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Processing of Gamma-Rays in the New AMPX System

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Processing of Gamma-Rays in the New AMPX System

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Abstract: Special modules were provided in earlier releases of the AMPX system to process gamma-ray data. The SMUG and LAPHNGAS modules were used to produce multi-group parameters for gamma-ray interactions and gamma-ray yields, respectively.

This report describes how a collection of new routines provides equivalent (and improved) capabilities. The procedures are quite different in that they serve for both neutrons and gamma-rays (and other particles, if needed).

Comparison studies with the Los Alamos NJOY system show that the AMPX and NJOY code systems produce equivalent parameters.

1. INTRODUCTION

1.1. Background

The development of the AMPX[1–3] system was initiated in the late 1960's under the sponsorship of the Defense Nuclear Agency [now the Defense Threat Reduction Agency (DTRA)]. The core philosophy in the design of this system was to use modular programming techniques that would allow integrating neutron and gamma-ray cross-section processing to produce coupled-neutron-gamma-ray cross-sections that are needed in weapon's effects analyses, along with other applications.

Three major processing capabilities are required to make coupled cross-section libraries:

1. a procedure to produce group-averaged neutron cross-sections,

2. a procedure to produce group-averaged gamma-ray cross-sections, and

3. a procedure to produce the group-averaged parameters that are used to specify where gamma-ray sources are produced from neutron interactions.

These capabilities were provided in previous versions of AMPX by codes named XLACS, SMUG, and LAPHNGAS, respectively. (It is the capabilities provided by the latter two codes that this report is addressing. However, to put our remarks into perspective, some of the basic tenets of the AMPX-2000 system should be reviewed.)

The AMPX upgrade task was initiated in order to provide a cross-section processing system capable of treating the latest (Version 6) formats in the Evaluated Nuclear Data Files (ENDF/B).[4]

The most obvious approach would have been to start with the three codes mentioned above and modify them. This was considered and quickly dismissed, based on our knowledge of weaknesses in the 30-year-old codes. Certainly, the fact that all of the major cross-section libraries used in the SCALE[5] system were produced by these codes suggests that they can produce credible libraries. However, the XLACS code, in particular, was constructed by combining and patching several existing codes that were independently developed. As a consequence, many different approaches are used for operations as simple as integrating the product of two functions. Another upgrading would increase this patching and result in a set of codes that are difficult to maintain.

A second alternative would have been to simply cease development on AMPX and to adopt the NJOY[6] system developed at Los Alamos National Laboratory. NJOY was an obvious choice, since it was the only existing code system that could process all data formats used in the newer ENDF/B evaluations. However, the primary resonance self-shielding method for this system is typically based on the Narrow Resonance approximation, which does not give results that are as accurate as the methods based on Integral Transport theory that have always been used in the SCALE system.

The third approach would be to keep our AMPX system, and to rewrite some of the major processing routines. In addition to concerns with NJOY's resonance processing, the consideration that tipped the scales in favor of this approach was the observation that the processing of a new class of data widely used in new ENDF/B-VI evaluations (called File 6 data) requires coding that is considerably more complicated than any used in the existing codes. These new data allow describing the energy and angular variation of the kinematics of a reaction in a very general manner. It was recognized that if we were careful in writing a processor for this most complicated situation, that it should also be able to process the simpler situations, thereby, allowing us to eliminate redundant coding. Finally, another important consideration was simply

the belief that cross-section processing is important enough to warrant at least two code systems that can deal with the latest evaluations for quality assurance reasons and to allow independent testing of processed results.

With those observations, it is now noted that the present upgrade went beyond the goals stated above by recognizing that one does not have to create a collection of individual codes, such as XLACS, LAPHNGAS, and SMUG. These codes all deal with a particular class of data for a single particle. Rather it is possible to create a single collection of generic codes that can be used to treat any particle. This would unify the processing, since all particles would share the coding for performing all similar procedures, such as integration.

There are a few exceptions to this rule. For example, to produce multigroup neutron cross-sections, one must have arrays of cross-sections given as a function of energy. In the present evaluations, neutron cross sections are given as a combination of resonance parameters and tabulations of energy-value pairs. The POLIDENT module is used to reconstruct energy-value pairs from resonance parameters and to combine these data with the other energy-value pairs to form complete functions for elastic scattering, capture and fission. There is presently no equivalent requirement for this kind of processing for any other particle.

A major innovation in AMPX is to convert ENDF/B data into a “kinematics file” that uses a tabular format for all scattering reactions. This file is used in combination with a file of “point data” in integration routines to produce all group-averaged parameters. These routines know nothing about the “physics” of any reaction, or even what kind of particle is being treated. They are used to treat neutron-to-neutron, neutron-to-gamma-ray, and gamma-ray-to-gamma-ray scattering. Obviously, the coding that calls the integration routines must know what kind of data is being processed, because it has to know where it should be included in a cross-section library. In AMPX, the Y12 module produces the kinematics file for all particles.

The X10 module reads point data files and kinematics files and calls the integration routines mentioned above to produce the parameters needed in a multigroup cross-section library. X10 constructs the collections of averaged parameters for neutron interactions, photon interaction, and/or photon production interactions, and exports these parameters in the AMPX master cross-section library format that is format used by the SCALE system.

1.2. Purpose and Scope

The purpose of this work is to provide a replacement for the LAPHNGAS and SMUG modules for producing multigroup photon yield data and photon interaction data, respectively. An important part of the upgrade is to write simpler and cleaner coding, that takes advantage of knowledge gained from using and studying the methods in the older codes.

2. METHODOLOGY

When confronted with the need to produce multigroup cross-sections for a new process, it is natural for a code developer to examine the situation and consider developing coding that is specifically tailored to take advantage of the specific characteristics of the process. For example, one needs to know if the process has temperature dependence, if it scatters isotropically, etc. In some instances, such as for hydrogen elastic scattering, codes have been written that treat the complete process using analytical techniques.

In the new AMPX, we have gone to the other extreme and designed a collection of codes where no process for any particle will use its own collection of subroutines. This has been accomplished by imposing a requirement that all ENDF/B data be converted to tabular point data and kinematics files. (Obviously, the procedures that write these two files must have intimate knowledge of the ENDF/B formats and the physics of all interactions!) The structure of the kinematics file is described in Table 1. Note that the concise structure consists of a temperature loop, a subsection loop, an angular loop, and a source energy loop. The important thing here is that all processes for all particles use all loops. Most processes don't vary as a function of temperature; however, in this structure, they must have at least one temperature, whose value is irrelevant, and one subsection, and one angle (even if the process is isotropic), etc. The reason for this imposition is three-fold: (1) it makes the coding simpler by removing the requirement of having to make tests to branch to different sections of coding, (2) all of the actual data processing is done within the source energy loop, and (3) it works for the most complicated situation needed.

Table 1 Structure of kinematics data file for fixed angular quadrature

Record 1: MT, NTEMP				
	Loop over NTEMP Temperatures			
	Record Type 2: T, NSUB			
		Loop over NSUB Subsections		
		Record Type 3: MM, (μ_m, m=1,MM),(w_m, *m*=1,MM), ZAP, AWP		
			Loop over MM Angles	
			Record Type 4: μ_m, w_m, NE	
				Loop over NE Source Energies
				Record Type 5: E, NF
				Record Type 6: (EF_i, I=1,NF), (F_i, I=1,NF),(P_i,I=1,NF)

In Table 1, MT is the process identifier (ENDF/B identifiers are used), T is the temperature in Kelvin, μ_m is the cosine of the angle in a Lobatto quadrature of order MM, w_m is the corresponding integration weight, E is the source energy at which the interaction takes place, and the EF, F, and P arrays are the actual scattering distribution given at NF sink-energy points (which will be one point for functions that are described by a δ-function). The subsection loop may seem superfluous, but it is a kind of "insurance" loop (an extra loop for situations not foreseen), that is used to account for a variety of things, such as the cases wherein the scattering for a reaction is given in different ways.

The tabulation of δ-functions is new to AMPX, and is not, to the knowledge of the author, done in any other processing system. As it turns out, it affords a very convenient and easy way to process these data, and allows one to write integration routines that do not need to know the "physics" for a reaction.

For neutron interactions, point data are written to a file in the ENDF/B "TAB1" record format by the POLIDENT module mentioned above. In the special case for photons, Y12 writes a point file of photon data, which uses the same format.

AMPX has a special module named JERGENS that contains a variety of options that are useful for constructing the weighting functions, $\phi(E)$, that are needed to produce multigroup cross-sections. These are also written in the TAB1 format.

Many multigroup parameters require calculating averages of a cross-section function by the weighting function just mentioned according to the following expression:

$$\overline{\sigma}_g = \frac{\int_g dE\ \sigma(E)\ \phi(E)}{\int_g dE\ \phi(E)} \tag{2.1}$$

Other situations (e.g., the average number of fission neutrons produced by a fission) require that one weight the function over the product of a cross-section times a weighting function:

$$\overline{\nu}_g = \frac{\int_g dE\ \nu(E)\ \sigma_f(E)\ \phi(E)}{\int_g dE\ \sigma_f(E)\ \phi(E)} \tag{2.2}$$

Typical group-to-group scattering cross-sections are the Legendre coefficients of a fit to the angular variation of the cross-section, and are defined by:

$$\sigma_n(g \to g') = \frac{2n+1}{2} \frac{\int_{-1}^{+1} d\mu\ P_n(\mu) \int_g dE\ \sigma(E,\mu)\ \phi(E) \int_{g'} dE' f(E \to E',\mu)}{\int_g dE\ \phi(E)} \tag{2.3}$$

For the special case of gamma ray yields, one encounters a case exactly analogous to that described by Equation (2.2), wherein the desired parameter is a yield that must be weighted by the product of a cross-section times a weighting function:

$$y_n(g \to g') = \frac{2n+1}{2} \frac{\int_{-1}^{+1} d\mu\ P_n(\mu) \int_g dE\ y(E)\ \sigma(E,\mu)\ \phi(E) \int_{g'} dE' f(E \to E',\mu)}{\int_g dE\ \sigma(E)\ \phi(E)} \tag{2.4}$$

In all of these cases, g denotes the energy limits of an energy group. For the scattering distributions, g is the source energy group and g' is the sink energy group. Note that in the case of gamma yields that the source and sink energies are in different spaces.

The X10 module reads the point data file, kinematics file, and the weight function file and evaluates Equations (2.1) through (2.4) to determine parameters to create a new multigroup library.

X10 contains 3 primary control routines, corresponding to the production of neutron, gamma yield, or photon multigroup parameters. These three control modules call a single collection of routines to calculate their parameters. Moreover, the control modules survey the input files to determine what data are present, call integration routines to obtain multigroup parameters, and assemble the multigroup parameters in the formats that are provided by the AMPX master cross-section libraries.

The primary integration subroutine is X2D6 which acts according to the instructions provided in the calling sequence. The integration routines are the core of the X10 module, and, as already emphasized several times, do not have to know whether they are producing neutron cross-sections, photon yields, or photon cross-sections. No special instructions tell them about the types of data involved. The control routines are expected to combine and parse the data as needed for a cross-section library.

The following items are the primary information classes that are passed to X2D6:

1. the cross-section,
2. a weighting spectrum,
3. a possible multiplicity (such as nu-bar),
4. an auxiliary function that may be needed (see the discussion of how special integrands can be constructed in the text that follows),
5. a parameter that tells whether kinematics data are present,
6. a primary (source) energy group structure,
7. a secondary (sink) energy group structure,
8. optional instructions to tell about the structure of the integrand.

Other parameters that are input to X2D6 are outside the scope of this report.

If multiplicity data or kinematics data are not supplied, X2D6 uses an adaptive procedure to determine multigroup averages according to Equation (2.1).

When multiplicity data are given, but no kinematics data are supplied, X2D6 returns both the averaged cross-sections from Equation (2.1), and the multiplicity determined from Equation (2.2).

When no multiplicity data are given, but kinematics data are supplied, X2D6 uses Equation (2.3) to return group-to-group scattering terms and group averaged cross-sections using Equation (2.1).

When kinematics data and multiplicity data are included, X2D6 uses Equation (2.4) to return gamma yield group-to-group terms.

Note that when kinematics data are present, a primary and a secondary group structure are always required, even when they are exactly the same. The code assumes that the kinematics describe the scattering from the primary energy group space to the secondary energy group space. It is because of design details, such as this one, that X2D6 will work for all cases that are needed.

A brief comment that was made earlier indicated that a user could have control over the structure of the integrand. Most parameters in a multigroup library can be calculated by the 4 equations given above; however, there are other situations that involve alternate and more complicated forms of the integrands. A special "machine-language" type set of primitive commands can be optionally used to direct X2D6 to calculate terms such as Bondarenko factors, or even to calculate Bondarenko factors for the group-to-group scattering matrix terms.

2.1. Gamma Yield Processing

In ENDF/B Version 6, gamma-ray yield data are given in a variety of ways. Without trying to describe the structure of an ENDF/B library in detail, cross-section information is provided in arbitrary collections of data referred to as "files". These files are not physical files in the usual sense, but rather refer to different types of data that are collected together in the same physical file.

File 6 is a new ENDF/B file that is used to describe how neutron interactions produce other particles, such as neutrons, gamma-rays, electrons, etc. When gamma-ray yield information is given in File 6, the kinematics file will be produced according to Table 1, with one temperature (0 K), one subsection and an angular quadrature. For each angular quadrature, a source energy quadrature is provided with the probability distributions for scattering through a particular angle and from a particular source energy.

File 12 is reserved for gamma-ray multiplicity data. These parameters specify the number of gamma-rays that are produced from a neutron interaction. In many cases, such as the case of transition energies between inelastic scattering energy levels, the energies of the gamma-rays are also given in this file, and the next file that is discussed.

Many times, the evaluator does not know the multiplicity, but can estimate the neutron cross-section times the multiplicity as a reaction rate. Data of this type are given in File 13 which are referred to as being in cross-section units.

The angular variation for gamma-rays that are produced from neutron interactions can be specified in File 14, while File 15 can be used to specify energy distributions for gamma-rays for the case wherein the neutron interaction does not lead to a single gamma-ray energy.

These files are given sequentially on an ENDF/B library (i.e., File 12 data are given before File 13 data, etc.). This creates some data management difficulties because data given in Files 12 or 13 may have related data in File 14 and/or File 15. All combinations of these situations must be addressed, with the only absolute fact being that each process must have data in either File 12 or 13. When photon yield data are given in File 6, they are complete, and do not require looking at other files.

All of this leads to a situation wherein one must access the data in 4 separate files before the kinematics data can be written. This means that data must be either retained in memory or written to scratch files and combined, as necessary, when all data have been accessed. Ultimately, everything must be written in the structure shown in Table 1.

Photon multiplicities are represented three ways in File 12:

1. the most common form is to give the energy of the photon produced by the reaction (in the case of discrete level inelastic scattering, these could, for example, be the discrete gamma-rays resulting from transitions between the inelastic levels),

2. another form specifies that this photon energy is zero, and expects that an energy distribution be given in File 15,

3. a special form (rarely used because most nuclides have not been measured well enough to use it) that describes a complete transition probability matrix using level energies

along with probabilities that specify whether the transition between a pair of level produces a photon.

The multiplicities for these three forms may need to be combined with the data from File 14 if the angular variation is used.

In many (perhaps most) evaluations, the data for a process will involve specifying multiplicities for many photons. This is where the "subsection" loop in the structure described in Table 1 offers an ideal means for treating this situation. When Y12 reads the data for one of the photons or photon distributions for a process, it is simply treated as a separate subsection and placed in the appropriate place in the kinematics file. Some of the evaluations processed to date literally contain data for 50 or more photons, and, hence, use a corresponding number of subsections.

The special transition probability format is treated in an exactly analogous manner, with each "transition" photon relegated to a subsection in the kinematics file.

This is just one of the several situations where we have found that the inclusion of the subsection loop in the kinematics file allows us to do things that would be very difficult, otherwise. In this case, we would need a means of specifying multiple δ-functions, something that is very difficult to represent in a tabular fashion, or would have to calculate something like an "equivalent" photon energy that preserved energy, for example.

