)dE' =$$

$$\left[\left(\frac{2}{5}\alpha^2 B^{5/2} - \frac{1}{2}\alpha\beta B^2\right)E_1(B) - \left(\frac{2}{5}\alpha^2 A^{5/2} - \frac{1}{2}\alpha\beta A^2\right)E_1(A)\right]$$

$$-\left[\left(\frac{2}{5}\alpha^2 B'^{5/2} + \frac{1}{2}\alpha\beta B'^2\right)E_1(B') - \left(\frac{2}{5}\alpha^2 A'^{5/2} + \frac{1}{2}\alpha\beta A'^2\right)E_1(A')\right]$$

$$+\left[\left(\alpha^2 B - 2\alpha\beta B^{1/2}\right)\gamma(3/2,B) - \left(\alpha^2 A - 2\alpha\beta A^{1/2}\right)\gamma(3/2,A)\right]$$

$$-\left[\left(\alpha^2 B' + 2\alpha\beta B'^{1/2}\gamma(3/2,B'\right) - \left(\alpha^2 A' + 2\alpha\beta A'^{1/2}\right)\gamma(3/2,A')\right]$$

$$-\frac{3}{5}\alpha^2\left[\gamma(5/2,B) - \gamma(5/2,A) - \gamma(5/2,B') - \gamma(5/2,A')\right]$$

$$-\frac{3}{5}\alpha\beta\left[e^{-B}(1+B) - e^{-A}(1+A) + e^{-B'}(1+B') - e^{-A'}(1+A')\right]$$

<u>Region II</u> (a < EF, b < EF)

$$3(ET\ TM)^{1/2}\int_a^b g(E',EF)dE' =$$

$$\left[\left(\frac{2}{5}\alpha^2 B^{5/2} - \frac{1}{2}\alpha\beta B^2\right)E_1(B) - \left(\frac{2}{5}\alpha^2 A^{5/2} - \frac{1}{2}\alpha\beta A^2\right)E_1(A)\right]$$

$$-\left[\left(\frac{2}{5}\alpha^2 B'^{5/2} + \frac{1}{2}\alpha\beta B'^2\right)E_1(B') - \left(\frac{2}{5}\alpha^2 A'^{5/2} + \frac{1}{2}\alpha\beta A'^2\right)E_1(A')\right]$$

$$+\left[\left(\alpha^2 B - 2\alpha\beta B^{1/2}\right)\gamma(3/2,B) - \left(\alpha^2 A - 2\alpha\beta A^{1/2}\right)\gamma(3/2,A)\right]$$

$$-\left[\left(\alpha^2 B' - 2\alpha\beta B'^{1/2}\gamma(3/2,B'\right) - \left(\alpha^2 A' - 2\alpha\beta A'^{1/2}\right)\gamma(3/2,A')\right]$$

$$-\frac{3}{5}\alpha^2\left[\gamma(5/2,B) - \gamma(5/2,A) - \gamma(5/2,B') + \gamma(5/2,A')\right]$$

$$-\frac{3}{5}\alpha\beta\left[e^{-B}(1+B) - e^{-A}(1+A) + e^{-B'}(1+B') + e^{-A'}(1+A')\right]$$

Region III (a < EF, b > EF)

$$3(ET\ TM)^{1/2}\int_a^b g(E',EF)dE' =$$

$$\left[\left(\frac{2}{5}\alpha^2 B^{5/2}-\frac{1}{2}\alpha\beta B^2\right)E_1(B)-\left(\frac{2}{5}\alpha^2 A^{5/2}-\frac{1}{2}\alpha\beta A^2\right)E_1(A)\right]$$

$$-\left[\left(\frac{2}{5}\alpha^2 B'^{5/2}+\frac{1}{2}\alpha\beta B'^2\right)E_1(B')-\left(\frac{2}{5}\alpha^2 A'^{5/2}+\frac{1}{2}\alpha\beta A'^2\right)E_1(A')\right]$$

$$+\left[\left(\alpha^2 B-2\alpha\beta B^{1/2}\right)\gamma(3/2,B)-\left(\alpha^2 A-2\alpha\beta A^{1/2}\right)\gamma(3/2,A)\right]$$

$$-\left[\left(\alpha^2 B'+2\alpha\beta B'^{1/2}\gamma(3/2,B'\right)-\left(\alpha^2 A'+2\alpha\beta A'^{1/2}\right)\gamma(3/2,A')\right]$$

$$-\frac{3}{5}\alpha^2\left[\gamma(5/2,B)-\gamma(5/2,A)-\gamma(5/2,B')+\gamma(5/2,A')\right]$$

$$-\frac{3}{5}\alpha\beta\left[e^{-B}(1+B)-e^{-A}(1+A)+e^{-B'}(1+B')+e^{-A'}(1+A')\right]$$

The expression for Region III would be used to calculate a normalization integral I for the finite integration constant U, if a physical basis existed by which U could be well determined.