File 13 uses the first two forms mentioned above for File 12, the only difference being that the data are given in "cross-section" versus multiplicity units. As with File 12, the subsection loop feature is employed to store the data in the kinematics file.

In File 14, angular distribution data are given either using Legendre polynomial fits or tabular distributions. Since the files assume the angular and energy distributions are separable, a simple interpolation or polynomial evaluation is all that is required to get values at the angles in the angular loop.

File 15 can be used to represent scattering distributions from an arbitrary number of source energies for a reaction. These distributions are multiplied by the angular distribution values from File 14 and placed in the kinematics file in the slot reserved for record type 6. (In the cases that don't involve File 15, the distribution that is written at this place consists of the photon energy, followed by the angular distribution value.)

2.2. Gamma-Ray Interaction Processing

Two files are in ENDF/B for photon interaction data. File 23 is used to present the energy variation of the cross-sections for all reactions. File 27 contains the atomic form factors that are needed to represent coherent photon scattering, and it also contains scattering functions associated with the Klein-Nishina equations that describe incoherent scattering.

Y12 acts as a pass-through for the point photon cross-sections in File 23. It simply reads them and writes a file that is exactly analogous to the neutron point data files produced by the POLIDENT module.

For the two scattering processes, Y12 reads the atomic form factors and scattering functions and uses them to create a kinematics file that uses the structure shown in Table 1. As noted below, there is an alternate manner in which this structure can be used to accommodate the coherent and incoherent scattering cases. (This same alternate scheme is also needed for thermal neutron scattering and for hydrogen elastic scattering).

2.2.1. Coherent Photon Scattering

The equation for coherent scattering has the following form:

$$\frac{d\sigma_{coh}(E \to E', \mu)}{d\mu} = \frac{3\sigma_{TH}}{8}(1+\mu^2)\left[\left(F(q;Z)+F'(E)\right)^2 + F''(E)^2\right]\delta(E,E') \qquad (2.5)$$

where

σ_{Th} = 0.66524486, the classical Thompson cross section for the electron,

q = $\alpha[2(1-\mu)]^{1/2}$, the recoil momentum of the atom (in inverse angstroms),

α = $E_\gamma / m_0 c^2$,

m_0 = mass of an electron,

c = speed of light,

$F(q;Z)$ = the form factor,

$F'(E)$ = the real anomalous scattering factor, and

$F''(E)$ = the imaginary anomalous scattering factor.

Equation (2.5) does not lend itself to using a mechanical quadrature for integrating out the angular variation of the differential cross-section, because it is a very forward peaked function, and no angular quadrature will be fine enough to adequately sample it. In fact, this function is

very much like a δ-function, and even very high order angular quadratures will only "see" the most forward angle, thereby, resulting in a very inaccurate representation of the function's angular dependence.

Consider the following. Using the structure that is described in Table 1, X2D6 uses the angular loop to evaluate group-to-group parameters that are a function of scattering through a particular angle, and calculates terms of the form: $\sigma(g \rightarrow g',\ \mu_m)$. This approach is obviously solving Equation (2.3) by integrating over the source group g and sink group g', before performing the integration over angle. This allows a simple mechanical quadrature to be used to determine the Legendre coefficients of the group-to-group scattering cross section from:

$$\sigma_n(g \rightarrow g') = \frac{2n+1}{2} \sum_{m=1}^{MM} w_m\ \sigma(g \rightarrow g', \mu_m)\ P_n(\mu_m) \tag{2.6}$$

In fact, most codes that calculate Legendre coefficients for group-to-group cross-sections do not employ this scheme, but assume that Equation (2.3) is written with the first two integrations reversed:

$$\sigma_n(g \rightarrow g') = \frac{2n+1}{2} \frac{\int_g dE\ \phi(E) \int_{-1}^{+1} d\mu\ \sigma(E,\mu)\ P_n(\mu) \int_{g'} dE'\ f(E \rightarrow E', \mu)}{\int_g dE\ \phi(E)} \tag{2.7}$$

When the angular integration is performed, the resulting equation is:

$$\sigma_n(g \rightarrow g') = \frac{2n+1}{2} \frac{\int_g dE\ \phi(E) \int_{g'} dE'\ \sigma_n(E \rightarrow E')}{\int_g dE\ \phi(E)} \tag{2.8}$$

The term $\sigma_n(E \rightarrow E')$ is:

$$\sigma_n(E \rightarrow E') = \int_{-1}^{+1} d\mu\ \sigma(E,\mu)\ P_n(\mu)\ f(E \rightarrow E', \mu) \tag{2.9}$$

that ultimately leaves one with a much simpler expression to evaluate:

$$\sigma_n(g \to g') = \frac{2n+1}{2} \frac{\int_g dE\, \phi(E)\, \sigma_n(E \to g')}{\int_g dE\, \phi(E)} \tag{2.10}$$

We recognize that this expression is simpler, but its primary difficulty is associated with evaluating Equation (2.9) because of the polynomial that is part of the expression.

As it turns out, a different approach removes a great deal of the difficulty, and this is to use functions defined by:

$$\sigma_n^*(E \to E') = \int_{-1}^{+1} d\mu\, \sigma(E,\mu)\, \mu^n\ f(E \to E',\mu) \tag{2.11}$$

What we are, of course, suggesting is that it is easier to calculate cosine moments than it is to calculate Legendre moments, since μ^n is a much smoother function, and will, in many cases, yield expressions that can be treated analytically. We further recognize that, Legendre Polynomials are linear combinations of μ raised to different powers. Therefore, if we simply calculate group-to-group matrices using the μ-moments, these can be linearly combined to yield the Legendre group-to-group matrices. Exactly the same coefficients are used.

All of this led us to define an alternate form for the kinematics file described in Table 1, that has exactly the same structure (so that the X2D6 integration routines don't have to be modified) and yields group-to-group cosine moments that are easily converted to Legendre moments. The alternate kinematics file is shown in Table 2.

Table 2 Structure of kinematics data file when cosine moments are used

Record 1: MT, NTEMP				
	Loop over NTEMP Temperatures			
	Record Type 2: T, NSUB			
		Loop over NSUB Subsections		
		Record Type 3: MM, (DUMMY, m=1,2*MM), ZAP, AWP		
			Loop over MM Cosine Moments	
			Record Type 4: ORDER,0.0, NE	
				Loop over NE Source Energies
				Record Type 5: E, NF
				Record Type 6: (EF_i, I=1,NF), (F_i, I=1,NF),(P_i,I=1,NF)

Note that this structure is identical to that in Table 1, except for Record Type 3 and 4. Here we don't need an angular quadrature, so we simply use dummy words to keep the structures the same. Record Type 4 has the order of the cosine moment, and has a zero where the angular weight is carried. Finally, the quantities put in the F array are the terms from Equation (2.11). Since the integration routines do not actually use the angular quadrature anywhere within the X2D6 family (the determination of the Legendre coefficients is made outside these routines), many of the variables in a cosine-moments kinematics file are simply placeholders to allow our programs to avoid considering new situations.

Both structures we have described produce collections of group-to-group coefficients that must be combined (e.g., as described in Equation 2.6), in order to determine Legendre coefficients. Two relatively simple collections of subroutines have been written to perform these operations. The X2D7 collection is used when an angular quadrature is used to represent kinematics, while the X2D8 collection is used whenever cosine-moments are employed.

As mentioned before in the description of why we allow two forms of our kinematics file, the treatment of photon coherent scattering employs the cosine-moments procedures.

2.2.2. Incoherent Photon Scattering

The incoherent photon cross-section is described by the following equation:

$$\frac{d\sigma_i(E \to E', \mu)}{d\mu} = S(q;Z)\frac{d\sigma_{KN}(E \to E', \mu)}{d\mu} \tag{2.12}$$

where

$d\sigma_{KN}/d\mu$ = the Klein-Nishina equation,
$S(q;Z)$ = the incoherent scattering cross-section (this approaches Z at high energies and 0 at low energies),
q = the momentum of the recoil electron (in inverse angstrom units).

$$q = \alpha\left[1+\left(\frac{\alpha'}{\alpha}\right)^2 - 2\,\mu\left(\frac{\alpha'}{\alpha}\right)\right]^{1/2} \tag{2.13}$$

where

α = E_γ/m_0c^2,
m_0 = rest mass of the electron,
c = speed of light,
E_γ = incident photon energy,
E'_γ = scattered photon energy, and
μ = cosine of scattering angle.

The Klein-Nishina equation is:

$$\sigma_{KN}(E \to E', \theta)\, d\Omega = \frac{3\,\sigma_{TH}}{16}\left(\frac{E'}{E}\right)^2\left(\frac{E}{E'} + \frac{E'}{E} + \mu^2 - 1\right)d\Omega \tag{2.14}$$

A more useful form is to express it in terms of the cosine of scatter μ. Here we note that

$$d\Omega = -2\sin\theta\, d\theta = -2\, d(\cos\theta) = -2\, d\mu \tag{2.15}$$

Making the substitution into Equation (2.14) yields

$$\sigma_{KN}(E \rightarrow E', \mu)\, d\mu = \frac{3\,\sigma_{TH}}{8}\left(\frac{E'}{E}\right)^2 \left(\frac{E}{E'} + \frac{E'}{E} + \mu^2 - 1\right) d\mu \qquad (2.16)$$

An angular quadrature treatment was initially used to process incoherent gamma-ray scattering; however, the test case selected was the standard SCALE 44-group structure. This group structure is characterized by small groups that isolate photons that are important for some applications. Tests using the approach quickly noted one of the principal weaknesses of the quadrature approach; namely, that for very fine group structures, the angular quadrature must be very dense to allow "seeing" all possible group-to-group transfers. Otherwise, the scattering matrix will contain improper zero-value terms. In point of fact, while these terms are disturbing, they may not be of practical consequence in many cases, since the terms that scatter above and below the zero terms will be slightly larger, to preserve the contributions of the zero values.

In any event, it was decided to abandon the quadrature method for this case and to use the cosine-moments approach that was described for coherent gamma-ray scattering.

If we express photon energy in electron rest mass units; i.e., $e = E_\gamma / 0.5110034$, where E_γ is in terms of MeV, energy and momentum conservation leads to:

$$\frac{e'}{e} = \frac{1}{1 + e\,(1-\mu)} \qquad (2.17)$$

Because we will be integrating the differential incoherent scattering cross section over the sink energies to calculate $\sigma_n(E \rightarrow E')$, the Klein-Nishina cross section is transformed to per unit E' units, by differentiating Equation (2.17) to show

$$d\mu = \frac{de'}{e'^2} = \frac{dE'}{m_0 c^2} \qquad (2.18)$$

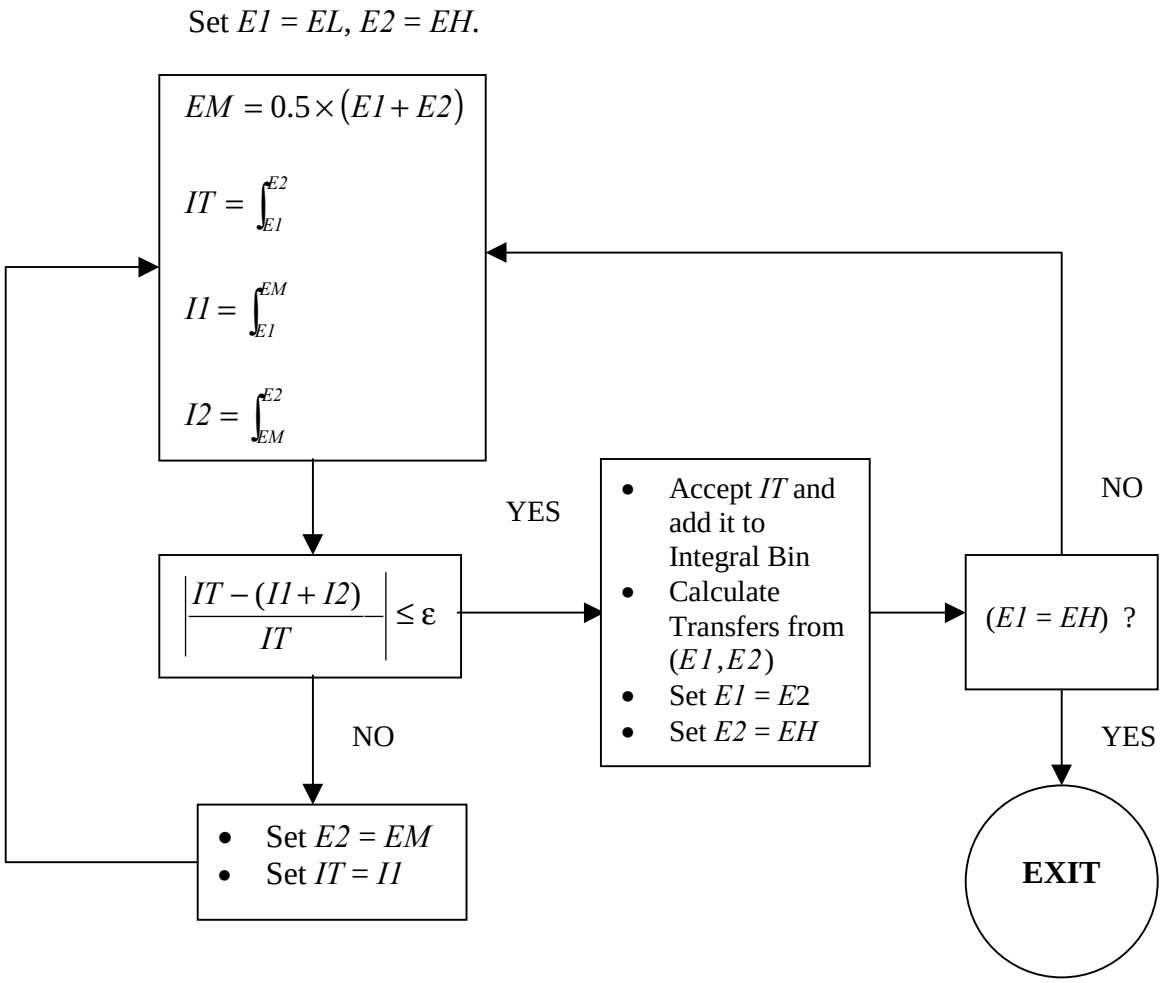
As a result, the Klein-Nishina equation becomes:

$$\sigma_{KN}(E \rightarrow E', \mu)\, dE' = \frac{3\,\sigma_{TH}}{8}\,\frac{1}{E^2 m_0 c^2}\left(\frac{E}{E'} + \frac{E'}{E} + \mu^2 - 1\right) dE' \qquad (2.19)$$

An adaptive scheme is used to generate $\sigma_0(E \rightarrow E')$ by constructing an E' mesh over the [e/(1 + 2 e), e] range that a photon can scatter to. As the E' mesh is generated, the μ values are also tabulated for later use to form $\sigma_n(E \rightarrow E')$ by simply multiplying $\sigma_0(E \rightarrow E')$ by μ^n.

2.2.2.1 Description of AMPX Integration Procedures

To determine averaged cross sections and transfer matrices, AMPX must process four sets of energies: (1) those in the cross section, (2) those in the weighting spectrum, (3) those in the kinematics file, and (4) those in the energy group structure. While a "union" grid is never really formed, the programming examines the energy grids of all functions to locate successive energy bins that do not contain an interior energy point from any function. These (*EL*,*EH*) bins are then subjected to the adaptive procedure described below. Here the precision, ε, is typically set to 0.0001 or 0.01%.



Case 1 (Delta Functions)

Though we have already stated the reasons why we did not use the quadrature method for calculating photon transfer matrices, the method will be described here for reasons of completeness.

In this case the situation can be illustrated as

$$ED_i \ldots\ldots\ldots EL \ldots\ldots\ldots E1 \ldots\ldots\ldots E2 \ldots\ldots\ldots EH \ldots\ldots\ldots ED_{i+1}$$

Here (*E1, E2*) are taken from the union grid of 4 functions, including the kinematics scattering panels at energies, *ED*. By definition (*E1, E2*) is interior to (ED_i, ED_{i+1}). For δ-functions, assume the sink group energy bounds involved are (E'_j, E'_{j+1}), such that our situation looks like:

Source Group Space: $ED_i \ldots\ldots\ldots E1 \ldots\ldots\ldots E2 \ldots\ldots\ldots ED_{i+1}$

Sink Group Space: $E_j' \ldots\ldots\ldots E1' \ldots\ldots\ldots E2' \ldots\ldots\ldots E_{j+1}$

$E1'$ *and* $E2'$ are determined from simple linear interpolations:

$$E1' = E'_j + \frac{E1 - ED_i}{ED_{i+1} - ED_i}(E'_{j+1} - E'_j) \tag{2.20}$$

$$E2' = E'_j + \frac{E2 - ED_i}{ED_{i+1} - ED_i}(E'_{j+1} - E'_j) \tag{2.21}$$

The differential scattering probabilities, F'_j, are tabulated on the E'_j mesh. This requires that we now use a similar scheme to evaluate the (*E1', F1'*), (*E2', F2'*) pairs:

$$F1' = F'_j + \frac{E1' - E'_j}{E'_{j+1} - E'_j}(F'_{j+1} - F'_j) \tag{2.22}$$

$$F2' = F'_j + \frac{E2' - E'_j}{E'_{j+1} - E'_j}(F'_{j+1} - F'_j) \tag{2.23}$$

When (*E1′, E2′*) falls within a single sink group, *g′*, one simply forms the 2-point functions consisting of the products, $\sigma 1 \times \phi 1 \times F1'$ and $\sigma 2 \times \phi 2 \times F2'$, and integrates the panel assuming linear variation, and adds it to the cross section bin for scattering from source group *g* to sink group *g′*. If, however, the (*e1′, e2′*) interval is subtended by one or more sink group boundaries, the integral from *E1′* to *E2′* must be apportioned to two or more groups as follows: assume *T* is the integral just mentioned and that we have the situation:

$$E_{g\text{-}1}' \ldots\ldots\ E1' \ \ldots\ldots\ E_g' \ \ldots\ldots\ E_{g+1}' \ \ldots\ldots\ E_{g+2}' \ \ldots\ldots\ E_{g+3}' \ \ldots\ldots\ E2' \ldots\ldots\ E_{g+4}'$$

In this case, $\sigma(g \rightarrow g')$ would have a piece, $(E_g'\text{-}E1') \times T/(E2'\text{-}E1')$, added to it, $\sigma(g \rightarrow g+1')$ would have $(E_{g+1}'\text{-}E_g') \times T/(E2'\text{-}E1')$ added to it, etc., etc.

Aside from its simplicity, the things to note about this procedure are:

- No "physics" are involved. Those are in the kinematics file.
- The program does not think about "upscatter" or downscatter. The particle scatters in one space, and a particle (the same or another) emerges in another space.
- The procedure inherently preserves balance. The tiny pieces that contribute to the average cross-section value are simply further subdivided to one or more group transfers.

Scattering Treatment for "Non-Delta" Functions

When the secondary energy distributions contain more than one point, a different procedure must be used. It does share the same simplicity and consistency features noted for the procedure just discussed, however.

The procedure is based on the "unit-base transform" approach as will be discussed below.

Assume the adaptive integration procedure has located a segment (*E1*, *E2*) from the union of 4 energy grids. The problem is to apportion the piece(s) of this interval to the group-to-group scattering matrix.

The apportionment is accomplished in a subroutine called XPANEL. The first implementation simply picked the midpoint energy in (*E1*, *E2*) and used the distribution from this point to

calculate the transfers from the interval to the sink energy groups. It was soon realized that there were cases where the scattering distributions did not extend as far down (or up) in energy as they should. From a physical standpoint, this is obvious (e.g., the transfers from *E1* will usually go lower than those from *E2*, and *vice versa*). The next thing tried was to use these two energy points, and to weight them equally. The results were still not satisfactory for some cases. At this point, several Lobatto quadratures were used to assign the energy points in the panel. This study indicated that a 4-point Lobatto quadrature was sufficient to yield results that were virtually indistinguishable from using higher orders. Therefore, XPANEL is presently programmed to use this approach.

Assume a panel representing the scattering from an energy E_i and another panel at energy E_{i+1} are involved. By definition, (*E1*, *E2*) is in (E_i, E_{i+1}). The panel at E_i ranges from EA_1 up to EA_{NA}, while the one at E_{i+1} ranges from EB_1 to EB_{NB}. The scatters from any E in (E_i, E_{i+1}) will cover the range (*E1′, E2′*) which is defined by

$$E1' = EA_1 + \frac{E - E_i}{E_{i+1} - E_i}\left(EB_1 - EA_1\right) \tag{2.24}$$

$$E2' = EA_{NA} + \frac{E - E_i}{E_{i+1} - E_i}\left(EB_{NB} - EA_{NA}\right) \tag{2.25}$$

Two things should be noted about unit-base transform: (1) by definition, the complete panels at E_i and E_{i+1} contribute to every energy in between, and (2) it is unnecessary to actually convert a panel to a (0, 1) space, or, in many cases, to even interpolate to explicitly form intermediate panels. Rather, one can make the unit-base transform into the panels where the data are given, perform the integrations in these panels, and use the calculated values at an intermediate energy by combining the results from the two surrounding panels.

Note that the two panels will generally be probability distributions that will integrate to unity. If we simply take the integrated pieces from the panel at E_i and multiply them by $f = (E - E_i) / (E_{i+1} - E_i)$, and take the integrated pieces in the panel at E_{i+1} and multiply them by $1 - f = (E_{i+1} - E) / (E_{i+1} - E_i)$, then we have multiplied one set that sums to unity by a factor, f, and another set that sums to unity by $1 - f$; hence, the sum at any intermediate point is unity.

When we combine the scattering values from the four energies used with our Lobatto quadrature, we also include an accounting for the variation of the cross section and weighting spectrum in the (*E1*, *E2*) interval. The Lobatto quadrature consists of 4 values of cosine and corresponding weights, μ_m and w_m, respectively. We calculate an approximation of the reaction rate using

$$RR = \sum_{m=1}^{4} \sigma_m \times \phi_m \times w_m \tag{2.26}$$

Here the σ and ϕ values are those corresponding to the four energies taken in (*E1*, *E2*). The scattering terms from *E1* are multiplied by $\sigma 1 \times \phi 1 \times w_1$ and divided by *RR*, etc. Note that this will properly weight the transfers from each energy point, and will, as noted above, preserve the balances we desire, all accomplished without performing any sort of normalization.

Additional Comments Regarding the Unit-Base Transform

The unit-base transform method for performing a two-dimensional interpolation in panels of data such as probability distributions can avoid some of the problems that are associated with Cartesian interpolation. Cartesian interpolation, of course, is the situation wherein one takes two panels at energies E_i and E_{i+1}, and simply interpolates between the values at the same energies in the two panels. This can lead to very non-physical results, unless a very large number of panels is used, and taking a large number of panels is simply impractical for most situations.

The unit-base transform method avoids the problem by transforming each panel to the same (0, 1) x-variable, and then interpolates between corresponding values in the transformed variable, which can easily be converted back to the real variable after the interpolation is made. Note, however, that if all panels use the same energy range, then the interpolation is exactly the same as for Cartesian interpolation. This observation recognizes a potential problem with simple unit-base transform, that can be even worse than simply emulating Cartesian interpolation; viz., the choice of the independent variable limits for a panel mean as much or more than the function values.

In AMPX we use a simple procedure to attempt to avoid including quantitatively meaningless “tails” in the panels. Here we are assuming that a panel can be characterized by a meaningful “real” body, that describes the important range, but may have segments on either end that were included because of the calculational method, or for whatever reason. A routine called LOPPER is used that allows a user to specify a percentage of the integral that can be cut, when the integrals

in the pieces of the distribution are less than the percentage. LOPPER begins by integrating the full distribution. It then looks at both ends of the function, and cuts pieces until the percentage is exceeded. For example, if the percentage is 0.1%, it will only drop pieces on the ends that integrate to less than 0.1% of the total integral. Note that this is a cumulative test. If the first piece integrates to more than 0.1%, no pieces will be cut from that end, etc. This is certainly not perfect, but it does ensure that the "real" part of each distribution is kept. After pieces are cut, the remaining distribution is renormalized to preserve the original integral.

The NJOY system obviously recognized this problem and developed a method for dealing with meaningless tails, and other weaknesses, based on a treatment called the "method of corresponding energies". Here, a panel is integrated, and the energies that delimit a specified fraction of the total integral are determined; for example, if this fraction is 5%, then 21 energy points will specify where the "corresponding energies" are for a panel. All panels are subjected to the same examination. The important thing to note; however, is that the subpanels that are defined by this procedure are then treated by a procedure that is just a unit-base transform. In our example, the first 5% subpanels are used together; the next two 5% subpanels are used together, etc. In other words, 5% bins will define 20 unit-base transform systems.

Around 1990, I read a Chinese paper[7] that described a variation on unit-base transform that demonstrated how one could define multiple regions for making unit-base transforms by locating "characteristic points" in the distributions. In this case, they chose the maxima as the characteristic points, such that a distribution was split into a subpanel that ranged from the first point to the first maxima, the next subpanel from that maxima to the next maxima, etc., up to the last point. All panels were divided this way, and unit-base transforms were used between the corresponding subpanels.

It is easy to imagine how the Chinese approach could be extended to produce an even more physical set of distributions; for example, one could include the minima. The next extension could be to include the inflection points that occur between the maxima and minima, etc.

At present, AMPX has no programming for anything except for its own implementation of unit-base transform; however, all of these alternative approaches can be easily accommodated by the present integration routines without having to modify the coding in Y12 and X10 at all. This would involve yet another way of making use of the "subsection" loops in the kinematics file. It would be very simple to write a set of subroutines that would read the kinematics file, integrate

the distributions, and calculate "corresponding energies" using NJOY's approach, or that could use the Chinese method for matching characteristic features, or use an extension of the Chinese method, and to write a new kinematics file with more subsections (20, for example) and, thereby, accomplish whichever of the approaches one chooses. Since the implementation is thought to be so simple, these alternate approaches will be examined as time permits.

3. RESULTS

It was decided to use carbon to test the gamma production and gamma interaction processing. Carbon was selected for the gamma production, because it was known that carbon was an evaluation that used an anisotropic representation for some reactions, such that it would allow testing this situation. In addition, it was also noted that carbon produced several gamma rays for some reactions, thereby testing whether the code properly summed the contributions.

For testing gamma interaction processing, note that the most important gamma scattering for most applications is the incoherent scattering component, and this is based on the Klein-Nishina scattering that is based on a gamma-ray interaction with an electron; i.e., most of the incoherent scattering is described by the Klein-Nishina equation, and the cross section for any isotope of an element is simply the number of electrons, Z, times the scattering cross section for a single electron. In point of fact, the incoherent cross section tends to zero at very low energies, where the coherent cross section builds up, but these effects are generally not that important except in the sub-KeV energy region.

In both situations, values produced by AMPX were compared with equivalent results calculated by NJOY.

3.1. Gamma Production Testing

For the gamma production testing, a sample problem was selected that used 1/E as a weighting spectrum and employed the standard NJOY 30-group neutron energy structure shown in Table 3 and the standard NJOY 12-group gamma energy structure shown in Table 4.

Table 3 NJOY standard 30 group neutron group energy structure

bdy	energy	bdy	energy	bdy	energy	bdy	energy
1	1.7000E+07	11	1.7380E+06	21	1.2350E+03	31	1.3900E-04
2	1.5000E+07	12	1.3530E+06	22	4.5400E+02		
3	1.3500E+07	13	8.2300E+05	23	1.6700E+02		
4	1.2000E+07	14	5.0000E+05	24	6.1400E+01		
5	1.0000E+07	15	3.0300E+05	25	2.2600E+01		
6	7.7900E+06	16	1.8400E+05	26	8.3200E+00		
7	6.0700E+06	17	6.7600E+04	27	3.0600E+00		
8	3.6800E+06	18	2.4800E+04	28	1.1300E+00		
9	2.8650E+06	19	9.1200E+03	29	4.1400E-01		
10	2.2320E+06	20	3.3500E+03	30	1.5200E-01		

Table 4 NJOY standard 12 group gamma energy structure

bdy	energy	bdy	energy
1	2.0000E+07	11	5.0000E+05
2	9.0000E+06	12	1.0000E+05
3	8.0000E+06	13	1.0000E+04
4	7.0000E+06		
5	6.0000E+06		
6	5.0000E+06		
7	4.0000E+06		
8	3.0000E+06		
9	2.0000E+06		
10	1.0000E+06		

Tables 5 and 6 show the group-averaged neutron cross sections from AMPX and NJOY, respectively. Note that we are testing the gammas that are produced by neutron interactions; hence there is generally an averaged neutron cross section that is multiplied by an average “yield” value to produce the neutron-to-gamma cross-section value.

Table 5 Averaged neutron cross sections produced by AMPX

Neutron Group Parameters

Group	MT= 1099 Flux	MT= 1 Total	MT= 2 Elastic	MT= 3002	MT= 3 NonElastic
1	4.90266E-03	1.44769E+00	9.67657E-01	9.67657E-01	4.80702E-01
2	4.12697E-03	1.35255E+00	8.66368E-01	8.66368E-01	4.87139E-01
3	4.61356E-03	1.38598E+00	8.95890E-01	8.95890E-01	4.90289E-01
4	7.14155E-03	1.29845E+00	7.88342E-01	7.88342E-01	5.10464E-01
5	9.78246E-03	1.33515E+00	8.53670E-01	8.53670E-01	4.81570E-01
6	9.77221E-03	1.23833E+00	9.76671E-01	9.76671E-01	2.61656E-01
7	1.96025E-02	1.56316E+00	1.51004E+00	1.51004E+00	5.38362E-02
8	9.80598E-03	2.24115E+00	2.24114E+00	2.24114E+00	1.57296E-05
9	9.77956E-03	1.69982E+00	1.69981E+00	1.69981E+00	1.03228E-05
10	9.79888E-03	1.81908E+00	1.81908E+00	1.81908E+00	5.54739E-06
11	9.80859E-03	2.00387E+00	2.00387E+00	2.00387E+00	3.46514E-06
12	1.94723E-02	2.50409E+00	2.50408E+00	2.50408E+00	1.60029E-06
13	1.95203E-02	3.12665E+00	3.12665E+00	3.12665E+00	1.08331E-06
14	1.96193E-02	3.63705E+00	3.63705E+00	3.63705E+00	9.73327E-07
15	1.95379E-02	4.01408E+00	4.01408E+00	4.01408E+00	9.34643E-07
16	3.92220E-02	4.36988E+00	4.36988E+00	4.36988E+00	9.13247E-07
17	3.92783E-02	4.59044E+00	4.59044E+00	4.59044E+00	9.63687E-07
18	3.91846E-02	4.67991E+00	4.67991E+00	4.67991E+00	1.06726E-06
19	3.92291E-02	4.71394E+00	4.71393E+00	4.71393E+00	8.58365E-06
20	3.90873E-02	4.72942E+00	4.72941E+00	4.72941E+00	1.50992E-05
21	3.91985E-02	4.73561E+00	4.73559E+00	4.73559E+00	2.01184E-05
22	3.91741E-02	4.73789E+00	4.73786E+00	4.73786E+00	3.29442E-05
23	3.91928E-02	4.73874E+00	4.73869E+00	4.73869E+00	5.43216E-05
24	3.91488E-02	4.73909E+00	4.73900E+00	4.73900E+00	8.95531E-05
25	3.91421E-02	4.73926E+00	4.73911E+00	4.73911E+00	1.47601E-04
26	3.91797E-02	4.73940E+00	4.73916E+00	4.73916E+00	2.43330E-04
27	3.90210E-02	4.73958E+00	4.73918E+00	4.73918E+00	4.00793E-04
28	3.93309E-02	4.73985E+00	4.73919E+00	4.73919E+00	6.60953E-04
29	3.92477E-02	4.74028E+00	4.73919E+00	4.73919E+00	1.09134E-03
30	2.74079E-01	4.64893E+00	4.63623E+00	4.63623E+00	1.26972E-02

Neutron Group Parameters

Group	MT= 4 Inelastic	MT= 51 Level 01	MT= 52 Level 02	MT= 53 Level 03	MT= 54 Level 04
1	4.12041E-01	1.41001E-01	1.11363E-02	7.40834E-02	2.31837E-02
2	4.20291E-01	2.03271E-01	1.84856E-02	6.86688E-02	2.06117E-02
3	3.96309E-01	2.33397E-01	2.70863E-02	4.35473E-02	1.05576E-02
4	4.10992E-01	3.17493E-01	2.29570E-02	1.07614E-02	0.00000E+00
5	3.33016E-01	3.21998E-01	5.97835E-03	0.00000E+00	0.00000E+00
6	2.46000E-01	2.46000E-01	0.00000E+00	0.00000E+00	0.00000E+00
7	5.30912E-02	5.30912E-02	0.00000E+00	0.00000E+00	0.00000E+00
8	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
	Group 9 to Group 29 are identical to Group 8				
30	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00

Neutron Group Parameters

Group	MT= 55 Level 05	MT= 56 Level 06	MT= 57 Level 07	MT= 58 Level 08	MT= 59 Level 09
1	2.03299E-02	3.18436E-02	8.96965E-03	4.42299E-03	2.21374E-03
2	1.26036E-02	8.73830E-03	0.00000E+00	0.00000E+00	0.00000E+00
3	1.49341E-03	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
4	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
	Group 5 to Group 29 are identical to Group 4				
30	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00

Table 5 (continued)

Neutron Group Parameters

Group	MT= 91 Evaporation	MT= 102 n,g	MT= 3102	MT= 103 n,p	MT= 104 n,d
1	9.48564E-02	1.91196E-04	1.91196E-04	7.57331E-03	1.82547E-02
2	8.79116E-02	1.25782E-04	1.25782E-04	1.61077E-04	0.00000E+00
3	8.02278E-02	7.14179E-05	7.14179E-05	0.00000E+00	0.00000E+00
4	5.97807E-02	7.23872E-05	7.23872E-05	0.00000E+00	0.00000E+00
5	5.03960E-03	1.07374E-04	1.07374E-04	0.00000E+00	0.00000E+00
6	0.00000E+00	7.41590E-05	7.41590E-05	0.00000E+00	0.00000E+00
7	0.00000E+00	2.82937E-05	2.82937E-05	0.00000E+00	0.00000E+00
8	0.00000E+00	1.57293E-05	1.57293E-05	0.00000E+00	0.00000E+00
9	0.00000E+00	1.03225E-05	1.03225E-05	0.00000E+00	0.00000E+00
10	0.00000E+00	5.54725E-06	5.54725E-06	0.00000E+00	0.00000E+00
11	0.00000E+00	3.46510E-06	3.46510E-06	0.00000E+00	0.00000E+00
12	0.00000E+00	1.60028E-06	1.60028E-06	0.00000E+00	0.00000E+00
13	0.00000E+00	1.08331E-06	1.08331E-06	0.00000E+00	0.00000E+00
14	0.00000E+00	9.73327E-07	9.73327E-07	0.00000E+00	0.00000E+00
15	0.00000E+00	9.34643E-07	9.34643E-07	0.00000E+00	0.00000E+00
16	0.00000E+00	9.13246E-07	9.13246E-07	0.00000E+00	0.00000E+00
17	0.00000E+00	9.63688E-07	9.63688E-07	0.00000E+00	0.00000E+00
18	0.00000E+00	1.06726E-06	1.06726E-06	0.00000E+00	0.00000E+00
19	0.00000E+00	8.58365E-06	8.58365E-06	0.00000E+00	0.00000E+00
20	0.00000E+00	1.50992E-05	1.50992E-05	0.00000E+00	0.00000E+00
21	0.00000E+00	2.01184E-05	2.01184E-05	0.00000E+00	0.00000E+00
22	0.00000E+00	3.29442E-05	3.29442E-05	0.00000E+00	0.00000E+00
23	0.00000E+00	5.43216E-05	5.43216E-05	0.00000E+00	0.00000E+00
24	0.00000E+00	8.95531E-05	8.95531E-05	0.00000E+00	0.00000E+00
25	0.00000E+00	1.47601E-04	1.47601E-04	0.00000E+00	0.00000E+00
26	0.00000E+00	2.43330E-04	2.43330E-04	0.00000E+00	0.00000E+00
27	0.00000E+00	4.00793E-04	4.00793E-04	0.00000E+00	0.00000E+00
28	0.00000E+00	6.60953E-04	6.60953E-04	0.00000E+00	0.00000E+00
29	0.00000E+00	1.09134E-03	1.09134E-03	0.00000E+00	0.00000E+00
30	0.00000E+00	1.26972E-02	1.26972E-02	0.00000E+00	0.00000E+00

Neutron Group Parameters

Group	MT= 107 n,a	MT= 27 Absorption	MT= 101 Capture
1	4.19738E-02	6.79930E-02	6.79930E-02
2	6.55995E-02	6.58864E-02	6.58864E-02
3	9.37096E-02	9.37811E-02	9.37811E-02
4	9.90414E-02	9.91138E-02	9.91138E-02
5	1.48356E-01	1.48464E-01	1.48464E-01
6	1.55828E-02	1.56570E-02	1.56570E-02
7	0.00000E+00	2.82937E-05	2.82937E-05
8	0.00000E+00	1.57293E-05	1.57293E-05
9	0.00000E+00	1.03225E-05	1.03225E-05
10	0.00000E+00	5.54725E-06	5.54725E-06
11	0.00000E+00	3.46510E-06	3.46510E-06
12	0.00000E+00	1.60028E-06	1.60028E-06
13	0.00000E+00	1.08331E-06	1.08331E-06
14	0.00000E+00	9.73327E-07	9.73327E-07
15	0.00000E+00	9.34643E-07	9.34643E-07
16	0.00000E+00	9.13246E-07	9.13246E-07
17	0.00000E+00	9.63688E-07	9.63688E-07
18	0.00000E+00	1.06726E-06	1.06726E-06
19	0.00000E+00	8.58365E-06	8.58365E-06
20	0.00000E+00	1.50992E-05	1.50992E-05
21	0.00000E+00	2.01184E-05	2.01184E-05
22	0.00000E+00	3.29442E-05	3.29442E-05
23	0.00000E+00	5.43216E-05	5.43216E-05
24	0.00000E+00	8.95531E-05	8.95531E-05
25	0.00000E+00	1.47601E-04	1.47601E-04
26	0.00000E+00	2.43330E-04	2.43330E-04
27	0.00000E+00	4.00793E-04	4.00793E-04
28	0.00000E+00	6.60953E-04	6.60953E-04
29	0.00000E+00	1.09134E-03	1.09134E-03
30	0.00000E+00	1.26972E-02	1.26972E-02

Table 6 Average neutron cross sections produced by NJOY

Neutron Group Parameters

Group	MT= 1 Total	MT= 2 Elastic	MT= 4 Inelastic	MT= 51 Level 01	MT= 52 Level 02
1	1.44836E+00	9.67659E-01	4.12710E-01	1.41005E-01	1.11363E-02
2	1.35351E+00	8.66370E-01	4.21251E-01	2.03273E-01	1.84853E-02
3	1.38618E+00	8.95889E-01	3.96508E-01	2.33399E-01	2.70862E-02
4	1.29880E+00	7.88340E-01	4.11348E-01	3.17495E-01	2.29572E-02
5	1.33525E+00	8.53679E-01	3.33106E-01	3.21997E-01	6.06391E-03
6	1.23829E+00	9.76634E-01	2.45997E-01	2.45997E-01	0.00000E+00
7	1.56388E+00	1.51005E+00	5.38053E-02	5.38053E-02	0.00000E+00
8	2.24113E+00	2.24111E+00	0.00000E+00	0.00000E+00	0.00000E+00
9	1.69981E+00	1.69980E+00	0.00000E+00	0.00000E+00	0.00000E+00
10	1.81909E+00	1.81909E+00	0.00000E+00	0.00000E+00	0.00000E+00
11	2.00388E+00	2.00388E+00	0.00000E+00	0.00000E+00	0.00000E+00
12	2.50409E+00	2.50409E+00	0.00000E+00	0.00000E+00	0.00000E+00
13	3.12665E+00	3.12665E+00	0.00000E+00	0.00000E+00	0.00000E+00
14	3.63705E+00	3.63705E+00	0.00000E+00	0.00000E+00	0.00000E+00
15	4.01409E+00	4.01409E+00	0.00000E+00	0.00000E+00	0.00000E+00
16	4.37034E+00	4.37034E+00	0.00000E+00	0.00000E+00	0.00000E+00
17	4.59044E+00	4.59044E+00	0.00000E+00	0.00000E+00	0.00000E+00
18	4.67991E+00	4.67991E+00	0.00000E+00	0.00000E+00	0.00000E+00
19	4.71394E+00	4.71393E+00	0.00000E+00	0.00000E+00	0.00000E+00
20	4.72943E+00	4.72941E+00	0.00000E+00	0.00000E+00	0.00000E+00
21	4.73562E+00	4.73560E+00	0.00000E+00	0.00000E+00	0.00000E+00
22	4.73791E+00	4.73787E+00	0.00000E+00	0.00000E+00	0.00000E+00
23	4.73880E+00	4.73874E+00	0.00000E+00	0.00000E+00	0.00000E+00
24	4.73923E+00	4.73914E+00	0.00000E+00	0.00000E+00	0.00000E+00
25	4.73965E+00	4.73950E+00	0.00000E+00	0.00000E+00	0.00000E+00
26	4.74047E+00	4.74023E+00	0.00000E+00	0.00000E+00	0.00000E+00
27	4.74248E+00	4.74208E+00	0.00000E+00	0.00000E+00	0.00000E+00
28	4.74774E+00	4.74708E+00	0.00000E+00	0.00000E+00	0.00000E+00
29	4.76176E+00	4.76067E+00	0.00000E+00	0.00000E+00	0.00000E+00
30	8.09965E+00	8.08699E+00	0.00000E+00	0.00000E+00	0.00000E+00

Neutron Group Parameters

Group	MT= 53 Level 03	MT= 54 Level 04	MT= 55 Level 05	MT= 56 Level 06	MT= 57 Level 07
1	7.40834E-02	2.31837E-02	2.03299E-02	3.18433E-02	8.96936E-03
2	6.86694E-02	2.06119E-02	1.26038E-02	9.10426E-03	5.90232E-04
3	4.35472E-02	1.06125E-02	1.63570E-03	0.00000E+00	0.00000E+00
4	1.08826E-02	2.32684E-04	0.00000E+00	0.00000E+00	0.00000E+00
5	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00
	Group 6 to Group 29 are identical to Group 5				
30	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00	0.00000E+00

Neutron Group Parameters

Group	MT= 58 Level 08	MT= 59 Level 09	MT= 91 Evaporation	MT= 102 n,g	MT= 103 n,p
1	4.53593E-03	2.76798E-03	9.48562E-02	1.91196E-04	7.57279E-03
2	0.00000E+00	0.00000E+00	8.79132E-02	1.25784E-04	1.61069E-04
3	0.00000E+00	0.00000E+00	8.02275E-02	7.14181E-05	0.00000E+00
4	0.00000E+00	0.00000E+00	5.97806E-02	7.23863E-05	0.00000E+00
5	0.00000E+00	0.00000E+00	5.04436E-03	1.07373E-04	0.00000E+00
6	0.00000E+00	0.00000E+00	0.00000E+00	7.41606E-05	0.00000E+00
7	0.00000E+00	0.00000E+00	0.00000E+00	2.82940E-05	0.00000E+00
8	0.00000E+00	0.00000E+00	0.00000E+00	1.57295E-05	0.00000E+00
9	0.00000E+00	0.00000E+00	0.00000E+00	1.03226E-05	0.00000E+00
10	0.00000E+00	0.00000E+00	0.00000E+00	5.54730E-06	0.00000E+00
11	0.00000E+00	0.00000E+00	0.00000E+00	3.46511E-06	0.00000E+00
12	0.00000E+00	0.00000E+00	0.00000E+00	1.60027E-06	0.00000E+00
13	0.00000E+00	0.00000E+00	0.00000E+00	1.08331E-06	0.00000E+00
14	0.00000E+00	0.00000E+00	0.00000E+00	9.73327E-07	0.00000E+00
15	0.00000E+00	0.00000E+00	0.00000E+00	9.34643E-07	0.00000E+00
16	0.00000E+00	0.00000E+00	0.00000E+00	9.13249E-07	0.00000E+00

Table 6 (continued)

17	0.00000E+00	0.00000E+00	0.00000E+00	9.63688E-07	0.00000E+00
18	0.00000E+00	0.00000E+00	0.00000E+00	1.06810E-06	0.00000E+00
19	0.00000E+00	0.00000E+00	0.00000E+00	8.58368E-06	0.00000E+00
20	0.00000E+00	0.00000E+00	0.00000E+00	1.50992E-05	0.00000E+00
21	0.00000E+00	0.00000E+00	0.00000E+00	2.01489E-05	0.00000E+00
22	0.00000E+00	0.00000E+00	0.00000E+00	3.30580E-05	0.00000E+00
23	0.00000E+00	0.00000E+00	0.00000E+00	5.44245E-05	0.00000E+00
24	0.00000E+00	0.00000E+00	0.00000E+00	8.99126E-05	0.00000E+00
25	0.00000E+00	0.00000E+00	0.00000E+00	1.47966E-04	0.00000E+00
26	0.00000E+00	0.00000E+00	0.00000E+00	2.44038E-04	0.00000E+00
27	0.00000E+00	0.00000E+00	0.00000E+00	4.01779E-04	0.00000E+00
28	0.00000E+00	0.00000E+00	0.00000E+00	6.62644E-04	0.00000E+00
29	0.00000E+00	0.00000E+00	0.00000E+00	1.09318E-03	0.00000E+00
30	0.00000E+00	0.00000E+00	0.00000E+00	1.26663E-02	0.00000E+00

Neutron Group Parameters

Group	MT= 104 n,d	MT= 107 n,a	MT= 1099 Flux	MT= 27 Absorption	MT= 101 Capture
1	1.82527E-02	4.19742E-02	1.25164E-01	6.79908E-02	6.79908E-02
2	0.00000E+00	6.56001E-02	1.05361E-01	6.58870E-02	6.58870E-02
3	0.00000E+00	9.37123E-02	1.17784E-01	9.37837E-02	9.37837E-02
4	0.00000E+00	9.90426E-02	1.82323E-01	9.91150E-02	9.91150E-02
5	0.00000E+00	1.48356E-01	2.49745E-01	1.48464E-01	1.48464E-01
6	0.00000E+00	1.55815E-02	2.49484E-01	1.56556E-02	1.56556E-02
7	0.00000E+00	0.00000E+00	5.00449E-01	2.82940E-05	2.82940E-05
8	0.00000E+00	0.00000E+00	2.50345E-01	1.57295E-05	1.57295E-05
9	0.00000E+00	0.00000E+00	2.49673E-01	1.03226E-05	1.03226E-05
10	0.00000E+00	0.00000E+00	2.50165E-01	5.54730E-06	5.54730E-06
11	0.00000E+00	0.00000E+00	2.50414E-01	3.46511E-06	3.46511E-06
12	0.00000E+00	0.00000E+00	4.97131E-01	1.60027E-06	1.60027E-06
13	0.00000E+00	0.00000E+00	4.98356E-01	1.08331E-06	1.08331E-06
14	0.00000E+00	0.00000E+00	5.00883E-01	9.73327E-07	9.73327E-07
15	0.00000E+00	0.00000E+00	4.98805E-01	9.34643E-07	9.34643E-07
16	0.00000E+00	0.00000E+00	1.00134E+00	9.13249E-07	9.13249E-07
17	0.00000E+00	0.00000E+00	1.00278E+00	9.63688E-07	9.63688E-07
18	0.00000E+00	0.00000E+00	1.00039E+00	1.06810E-06	1.06810E-06
19	0.00000E+00	0.00000E+00	1.00153E+00	8.58368E-06	8.58368E-06
20	0.00000E+00	0.00000E+00	9.97906E-01	1.50992E-05	1.50992E-05
21	0.00000E+00	0.00000E+00	1.00075E+00	2.01489E-05	2.01489E-05
22	0.00000E+00	0.00000E+00	1.00012E+00	3.30580E-05	3.30580E-05
23	0.00000E+00	0.00000E+00	1.00060E+00	5.44245E-05	5.44245E-05
24	0.00000E+00	0.00000E+00	9.99476E-01	8.99126E-05	8.99126E-05
25	0.00000E+00	0.00000E+00	9.99304E-01	1.47966E-04	1.47966E-04
26	0.00000E+00	0.00000E+00	1.00026E+00	2.44038E-04	2.44038E-04
27	0.00000E+00	0.00000E+00	9.96213E-01	4.01779E-04	4.01779E-04
28	0.00000E+00	0.00000E+00	1.00412E+00	6.62644E-04	6.62644E-04
29	0.00000E+00	0.00000E+00	1.00200E+00	1.09318E-03	1.09318E-03
30	0.00000E+00	0.00000E+00	6.99727E+00	1.26663E-02	1.26663E-02

Neutron Group Parameters

Group	MT= 3102	MT= 3002
1	1.91196E-04	9.67659E-01
2	1.25784E-04	8.66370E-01
3	7.14181E-05	8.95889E-01
4	7.23863E-05	7.88340E-01
5	1.07373E-04	8.53679E-01
6	7.41606E-05	9.76634E-01
7	2.82940E-05	1.51005E+00
8	1.57295E-05	2.24111E+00
9	1.03226E-05	1.69980E+00
10	5.54730E-06	1.81909E+00
11	3.46511E-06	2.00388E+00
12	1.60027E-06	2.50409E+00
13	1.08331E-06	3.12665E+00
14	9.73327E-07	3.63705E+00

Table 6 (continued)

```
15 9.34643E-07 4.01409E+00
16 9.13249E-07 4.37034E+00
17 9.63688E-07 4.59044E+00
18 1.06810E-06 4.67991E+00
19 8.58368E-06 4.71393E+00
20 1.50992E-05 4.72941E+00
21 2.01489E-05 4.73560E+00
22 3.30580E-05 4.73787E+00
23 5.44245E-05 4.73874E+00
24 8.99126E-05 4.73914E+00
25 1.47966E-04 4.73950E+00
26 2.44038E-04 4.74023E+00
27 4.01779E-04 4.74208E+00
28 6.62644E-04 4.74708E+00
29 1.09318E-03 4.76067E+00
30 1.26663E-02 8.08699E+00
```

A review of the numbers in the preceding two tables will show no substantial differences. This was expected to be the case, since group averaging is a simple task.

The ENDF/B-VI carbon evaluation has data for two neutron processes that produce gammas: (1) MT=51, 1st level discrete level inelastic scattering scattering, and (2) MT=102, (n, γ) capture. (Note that the evaluation has data for several more inelastic levels that really do produce gammas; however, data are not included to describe these gammas, nor is it a simple matter to write a code to produce them without additional data). Table 7 and 8 show the averaged "yield" values calculated by AMPX and NJOY, respectively.

Table 7 Gamma yields for MT=51 produced by AMPX

Group 1

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	6.0867E-01	0.0000E+00	-1.3347E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	6.0867E-01	0.0000E+00	-1.3347E-06	0.0000E+00

Group 2

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	6.8308E-01	0.0000E+00	-1.4752E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	6.8308E-01	0.0000E+00	-1.4752E-06	0.0000E+00

Group 3

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	7.7544E-01	0.0000E+00	-1.7221E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	7.7544E-01	0.0000E+00	-1.7221E-06	0.0000E+00

Group 4

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	8.5157E-01	0.0000E+00	-1.7605E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	8.5157E-01	0.0000E+00	-1.7605E-06	0.0000E+00

Group 5

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	6.5226E-01	0.0000E+00	-3.1676E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	6.5226E-01	0.0000E+00	-3.1676E-06	0.0000E+00

Group 6

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	4.6633E-01	0.0000E+00	-1.3326E-06	0.0000E+00
SUM	1.0000E+00	0.0000E+00	4.6633E-01	0.0000E+00	-1.3326E-06	0.0000E+00

Group 7

To	P 0	P 1	P 2	P 3	P 4	P 5
6	1.0000E+00	0.0000E+00	1.3519E+00	0.0000E+00	3.3204E-05	0.0000E+00
SUM	1.0000E+00	0.0000E+00	1.3519E+00	0.0000E+00	3.3204E-05	0.0000E+00

Table 8 MT=51 gamma production cross sections produced by NJOY

Group 1

To	P 0	P 1	P 2
6	1.4100E-01	0.0000E+00	8.5838E-02
SUM	1.4100E-01	0.0000E+00	8.5838E-02

Group 2

To	P 0	P 1	P 2
6	2.0327E-01	0.0000E+00	1.3883E-01
SUM	2.0327E-01	0.0000E+00	1.3883E-01

Group 3

To	P 0	P 1	P 2
6	2.3340E-01	0.0000E+00	1.8104E-01
SUM	2.3340E-01	0.0000E+00	1.8104E-01

Group 4

To	P 0	P 1	P 2
6	3.1750E-01	0.0000E+00	2.7038E-01
SUM	3.1750E-01	0.0000E+00	2.7038E-01

Group 5

To	P 0	P 1	P 2
6	3.2200E-01	0.0000E+00	2.0963E-01
SUM	3.2200E-01	0.0000E+00	2.0963E-01

Group 6

To	P 0	P 1	P 2
6	2.4600E-01	0.0000E+00	1.1479E-01
SUM	2.4600E-01	0.0000E+00	1.1479E-01

Group 7

To	P 0	P 1	P 2
6	5.3805E-02	0.0000E+00	7.2680E-02
SUM	5.3805E-02	0.0000E+00	7.2680E-02

Note that the units of the data in Tables 7 and 8 are different. AMPX calculates in yield units (in this case it is obvious that one gamma is produced, but that it is produced with an anisotropic variation—note the P2 term value). NJOY, on the other hand, calculates the matrix in cross-section units. When one examines the averaged values for MT=51 in Table 6, note that they are identical with the transfer values in Table 8, indicating that the "yield" is unity. Note also that if the P2 term is divided by the cross-section values, the resultant values are essentially identical with those produced by AMPX and given in Table 7.

The "yield" matrices for MT=102 are given in Tables 9 and 10 for AMPX and NJOY, respectively. In this case, both codes calculate the same units. While these results look virtually identical, they are more interesting, because, in this case, three gammas are produced with yields of 0.68, 0.32 and 0.32, respectively, such that the sum of the yields should be 1.32 for all energy groups. An examination of the NJOY output for the transfers from neutron group 1 to all gamma groups showed a value of 1.08, while the remainder of the neutrons groups "correctly" summed to 1.32. Since AMPX showed a sum from group 1 of 1.1, with all other groups summing to 1.32, it was thought that both codes might have a problem. The "problem" is not with the processors, but is rather a combination of the data contained in the ENDF/B evaluation combined with the neutron and gamma energy group structures. The evaluation uses a format that assigns the energy of the gamma ray produced from a neutron reaction as

$$E_\gamma = EG + \frac{AWR}{AWR+1} E_n \qquad (3.1)$$

In this expression, EG is the binding energy, E_n is the energy of the incident neutron, AWR is the atomic weight ratio of the nuclide, and $E_\gamma$is the exit gamma energy. Neutron group 1 ranges from 15 MeV to 17 Mev, while the top of the gamma group structure is 20 MeV. In this case, the binding energy is such that neutron interactions in the first neutron group produce gammas with energies greater than 20 MeV; hence, they are scattering above the top of the gamma group structure.

Table 9 MT=102 gamma yield terms calculated by AMPX

Group 1		Group 2		Group 3		Group 4	
To	P 0	To	P 0	To	P 0	To	P 0
1	7.8383E-01	1	1.0000E+00	1	1.0000E+00	1	1.0000E+00
2	0.0000E+00	2	0.0000E+00	2	0.0000E+00	2	0.0000E+00
3	0.0000E+00	3	0.0000E+00	3	0.0000E+00	3	0.0000E+00
4	0.0000E+00	4	0.0000E+00	4	0.0000E+00	4	0.0000E+00
5	0.0000E+00	5	0.0000E+00	5	0.0000E+00	5	0.0000E+00
6	0.0000E+00	6	0.0000E+00	6	0.0000E+00	6	0.0000E+00
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
SUM	1.1038E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 5		Group 6		Group 7		Group 8	
To	P 0	To	P 0	To	P 0	To	P 0
1	9.0169E-01	1	6.8000E-01	1	4.9792E-01	2	3.1566E-01
2	9.8311E-02	2	1.1078E-01	2	1.8208E-01	3	3.6434E-01
3	0.0000E+00	3	1.8868E-01	3	0.0000E+00	4	0.0000E+00
4	0.0000E+00	4	2.0545E-02	4	1.4012E-01	5	0.0000E+00
5	0.0000E+00	5	0.0000E+00	5	1.3605E-01	6	2.8159E-01
6	0.0000E+00	6	0.0000E+00	6	4.3838E-02	7	3.5841E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	SUM	1.3200E+00
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00		

Group 9		Group 10		Group 11		Group 12	
To	P 0	To	P 0	To	P 0	To	P 0
3	6.8000E-01	3	9.6915E-03	4	6.8000E-01	4	3.1349E-01
4	0.0000E+00	4	6.7031E-01	5	0.0000E+00	5	3.6651E-01
5	0.0000E+00	5	0.0000E+00	6	0.0000E+00	6	0.0000E+00
6	0.0000E+00	6	0.0000E+00	7	3.2000E-01	7	3.2000E-01
7	6.4000E-01	7	5.5353E-01	8	3.2000E-01	8	3.2000E-01
SUM	1.3200E+00	8	8.6470E-02	SUM	1.3200E+00	SUM	1.3200E+00
		SUM	1.3200E+00				

Group 13		Group 14		Group 15		Group 16	
To	P 0	To	P 0	To	P 0	To	P 0
5	6.8000E-01	5	6.8000E-01	5	6.8000E-01	5	6.8000E-01
6	0.0000E+00	6	0.0000E+00	6	0.0000E+00	6	0.0000E+00
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	2.0190E-02	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	2.9981E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 17		Group 18		Group 19		Group 20	
To	P 0	To	P 0	To	P 0	To	P 0
5	1.0151E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
6	5.7849E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
7	3.2000E-01	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
8	0.0000E+00	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
9	3.2000E-01	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00
SUM	1.3200E+00						

Group 21		Group 22		Group 23		Group 24	
To	P 0	To	P 0	To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Table 9 (continued)

Group 25		Group 26		Group 27		Group 28	
To	P 0	To	P 0	To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 29		Group 30	
To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00

Table 10 MT=102 gamma yield values calculated by NJOY

Group 1		Group 2		Group 3		Group 4	
To	P 0	To	P 0	To	P 0	To	P 0
1	7.5960E-01	1	1.0000E+00	1	1.0000E+00	1	1.0000E+00
2	0.0000E+00	2	0.0000E+00	2	0.0000E+00	2	0.0000E+00
3	0.0000E+00	3	0.0000E+00	3	0.0000E+00	3	0.0000E+00
4	0.0000E+00	4	0.0000E+00	4	0.0000E+00	4	0.0000E+00
5	0.0000E+00	5	0.0000E+00	5	0.0000E+00	5	0.0000E+00
6	0.0000E+00	6	0.0000E+00	6	0.0000E+00	6	0.0000E+00
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
SUM	1.0796E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 5		Group 6		Group 7		Group 8	
To	P 0	To	P 0	To	P 0	To	P 0
1	9.0169E-01	1	6.8000E-01	1	4.9792E-01	2	3.1566E-01
2	9.8312E-02	2	1.1078E-01	2	1.8208E-01	3	3.6434E-01
3	0.0000E+00	3	1.8868E-01	3	0.0000E+00	4	0.0000E+00
4	0.0000E+00	4	2.0544E-02	4	1.4012E-01	5	0.0000E+00
5	0.0000E+00	5	0.0000E+00	5	1.3605E-01	6	2.8159E-01
6	0.0000E+00	6	0.0000E+00	6	4.3839E-02	7	3.5841E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	SUM	1.3200E+00
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00		

Group 9		Group 10		Group 11		Group 12	
To	P 0	To	P 0	To	P 0	To	P 0
3	6.8000E-01	3	9.6910E-03	4	6.8000E-01	4	3.1349E-01
4	0.0000E+00	4	6.7031E-01	5	0.0000E+00	5	3.6651E-01
5	0.0000E+00	5	0.0000E+00	6	0.0000E+00	6	0.0000E+00
6	0.0000E+00	6	0.0000E+00	7	3.2000E-01	7	3.2000E-01
7	6.4000E-01	7	5.5353E-01	8	3.2000E-01	8	3.2000E-01
SUM	1.3200E+00	8	8.6470E-02	SUM	1.3200E+00	SUM	1.3200E+00
		SUM	1.3200E+00				

Group 13		Group 14		Group 15		Group 16	
To	P 0	To	P 0	To	P 0	To	P 0
5	6.8000E-01	5	6.8000E-01	5	6.8000E-01	5	6.8000E-01
6	0.0000E+00	6	0.0000E+00	6	0.0000E+00	6	0.0000E+00
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	2.0189E-02	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	2.9981E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 17		Group 18		Group 19		Group 20	
To	P 0	To	P 0	To	P 0	To	P 0
5	1.0151E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
6	5.7849E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
7	3.2000E-01	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
8	0.0000E+00	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
9	3.2000E-01	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00
SUM	1.3200E+00						

Group 21		Group 22		Group 23		Group 24	
To	P 0	To	P 0	To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Table 10 (continued)

Group 25		Group 26		Group 27		Group 28	
To	P 0	To	P 0	To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01	6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01	7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00	8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01	9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00	SUM	1.3200E+00

Group 29		Group 30	
To	P 0	To	P 0
6	6.8000E-01	6	6.8000E-01
7	3.2000E-01	7	3.2000E-01
8	0.0000E+00	8	0.0000E+00
9	3.2000E-01	9	3.2000E-01
SUM	1.3200E+00	SUM	1.3200E+00

3.2. Gamma Interaction Testing

For the gamma interaction testing, a sample problem was selected that used flat weighting and employed the standard AMPX group gamma energy structure shown in Table 11.

Table 11 Standard AMPX 44-group gamma energy structure

bdy	energy	bdy	energy	bdy	energy
1	2.0000E+07	21	1.6600E+06	41	6.0000E+04
2	1.4000E+07	22	1.5700E+06	42	4.5000E+04
3	1.2000E+07	23	1.5000E+06	43	3.0000E+04
4	1.0000E+07	24	1.4400E+06	44	2.0000E+04
5	8.0000E+06	25	1.3300E+06	45	1.0000E+04
6	7.5000E+06	26	1.2000E+06		
7	7.0000E+06	27	1.0000E+06		
8	6.5000E+06	28	8.0000E+05		
9	6.0000E+06	29	7.0000E+05		
10	5.5000E+06	30	6.0000E+05		
11	5.0000E+06	31	5.1200E+05		
12	4.5000E+06	32	5.1000E+05		
13	4.0000E+06	33	4.5000E+05		
14	3.5000E+06	34	4.0000E+05		
15	3.0000E+06	35	3.0000E+05		
16	2.5000E+06	36	2.0000E+05		
17	2.3500E+06	37	1.5000E+05		
18	2.1500E+06	38	1.0000E+05		
19	2.0000E+06	39	7.5000E+04		
20	1.8000E+06	40	7.0000E+04		

Runs were made using AMPX and NJOY. Tables 12 and 13 report the averaged gamma cross sections for AMPX and NJOY, respectively. As expected, no differences of any significance are present.

Table 12 Gamma group averaged cross sections produced by AMPX

Gamma Ray Group Parameters

Group	MT= 1599	MT= 501	MT= 502	MT= 504	MT= 515
1	3.00150E-01	3.27122E-01	6.29725E-08	2.07818E-01	1.22311E-02
2	1.00050E-01	3.54305E-01	2.35731E-07	2.53213E-01	9.46300E-03
3	1.00050E-01	3.76338E-01	4.59992E-07	2.86825E-01	7.86696E-03
4	1.00050E-01	4.08343E-01	1.21371E-06	3.31933E-01	6.03667E-03
5	2.50125E-02	4.35228E-01	2.33709E-06	3.68575E-01	4.84089E-03
6	2.50125E-02	4.47175E-01	2.93457E-06	3.84648E-01	4.31867E-03
7	2.50125E-02	4.63822E-01	4.00392E-06	4.05420E-01	3.79645E-03
8	2.50125E-02	4.81653E-01	6.80400E-06	4.27768E-01	3.27423E-03
9	2.50125E-02	5.02560E-01	9.33672E-06	4.53650E-01	2.75201E-03
10	2.50125E-02	5.29015E-01	1.40829E-05	4.85079E-01	2.22979E-03
11	2.50125E-02	5.55437E-01	1.86580E-05	5.16574E-01	1.71398E-03
12	2.50125E-02	5.86637E-01	2.45588E-05	5.53458E-01	1.22437E-03
13	2.50125E-02	6.28163E-01	3.32542E-05	6.00735E-01	7.73305E-04
14	2.50125E-02	6.80841E-01	4.60246E-05	6.59327E-01	3.95757E-04
15	2.50125E-02	7.46235E-01	6.61722E-05	7.30898E-01	1.32437E-04
16	7.50375E-03	8.02951E-01	8.85767E-05	7.91530E-01	2.97876E-05
17	1.00050E-02	8.35658E-01	1.06051E-04	8.26266E-01	6.75021E-06
18	7.50375E-03	8.68982E-01	1.27344E-04	8.61600E-01	1.56001E-07
19	1.00050E-02	9.14526E-01	1.59634E-04	9.09073E-01	0.00000E+00
20	7.00350E-03	9.62315E-01	2.00593E-04	9.58559E-01	0.00000E+00
21	4.50225E-03	9.94781E-01	2.33163E-04	9.92034E-01	0.00000E+00
22	3.50175E-03	1.01754E+00	2.60360E-04	1.01541E+00	0.00000E+00
23	3.00150E-03	1.04138E+00	2.83891E-04	1.03969E+00	0.00000E+00
24	5.50275E-03	1.07742E+00	3.19882E-04	1.07621E+00	0.00000E+00
25	6.50325E-03	1.12852E+00	3.87351E-04	1.12777E+00	0.00000E+00
26	1.00050E-02	1.21205E+00	5.52010E-04	1.21144E+00	0.00000E+00
27	1.00050E-02	1.33874E+00	8.73528E-04	1.33783E+00	0.00000E+00
28	5.00250E-03	1.45967E+00	1.26758E-03	1.45835E+00	0.00000E+00
29	5.00250E-03	1.55323E+00	1.74397E-03	1.55141E+00	0.00000E+00
30	4.40220E-03	1.66651E+00	2.41902E-03	1.66398E+00	0.00000E+00
31	1.00050E-04	1.72270E+00	2.85714E-03	1.71971E+00	0.00000E+00
32	3.00150E-03	1.76983E+00	3.26079E-03	1.76641E+00	0.00000E+00
33	2.50125E-03	1.86791E+00	4.20120E-03	1.86349E+00	0.00000E+00
34	5.00250E-03	2.01105E+00	6.35361E-03	2.00430E+00	0.00000E+00
35	5.00250E-03	2.28117E+00	1.28096E-02	2.26721E+00	0.00000E+00
36	2.50125E-03	2.57161E+00	2.56234E-02	2.54253E+00	0.00000E+00
37	2.50125E-03	2.85485E+00	5.08723E-02	2.79307E+00	0.00000E+00
38	1.25063E-03	3.15023E+00	1.00564E-01	3.01566E+00	0.00000E+00
39	2.50125E-04	3.30752E+00	1.42083E-01	3.10469E+00	0.00000E+00
40	5.00250E-04	3.41390E+00	1.76320E-01	3.14921E+00	0.00000E+00
41	7.50375E-04	3.66907E+00	2.66879E-01	3.21570E+00	0.00000E+00
42	7.50375E-04	4.37825E+00	4.97080E-01	3.28130E+00	0.00000E+00
43	5.00250E-04	6.49384E+00	9.57461E-01	3.24179E+00	0.00000E+00
44	5.00250E-04	1.95451E+01	1.99669E+00	2.96752E+00	0.00000E+00

Table 12 (continued)

Gamma Ray Group Parameters

Group	MT= 516	MT= 517	MT= 522	MT= 534	MT= 535
1	1.19303E-01	1.07072E-01	5.77605E-07	5.48412E-07	2.91845E-08
2	1.01091E-01	9.16283E-02	7.62925E-07	7.24366E-07	3.85481E-08
3	8.95117E-02	8.16448E-02	9.15919E-07	8.69628E-07	4.62784E-08
4	7.64076E-02	7.03709E-02	1.14603E-06	1.08811E-06	5.79051E-08
5	6.66499E-02	6.18090E-02	1.34725E-06	1.27916E-06	6.80721E-08
6	6.25232E-02	5.82046E-02	1.45375E-06	1.38027E-06	7.34532E-08
7	5.83966E-02	5.46001E-02	1.58075E-06	1.50085E-06	7.98701E-08
8	5.38763E-02	5.06021E-02	1.72825E-06	1.64090E-06	8.73229E-08
9	4.88986E-02	4.61466E-02	1.90083E-06	1.80476E-06	9.60422E-08
10	4.39203E-02	4.16905E-02	2.12697E-06	2.01947E-06	1.07469E-07
11	3.88419E-02	3.71279E-02	2.39930E-06	2.27804E-06	1.21228E-07
12	3.31510E-02	3.19267E-02	2.75873E-06	2.61930E-06	1.39389E-07
13	2.73922E-02	2.66189E-02	3.22939E-06	3.06617E-06	1.63171E-07
14	2.14647E-02	2.10690E-02	3.88951E-06	3.69293E-06	1.96525E-07
15	1.52659E-02	1.51335E-02	4.89529E-06	4.64787E-06	2.47343E-07
16	1.13270E-02	1.12972E-02	5.84432E-06	5.54894E-06	2.95295E-07
17	9.27894E-03	9.27219E-03	6.47148E-06	6.14441E-06	3.26984E-07
18	7.24730E-03	7.24714E-03	7.25250E-06	6.88595E-06	3.66446E-07
19	5.28534E-03	5.28534E-03	8.29167E-06	7.87260E-06	4.18951E-07
20	3.54563E-03	3.54563E-03	9.63501E-06	9.14805E-06	4.86826E-07
21	2.50249E-03	2.50249E-03	1.07787E-05	1.02339E-05	5.44610E-07
22	1.86239E-03	1.86239E-03	1.16909E-05	1.11001E-05	5.90703E-07
23	1.39253E-03	1.39253E-03	1.25716E-05	1.19362E-05	6.35201E-07
24	8.76243E-04	8.76243E-04	1.39656E-05	1.32598E-05	7.05639E-07
25	3.49049E-04	3.49049E-04	1.63011E-05	1.54773E-05	8.23648E-07
26	3.78710E-05	3.78710E-05	2.08792E-05	1.98239E-05	1.05497E-06
27	0.00000E+00	0.00000E+00	3.39297E-05	3.22157E-05	1.71346E-06
28	0.00000E+00	0.00000E+00	5.02511E-05	4.77141E-05	2.53627E-06
29	0.00000E+00	0.00000E+00	7.06876E-05	6.71195E-05	3.56707E-06
30	0.00000E+00	0.00000E+00	1.05272E-04	9.99578E-05	5.31262E-06
31	0.00000E+00	0.00000E+00	1.29751E-04	1.23201E-04	6.54837E-06
32	0.00000E+00	0.00000E+00	1.54666E-04	1.46858E-04	7.80568E-06
33	0.00000E+00	0.00000E+00	2.16557E-04	2.05625E-04	1.09285E-05
34	0.00000E+00	0.00000E+00	3.94104E-04	3.74207E-04	1.98908E-05
35	0.00000E+00	0.00000E+00	1.15641E-03	1.09801E-03	5.83729E-05
36	0.00000E+00	0.00000E+00	3.45793E-03	3.28327E-03	1.74574E-04
37	0.00000E+00	0.00000E+00	1.09147E-02	1.03634E-02	5.51056E-04
38	0.00000E+00	0.00000E+00	3.40071E-02	3.22903E-02	1.71553E-03
39	0.00000E+00	0.00000E+00	6.07438E-02	5.76783E-02	3.06284E-03
40	0.00000E+00	0.00000E+00	8.83688E-02	8.39104E-02	4.45422E-03
41	0.00000E+00	0.00000E+00	1.86491E-01	1.77092E-01	9.38848E-03
42	0.00000E+00	0.00000E+00	5.99872E-01	5.69696E-01	3.01294E-02
43	0.00000E+00	0.00000E+00	2.29459E+00	2.17951E+00	1.14821E-01
44	0.00000E+00	0.00000E+00	1.45809E+01	1.38545E+01	7.23527E-01

Table 12 (continued)

Gamma Ray Group Parameters

Group	MT= 536	MT= 537	MT= 527	MT= 602
1	2.47227E-12	5.69831E-12	0.00000E+00	5.77605E-07
2	3.26547E-12	7.52657E-12	0.00000E+00	7.62925E-07
3	3.92032E-12	9.03593E-12	0.00000E+00	9.15919E-07
4	4.90524E-12	1.13060E-11	0.00000E+00	1.14603E-06
5	5.76650E-12	1.32911E-11	0.00000E+00	1.34725E-06
6	6.22234E-12	1.43418E-11	0.00000E+00	1.45375E-06
7	6.76593E-12	1.55948E-11	0.00000E+00	1.58075E-06
8	7.39725E-12	1.70499E-11	0.00000E+00	1.72825E-06
9	8.13591E-12	1.87525E-11	0.00000E+00	1.90083E-06
10	9.10387E-12	2.09835E-11	0.00000E+00	2.12697E-06
11	1.02695E-11	2.36701E-11	0.00000E+00	2.39930E-06
12	1.18079E-11	2.72160E-11	0.00000E+00	2.75873E-06
13	1.38225E-11	3.18593E-11	0.00000E+00	3.22939E-06
14	1.66479E-11	3.83717E-11	0.00000E+00	3.88951E-06
15	2.09528E-11	4.82941E-11	0.00000E+00	4.89529E-06
16	2.50148E-11	5.76566E-11	0.00000E+00	5.84432E-06
17	2.76992E-11	6.38438E-11	0.00000E+00	6.47148E-06
18	3.10421E-11	7.15489E-11	0.00000E+00	7.25250E-06
19	3.54900E-11	8.18007E-11	0.00000E+00	8.29167E-06
20	4.12397E-11	9.50535E-11	0.00000E+00	9.63501E-06
21	4.61348E-11	1.06336E-10	0.00000E+00	1.07787E-05
22	5.00393E-11	1.15336E-10	0.00000E+00	1.16909E-05
23	5.38089E-11	1.24024E-10	0.00000E+00	1.25716E-05
24	5.97757E-11	1.37777E-10	0.00000E+00	1.39656E-05
25	6.97725E-11	1.60818E-10	0.00000E+00	1.63011E-05
26	8.93694E-11	2.05982E-10	0.00000E+00	2.08792E-05
27	1.51168E-10	3.34304E-10	0.00000E+00	3.39297E-05
28	2.40691E-10	4.96801E-10	0.00000E+00	5.02511E-05
29	3.56480E-10	7.07296E-10	0.00000E+00	7.06876E-05
30	5.60309E-10	1.06959E-09	0.00000E+00	1.05272E-04
31	7.08512E-10	1.33858E-09	0.00000E+00	1.29751E-04
32	8.64837E-10	1.61998E-09	0.00000E+00	1.54666E-04
33	1.26829E-09	2.31430E-09	0.00000E+00	2.16557E-04
34	2.50532E-09	4.46542E-09	0.00000E+00	3.94104E-04
35	8.54571E-09	1.50223E-08	0.00000E+00	1.15641E-03
36	3.02115E-08	5.32100E-08	0.00000E+00	3.45793E-03
37	1.17983E-07	2.12583E-07	0.00000E+00	1.09147E-02
38	4.58751E-07	8.39615E-07	0.00000E+00	3.40071E-02
39	9.22657E-07	1.69963E-06	0.00000E+00	6.07438E-02
40	1.45854E-06	2.70598E-06	0.00000E+00	8.83688E-02
41	3.70982E-06	6.95690E-06	0.00000E+00	1.86491E-01
42	1.61555E-05	3.07548E-05	0.00000E+00	5.99872E-01
43	8.78225E-05	1.69169E-04	0.00000E+00	2.29459E+00
44	9.76262E-04	1.90053E-03	0.00000E+00	1.45809E+01

Table 13 Gamma group averaged cross sections produced by NJOY

Gamma Ray Group Parameters

Group	MT= 501	MT= 502	MT= 504	MT= 516	MT= 527
1	3.27124E-01	2.75906E-06	2.07818E-01	1.19303E-01	5.77606E-07
2	3.54310E-01	4.60083E-06	2.53213E-01	1.01091E-01	7.62927E-07
3	3.76345E-01	6.43197E-06	2.86826E-01	8.95116E-02	9.15921E-07
4	4.08352E-01	9.66025E-06	3.31934E-01	7.64074E-02	1.14603E-06
5	4.35239E-01	1.28557E-05	3.68575E-01	6.66497E-02	1.34725E-06
6	4.47188E-01	1.47195E-05	3.84649E-01	6.25230E-02	1.45375E-06
7	4.63836E-01	1.70196E-05	4.05421E-01	5.83964E-02	1.58076E-06
8	4.81666E-01	1.97922E-05	4.27768E-01	5.38762E-02	1.72826E-06
9	5.02575E-01	2.33787E-05	4.53651E-01	4.88985E-02	1.90083E-06
10	5.29030E-01	2.80765E-05	4.85079E-01	4.39202E-02	2.12698E-06
11	5.55453E-01	3.43506E-05	5.16575E-01	3.88418E-02	2.39930E-06
12	5.86656E-01	4.29666E-05	5.53459E-01	3.31509E-02	2.75873E-06
13	6.28186E-01	5.52420E-05	6.00736E-01	2.73921E-02	3.22940E-06
14	6.80870E-01	7.34770E-05	6.59328E-01	2.14647E-02	3.88952E-06
15	7.46273E-01	1.02980E-04	7.30900E-01	1.52658E-02	4.89530E-06
16	8.02995E-01	1.31457E-04	7.91531E-01	1.13270E-02	5.84434E-06
17	8.35705E-01	1.52622E-04	8.26267E-01	9.27889E-03	6.47150E-06
18	8.69035E-01	1.79141E-04	8.61601E-01	7.24726E-03	7.25252E-06
19	9.14583E-01	2.14739E-04	9.09074E-01	5.28529E-03	8.29169E-06
20	9.62374E-01	2.58850E-04	9.58560E-01	3.54560E-03	9.63504E-06
21	9.94845E-01	2.96015E-04	9.92036E-01	2.50244E-03	1.07787E-05
22	1.01761E+00	3.29469E-04	1.01541E+00	1.86235E-03	1.16910E-05
23	1.04146E+00	3.57966E-04	1.03969E+00	1.39249E-03	1.25717E-05
24	1.07750E+00	4.03001E-04	1.07621E+00	8.76227E-04	1.39657E-05
25	1.12862E+00	4.83612E-04	1.12777E+00	3.49038E-04	1.63012E-05
26	1.21215E+00	6.43997E-04	1.21144E+00	3.78675E-05	2.08792E-05
27	1.33883E+00	9.64689E-04	1.33783E+00	0.00000E+00	3.39298E-05
28	1.45978E+00	1.37958E-03	1.45835E+00	0.00000E+00	5.02513E-05
29	1.55332E+00	1.83852E-03	1.55141E+00	0.00000E+00	7.06879E-05
30	1.66660E+00	2.51030E-03	1.66399E+00	0.00000E+00	1.05272E-04
31	1.72280E+00	2.95187E-03	1.71972E+00	0.00000E+00	1.29753E-04
32	1.76992E+00	3.35676E-03	1.76641E+00	0.00000E+00	1.54667E-04
33	1.86799E+00	4.27921E-03	1.86349E+00	0.00000E+00	2.16558E-04
34	2.01111E+00	6.41492E-03	2.00430E+00	0.00000E+00	3.94106E-04
35	2.28116E+00	1.27945E-02	2.26721E+00	0.00000E+00	1.15641E-03
36	2.57143E+00	2.54409E-02	2.54253E+00	0.00000E+00	3.45795E-03
37	2.85418E+00	5.02003E-02	2.79307E+00	0.00000E+00	1.09148E-02
38	3.14802E+00	9.83602E-02	3.01566E+00	0.00000E+00	3.40073E-02
39	3.30360E+00	1.38164E-01	3.10469E+00	0.00000E+00	6.07442E-02
40	3.40821E+00	1.70634E-01	3.14921E+00	0.00000E+00	8.83693E-02
41	3.65842E+00	2.56231E-01	3.21570E+00	0.00000E+00	1.86492E-01
42	4.35703E+00	4.75857E-01	3.28130E+00	0.00000E+00	5.99875E-01
43	6.47881E+00	9.42416E-01	3.24179E+00	0.00000E+00	2.29460E+00
44	1.96371E+01	2.08855E+00	2.96752E+00	0.00000E+00	1.45810E+01

Table 13 (continued)

Gamma Ray Group Parameters

Group	MT= 1527	MT= 1599
1	4.43351E+00	5.99987E+06
2	3.51743E+00	1.99991E+06
3	3.06793E+00	1.99992E+06
4	2.62555E+00	1.99993E+06
5	2.34710E+00	4.99945E+05
6	2.22847E+00	4.99948E+05
7	2.12357E+00	4.99952E+05
8	2.01234E+00	4.99956E+05
9	1.90099E+00	4.99959E+05
10	1.79561E+00	4.99962E+05
11	1.67368E+00	4.99966E+05
12	1.54754E+00	4.99970E+05
13	1.42666E+00	4.99973E+05
14	1.30174E+00	4.99976E+05
15	1.16587E+00	4.99980E+05
16	1.07842E+00	1.49983E+05
17	1.02417E+00	1.99984E+05
18	9.64513E-01	1.49985E+05
19	9.10307E-01	1.99986E+05
20	8.52968E-01	1.39988E+05
21	8.09306E-01	8.99886E+04
22	7.76722E-01	6.99891E+04
23	7.52720E-01	5.99896E+04
24	7.22022E-01	1.09990E+05
25	6.73562E-01	1.29991E+05
26	6.03133E-01	1.99992E+05
27	5.11299E-01	1.99993E+05
28	4.36860E-01	9.99946E+04
29	3.82105E-01	9.99953E+04
30	3.29918E-01	8.79960E+04
31	3.03016E-01	1.99642E+03
32	2.84640E-01	5.99966E+04
33	2.52256E-01	4.99969E+04
34	2.04113E-01	9.99974E+04
35	1.39178E-01	9.99981E+04
36	8.98556E-02	4.99987E+04
37	5.79874E-02	4.99991E+04
38	3.64433E-02	2.49993E+04
39	2.92908E-02	4.99948E+03
40	2.65735E-02	9.99953E+03
41	2.42096E-02	1.49996E+04
42	2.96936E-02	1.49997E+04
43	5.87086E-02	9.99981E+03
44	1.93986E-01	9.99988E+03

A portion of the results of processing photon coherent scattering is shown in Tables 14 and 15 for AMPX and NJOY, respectively.

Table 14 Part of gamma coherent scattering matrix produced by AMPX

Group 1

To	P 0	P 1	P 2	P 3	P 4	P 5
1	2.7591E-06	8.2772E-06	1.3795E-05	1.9313E-05	2.4831E-05	3.0350E-05
SUM	2.7591E-06	8.2772E-06	1.3795E-05	1.9313E-05	2.4831E-05	3.0350E-05

Group 2

To	P 0	P 1	P 2	P 3	P 4	P 5
2	4.6008E-06	1.3802E-05	2.3004E-05	3.2206E-05	4.1407E-05	5.0609E-05
SUM	4.6008E-06	1.3802E-05	2.3004E-05	3.2206E-05	4.1407E-05	5.0609E-05

Group 3

To	P 0	P 1	P 2	P 3	P 4	P 5
3	6.4319E-06	1.9296E-05	3.2160E-05	4.5024E-05	5.7888E-05	7.0751E-05
SUM	6.4319E-06	1.9296E-05	3.2160E-05	4.5024E-05	5.7888E-05	7.0751E-05

Group 4

To	P 0	P 1	P 2	P 3	P 4	P 5
4	9.6602E-06	2.8981E-05	4.8301E-05	6.7621E-05	8.6942E-05	1.0626E-04
SUM	9.6602E-06	2.8981E-05	4.8301E-05	6.7621E-05	8.6942E-05	1.0626E-04

Group 5

To	P 0	P 1	P 2	P 3	P 4	P 5
5	1.2856E-05	3.8567E-05	6.4278E-05	8.9990E-05	1.1570E-04	1.4141E-04
SUM	1.2856E-05	3.8567E-05	6.4278E-05	8.9990E-05	1.1570E-04	1.4141E-04

Group 40

To	P 0	P 1	P 2	P 3	P 4	P 5
40	1.7063E-01	4.6251E-01	6.4104E-01	7.0577E-01	6.9238E-01	6.3443E-01
SUM	1.7063E-01	4.6251E-01	6.4104E-01	7.0577E-01	6.9238E-01	6.3443E-01

Group 41

To	P 0	P 1	P 2	P 3	P 4	P 5
41	2.5623E-01	6.7791E-01	9.0692E-01	9.6258E-01	9.0971E-01	7.9401E-01
SUM	2.5623E-01	6.7791E-01	9.0692E-01	9.6258E-01	9.0971E-01	7.9401E-01

Group 42

To	P 0	P 1	P 2	P 3	P 4	P 5
42	4.7586E-01	1.1953E+00	1.5045E+00	1.5010E+00	1.3105E+00	1.0083E+00
SUM	4.7586E-01	1.1953E+00	1.5045E+00	1.5010E+00	1.3105E+00	1.0083E+00

Group 43

To	P 0	P 1	P 2	P 3	P 4	P 5
43	9.4241E-01	2.1657E+00	2.5431E+00	2.3126E+00	1.7136E+00	9.8281E-01
SUM	9.4241E-01	2.1657E+00	2.5431E+00	2.3126E+00	1.7136E+00	9.8281E-01

Group 44

To	P 0	P 1	P 2	P 3	P 4	P 5
44	2.0885E+00	4.1060E+00	4.3525E+00	3.0743E+00	1.4431E+00	2.2701E-01
SUM	2.0885E+00	4.1060E+00	4.3525E+00	3.0743E+00	1.4431E+00	2.2701E-01

Table 15 Part of gamma coherent scattering matrix produced by NJOY

Group 1

To	P 0	P 1	P 2	P 3	P 4	P 5
1	2.7591E-06	8.2772E-06	1.3795E-05	1.9313E-05	2.4831E-05	3.0349E-05
SUM	2.7591E-06	8.2772E-06	1.3795E-05	1.9313E-05	2.4831E-05	3.0349E-05

Group 2

To	P 0	P 1	P 2	P 3	P 4	P 5
2	4.6008E-06	1.3802E-05	2.3004E-05	3.2205E-05	4.1407E-05	5.0607E-05
SUM	4.6008E-06	1.3802E-05	2.3004E-05	3.2205E-05	4.1407E-05	5.0607E-05

Group 3

To	P 0	P 1	P 2	P 3	P 4	P 5
3	6.4320E-06	1.9296E-05	3.2160E-05	4.5023E-05	5.7886E-05	7.0748E-05
SUM	6.4320E-06	1.9296E-05	3.2160E-05	4.5023E-05	5.7886E-05	7.0748E-05

Group 4

To	P 0	P 1	P 2	P 3	P 4	P 5
4	9.6603E-06	2.8981E-05	4.8301E-05	6.7620E-05	8.6938E-05	1.0626E-04
SUM	9.6603E-06	2.8981E-05	4.8301E-05	6.7620E-05	8.6938E-05	1.0626E-04

Group 5

To	P 0	P 1	P 2	P 3	P 4	P 5
5	1.2856E-05	3.8567E-05	6.4277E-05	8.9987E-05	1.1569E-04	1.4140E-04
SUM	1.2856E-05	3.8567E-05	6.4277E-05	8.9987E-05	1.1569E-04	1.4140E-04

Group 40

To	P 0	P 1	P 2	P 3	P 4	P 5
40	1.7063E-01	4.7405E-01	6.9177E-01	8.1574E-01	8.6587E-01	8.7105E-01
SUM	1.7063E-01	4.7405E-01	6.9177E-01	8.1574E-01	8.6587E-01	8.7105E-01

Group 41

To	P 0	P 1	P 2	P 3	P 4	P 5
41	2.5623E-01	6.8485E-01	9.4975E-01	1.0624E+00	1.0812E+00	1.0631E+00
SUM	2.5623E-01	6.8485E-01	9.4975E-01	1.0624E+00	1.0812E+00	1.0631E+00

Group 42

To	P 0	P 1	P 2	P 3	P 4	P 5
42	4.7586E-01	1.1538E+00	1.4641E+00	1.5242E+00	1.5039E+00	1.4946E+00
SUM	4.7586E-01	1.1538E+00	1.4641E+00	1.5242E+00	1.5039E+00	1.4946E+00

Group 43

To	P 0	P 1	P 2	P 3	P 4	P 5
43	9.4242E-01	1.9109E+00	2.2840E+00	2.3185E+00	2.3355E+00	2.3774E+00
SUM	9.4242E-01	1.9109E+00	2.2840E+00	2.3185E+00	2.3355E+00	2.3774E+00

Group 44

To	P 0	P 1	P 2	P 3	P 4	P 5
44	2.0886E+00	3.3407E+00	4.2791E+00	4.1654E+00	3.9273E+00	3.5552E+00
SUM	2.0886E+00	3.3407E+00	4.2791E+00	4.1654E+00	3.9273E+00	3.5552E+00

To avoid making this document too large, the previous two tables only list values for energy groups 1 through 5, and then skip to groups 40 through 45. Those numbers left out would add nothing to the simple observation we make: agreement between the two codes is essentially exact at higher energies, and it increasingly diverges as we go to lower energies. The P_0 terms are identical for all cases, but the P_n terms ($n > 0$) differ at lower energies. For energy group 40 (6×10^4—7×10^4 eV), AMPX gives terms:

```
0.17063 0.46251 0.64104 0.70577 0.69238 0.63443
```

while NJOY gives:

```
0.17063 0.47405 0.69177 0.81574 0.86587 0.87105
```

The terms are the coefficients of the Legendre fit to the within-group scattering for this energy group, and will certainly not yield distributions that are very different. In fact, it is impossible without a more detailed study to assign a higher mark-of-merit to either set, except to note that AMPX uses a higher order of angular quadrature in forming its moments. NJOY apparently uses either 6th or 10th order Lobatto, whereas order 16 was selected by AMPX. In an attempt to see how much, if any, difference this would cause, the sample problem was run with AMPX forced to 6th order Lobatto yielding:

```
0.17063 0.47795 0.71338 0.89265 1.0613 1.2644
```

which does not agree much better with the NJOY results than the 16th order Lobatto. Forcing AMPX to use 10th order Lobatto yielded:

```
0.17063 0.45870 0.62462 0.66843 0.63116 0.55513
```

This doesn't agree any better with the NJOY results than the 6th order.

AMPX was then set to use 64th order Lobatto, which yielded:

```
0.17063 0.46236 0.64035 0.70405 0.68925 0.62971
```

The excellent agreement with this and the AMPX 16th order quadrature indicates that the 16th order is well converged, not necessarily that it is more correct than NJOY, however. The fact that there was a noticeable change when we compare 10th and 16th order, does, however, indicate that the differential cross section's angular variation is not tightly converged at order 10. However,

the small differences are probably of no significance at all. Also, note that NJOY does its calculations in double precision, and this definitely affects the accuracy more than using different quadratures.

The results in energy group 44 ($1 \times 10^4 - 2 \times 10^4$), show even more disagreement between AMPX and NJOY. The results of varying the Lobatto quadrature, gives the same behavior as was noted for group 40; however, the substantial differences in individual terms haven't been resolved. Note that other studies with our gamma processing have demonstrated a definite need to use double precision arithmetic at lower energies. At 1×10^4 eV, disagreements are observed between single versus double precision arithmetic of the order of several percent. Extending on down to very low gamma energies ($E \sim 100$ eV), the differences can be factors of 10 or 100 or more. Our future plans include a complete conversion of our ENDF/B processing procedures to double precision, which should eliminate many differences between NJOY and AMPX. (Note that NJOY uses double precision in its later versions, probably to solve the same numerical difficulties we have noted.)

Before we leave the impression that the differences in Legendre coefficients reported by AMPX and NJOY are as significant as the apparent relative disagreements, examine the plot of the two angular distributions shown in Figure 1.

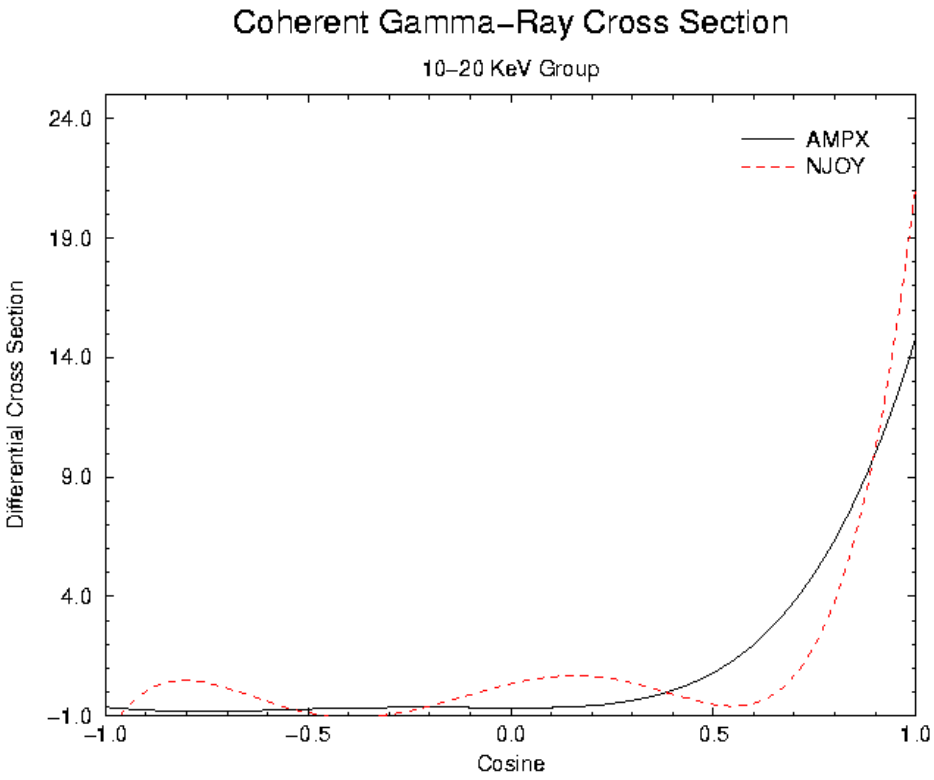


Figure 1 Comparison of AMPX and NJOY distributions for Group 44

As is noted, the two distributions are different, with a sharper peak for the NJOY curve, and more wiggles, since the higher ordered coefficents are significantly larger than those produced by AMPX. On the other hand, both distributions are very forward peaked, and either will probably produce good results, unless, of course, the application is such that the "answer" requires that the differential cross section be fitted as accurately as possible. In any event, until AMPX is converted to double precision, it is felt that it would be premature to look for other causes for the observed difference. One should also note that for typical shielding applications, coherent scattering is not that important, nor are low energy photons. All of the existing multigroup gamma libraries produced at Oak Ridge to date, have simply ignored coherent scattering, and have assumed all scattering was described by the Klein-Nishina equation.

The agreement between AMPX and NJOY in processing photon incoherent scattering is very gratifying, especially since the initial comparisons showed substantial differences.

In the ENDF/B instructions, the Klein-Nishina equation is simply referenced as "well-known" and something a processor should be able to take from the open literature. Because the manner in which the differential cross section was written (in terms of scattering per unit sink energy) in the MUG manual,[8] we elected to use this form of the equation. MUG was the predecessor of the AMPX SMUG module, which has been a part of all previous AMPX system releases. As noted, our original values for producing a 44-group gamma library, showed a scattering distribution that was very different from that produced by NJOY. The AMPX scattering distribution was considerably "harder"; i.e., the terms for the scatters nearer to the group in which the scattering occurs were significantly (factors of 3 to 4) larger than the corresponding NJOY terms, while the terms for the maximum energy transfers seemed to be around a factor of 10 smaller. This led to a series of simple modifications to SMUG to allow it to process ENDF/B-VI. It was subsequently noted that the scattering distributions from SMUG agreed with those from AMPX. At first, this seemed to support AMPX, since the two codes agreed, while NJOY was different. It was then realized that SMUG and AMPX were not independent, since they both started with the same documented equation. This led to some study, which finally showed that the SMUG equation contained an improper energy term that caused the scattering distribution to have the incorrect shape. A simple removal of the term resulted in the values given in Table 16, and which are in excellent agreement with those in Table 17 for NJOY. Note that there is a slight difference in the group 1-to-group 1 term and the group 2-to-group 2 terms, etc., but relative to the total scattering value for groups 1 and 2, respectively, the difference is insignificant. The cause of the difference is not understood at this time, nor is it known which code produces the most correct value. It is simply noted that AMPX uses an adaptive procedure that attempts to construct scattering distributions that are considerable more precise than the difference between the NJOY and AMPX values. It is thought that NJOY also uses an adaptive approach that would also almost assuredly be carried to a precision better than the differences noted. In any event, both codes give results that will yield essentially identical results. It is also noted that any of our existing gamma libraries are based on an incorrect equation, apparently, which give results that have been thought to be acceptable. This may indicate, for example, that photon transport is dominated by the photoelectric effect, or that errors in the scattering distribution produce secondary effects, etc.

Table 16 Part of gamma incoherent scattering matrix produced by AMPX

Group 1

To	P 0	P 1	P 2	P 3	P 4	P 5
1	1.2882E-02	3.8493E-02	6.3643E-02	8.8036E-02	1.1138E-01	1.3341E-01
2	1.1190E-02	3.3281E-02	5.4510E-02	7.4327E-02	9.2219E-02	1.0772E-01
3	1.1818E-02	3.4889E-02	5.6289E-02	7.4985E-02	9.0058E-02	1.0075E-01
4	1.2912E-02	3.7703E-02	5.9464E-02	7.6412E-02	8.7103E-02	9.0533E-02
5	3.5002E-03	1.0126E-02	1.5662E-02	1.9492E-02	2.1159E-02	2.0401E-02
6	3.6499E-03	1.0509E-02	1.6089E-02	1.9685E-02	2.0796E-02	1.9182E-02
7	3.8274E-03	1.0959E-02	1.6581E-02	1.9883E-02	2.0325E-02	1.7705E-02
8	4.0416E-03	1.1497E-02	1.7152E-02	2.0071E-02	1.9682E-02	1.5857E-02
9	4.3018E-03	1.2143E-02	1.7811E-02	2.0223E-02	1.8789E-02	1.3510E-02
10	4.6216E-03	1.2924E-02	1.8564E-02	2.0286E-02	1.7514E-02	1.0469E-02
11	5.0178E-03	1.3871E-02	1.9415E-02	2.0185E-02	1.5685E-02	6.5013E-03
12	5.5237E-03	1.5046E-02	2.0353E-02	1.9750E-02	1.2952E-02	1.1762E-03
13	6.1741E-03	1.6505E-02	2.1346E-02	1.8763E-02	8.9235E-03	-5.9011E-03
14	7.0642E-03	1.8385E-02	2.2247E-02	1.6565E-02	2.5780E-03	-1.5505E-02
15	8.2881E-03	2.0800E-02	2.2831E-02	1.2408E-02	-7.0615E-03	-2.8040E-02
16	2.7744E-03	6.7794E-03	6.8816E-03	2.5823E-03	-4.4624E-03	-1.1125E-02
17	3.9684E-03	9.5024E-03	9.0615E-03	2.1738E-03	-8.1827E-03	-1.6937E-02
18	3.2149E-03	7.5129E-03	6.6131E-03	4.0335E-04	-8.0981E-03	-1.4264E-02
19	4.6824E-03	1.0589E-02	8.2998E-03	-1.6999E-03	-1.3816E-02	-2.0576E-02
20	3.6124E-03	7.8416E-03	5.2082E-03	-3.2226E-03	-1.2002E-02	-1.4878E-02
21	2.4817E-03	5.2348E-03	3.0305E-03	-3.0615E-03	-8.7724E-03	-9.6256E-03
22	2.0234E-03	4.1747E-03	2.1484E-03	-2.9557E-03	-7.3583E-03	-7.3858E-03
23	1.8102E-03	3.6464E-03	1.6359E-03	-2.9926E-03	-6.6100E-03	-6.0356E-03
24	3.5515E-03	6.8320E-03	2.2288E-03	-6.8618E-03	-1.2528E-02	-9.3288E-03
25	4.6637E-03	8.2934E-03	9.2453E-04	-1.0819E-02	-1.5002E-02	-6.5683E-03
26	8.2376E-03	1.3177E-02	-2.6234E-03	-2.2675E-02	-2.2811E-02	8.1101E-04
27	9.9256E-03	1.3034E-02	-9.9453E-03	-2.9287E-02	-1.5651E-02	1.9136E-02
28	5.9929E-03	5.6332E-03	-9.9766E-03	-1.5170E-02	1.9317E-03	1.7733E-02
29	6.8766E-03	4.4871E-03	-1.4238E-02	-1.3611E-02	1.1529E-02	2.0668E-02
30	7.0544E-03	1.9034E-03	-1.6616E-02	-6.2406E-03	1.9497E-02	1.0908E-02
31	1.7444E-04	5.5491E-06	-4.1918E-04	-2.2309E-05	5.1692E-04	5.2434E-05
32	5.6184E-03	-1.1671E-03	-1.2882E-02	3.3925E-03	1.4242E-02	-4.5692E-03
33	5.3204E-03	-3.3927E-03	-1.0575E-02	9.8389E-03	7.2310E-03	-1.3292E-02
34	1.2967E-02	-1.8216E-02	-8.1109E-03	3.0195E-02	-1.6443E-02	-1.7797E-02
35	8.0546E-03	-2.0946E-02	2.7525E-02	-3.1444E-02	3.9822E-02	-5.6273E-02
SUM	2.0782E-01	3.6205E-01	4.3013E-01	4.1959E-01	4.7514E-01	3.4842E-01

Group 2

To	P 0	P 1	P 2	P 3	P 4	P 5
2	6.6042E-03	1.9760E-02	3.2758E-02	4.5496E-02	5.7871E-02	6.9785E-02
3	1.8305E-02	5.4515E-02	8.9535E-02	1.2260E-01	1.5299E-01	1.8004E-01
4	1.9326E-02	5.6943E-02	9.1508E-02	1.2115E-01	1.4423E-01	1.5942E-01
5	5.0911E-03	1.4862E-02	2.3430E-02	3.0082E-02	3.4243E-02	3.5507E-02
6	5.2471E-03	1.5245E-02	2.3794E-02	3.0057E-02	3.3385E-02	3.3366E-02
7	5.4437E-03	1.5727E-02	2.4256E-02	3.0047E-02	3.2376E-02	3.0851E-02
8	5.6807E-03	1.6307E-02	2.4809E-02	3.0031E-02	3.1183E-02	2.7929E-02
9	5.9691E-03	1.7008E-02	2.5465E-02	2.9986E-02	2.9724E-02	2.4453E-02
10	6.3366E-03	1.7892E-02	2.6258E-02	2.9845E-02	2.7781E-02	2.0057E-02
11	6.8037E-03	1.8995E-02	2.7189E-02	2.9510E-02	2.5131E-02	1.4433E-02
12	7.4042E-03	2.0381E-02	2.8248E-02	2.8804E-02	2.1399E-02	7.1045E-03
13	8.1959E-03	2.2148E-02	2.9397E-02	2.7386E-02	1.5937E-02	-2.6495E-03
14	9.2678E-03	2.4424E-02	3.0497E-02	2.4623E-02	7.7291E-03	-1.5667E-02
15	1.0770E-02	2.7394E-02	3.1200E-02	1.9333E-02	-4.7961E-03	-3.2658E-02
16	3.6465E-03	8.9638E-03	9.2788E-03	3.9207E-03	-5.0597E-03	-1.3760E-02
17	5.2045E-03	1.2544E-02	1.2224E-02	3.5618E-03	-9.7605E-03	-2.1368E-02
18	4.1764E-03	9.8671E-03	9.0100E-03	1.2974E-03	-9.7014E-03	-1.8262E-02
19	6.0473E-03	1.3872E-02	1.1444E-02	-1.0010E-03	-1.6905E-02	-2.6942E-02
20	4.6261E-03	1.0247E-02	7.3855E-03	-3.0881E-03	-1.4902E-02	-2.0181E-02
21	3.1836E-03	6.8346E-03	4.3059E-03	-3.3185E-03	-1.0982E-02	-1.3037E-02

Table 16 (continued)

22	2.6127E-03	5.4548E-03	3.0019E-03	-3.4729E-03	-9.3054E-03	-9.8154E-03
23	2.3382E-03	4.7730E-03	2.3135E-03	-3.6245E-03	-8.5053E-03	-8.1257E-03
24	4.5277E-03	8.9823E-03	3.5963E-03	-8.2194E-03	-1.6833E-02	-1.4121E-02
25	5.8758E-03	1.0940E-02	2.4745E-03	-1.2922E-02	-2.0960E-02	-1.2763E-02
26	1.0523E-02	1.7281E-02	-2.1222E-03	-2.8202E-02	-3.0909E-02	-3.2388E-03
27	1.2696E-02	1.7260E-02	-1.1602E-02	-3.7914E-02	-2.2986E-02	2.2185E-02
28	7.5980E-03	7.7957E-03	-1.1916E-02	-2.1058E-02	-7.5993E-04	2.4145E-02
29	8.9063E-03	5.7609E-03	-1.8067E-02	-1.6930E-02	1.3708E-02	2.3845E-02
30	9.0895E-03	2.7267E-03	-2.1375E-02	-8.8745E-03	2.5050E-02	1.5372E-02
31	2.2482E-04	1.4077E-05	-5.4197E-04	-4.7738E-05	6.7060E-04	8.8508E-05
32	7.1959E-03	-9.3934E-04	-1.7307E-02	3.1624E-03	2.1313E-02	-5.8028E-03
33	6.8294E-03	-3.9498E-03	-1.4004E-02	1.1103E-02	1.1759E-02	-1.4559E-02
34	1.6793E-02	-2.3136E-02	-1.2057E-02	4.2364E-02	-2.7436E-02	-1.4120E-02
35	1.0674E-02	-2.7149E-02	3.3158E-02	-3.2231E-02	3.4613E-02	-4.9180E-02
SUM	2.5321E-01	4.2974E-01	4.9754E-01	4.8345E-01	5.1129E-01	3.9233E-01

Group 40

To	P 0	P 1	P 2	P 3	P 4	P 5
40	1.1039E+00	2.2406E+00	1.5161E+00	2.2069E-01	-3.2818E-01	-2.9069E-01
41	2.0494E+00	-1.6633E+00	-8.6445E-02	-3.2236E-01	1.8827E-01	-8.3762E-02
SUM	3.1533E+00	5.7727E-01	1.4297E+00	-1.0167E-01	-1.3991E-01	-3.7445E-01

Group 41

To	P 0	P 1	P 2	P 3	P 4	P 5
41	2.2548E+00	1.5147E+00	8.8751E-01	1.5033E-01	-8.8152E-01	-6.9259E-02
42	9.6406E-01	-1.1197E+00	4.8901E-01	-3.5214E-01	6.9091E-01	-3.7108E-01
SUM	3.2189E+00	3.9506E-01	1.3765E+00	-2.0181E-01	-1.9061E-01	-4.4034E-01

Group 42

To	P 0	P 1	P 2	P 3	P 4	P 5
42	2.8816E+00	3.8458E-01	1.1932E+00	-3.9566E-01	-3.6752E-01	-4.4154E-01
43	3.9959E-01	-3.3592E-01	6.4906E-02	3.7744E-02	1.0364E-01	-7.5999E-02
SUM	3.2812E+00	4.8656E-02	1.2581E+00	-3.5792E-01	-2.6388E-01	-5.1754E-01

Group 43

To	P 0	P 1	P 2	P 3	P 4	P 5
43	3.0410E+00	-2.7762E-01	1.1148E+00	-3.1782E-01	-1.0059E-01	-4.5485E-01
44	2.0084E-01	-3.1019E-02	7.0640E-02	-2.6981E-02	-1.0517E-02	-3.1992E-02
SUM	3.2418E+00	-3.0863E-01	1.1854E+00	-3.4480E-01	-1.1111E-01	-4.8685E-01

Group 44

To	P 0	P 1	P 2	P 3	P 4	P 5
44	2.8963E+00	-6.6206E-01	9.3644E-01	-4.9700E-01	-2.5045E-01	-7.0806E-01
SUM	2.8963E+00	-6.6206E-01	9.3644E-01	-4.9700E-01	-2.5045E-01	-7.0806E-01

Table 17 Part of gamma incoherent scattering matrix produced by NJOY

Group 1

To	P 0	P 1	P 2	P 3	P 4	P 5
1	1.4599E-02	4.3645E-02	7.2239E-02	1.0009E-01	1.2690E-01	1.5242E-01
2	1.1268E-02	3.3517E-02	5.4917E-02	7.4925E-02	9.3031E-02	1.0878E-01
3	1.1810E-02	3.4871E-02	5.6285E-02	7.5028E-02	9.0191E-02	1.0101E-01
4	1.2895E-02	3.7667E-02	5.9458E-02	7.6505E-02	8.7371E-02	9.1038E-02
5	3.4952E-03	1.0117E-02	1.5666E-02	1.9532E-02	2.1256E-02	2.0572E-02
6	3.6448E-03	1.0500E-02	1.6096E-02	1.9732E-02	2.0909E-02	1.9372E-02
7	3.8229E-03	1.0953E-02	1.6594E-02	1.9941E-02	2.0453E-02	1.7910E-02
8	4.0363E-03	1.1490E-02	1.7169E-02	2.0144E-02	1.9835E-02	1.6094E-02
9	4.2941E-03	1.2133E-02	1.7831E-02	2.0313E-02	1.8978E-02	1.3794E-02
10	4.6094E-03	1.2906E-02	1.8589E-02	2.0407E-02	1.7764E-02	1.0830E-02
11	5.0002E-03	1.3845E-02	1.9447E-02	2.0348E-02	1.6012E-02	6.9456E-03
12	5.4935E-03	1.5001E-02	2.0404E-02	2.0004E-02	1.3436E-02	1.7753E-03
13	6.1306E-03	1.6444E-02	2.1429E-02	1.9133E-02	9.5766E-03	-5.1902E-03
14	6.9788E-03	1.8275E-02	2.2436E-02	1.7280E-02	3.6774E-03	-1.4620E-02
15	8.1547E-03	2.0645E-02	2.3174E-02	1.3543E-02	-5.5182E-03	-2.7198E-02
16	2.7491E-03	6.7614E-03	6.9861E-03	2.8482E-03	-4.1735E-03	-1.1104E-02
17	3.9391E-03	9.4927E-03	9.2170E-03	2.5142E-03	-7.8896E-03	-1.7093E-02
18	3.1934E-03	7.5130E-03	6.7468E-03	6.6183E-04	-7.9347E-03	-1.4516E-02
19	4.6403E-03	1.0597E-02	8.5699E-03	-1.2529E-03	-1.3699E-02	-2.1354E-02
20	3.5584E-03	7.8462E-03	5.5227E-03	-2.7285E-03	-1.1960E-02	-1.5903E-02
21	2.4467E-03	5.2417E-03	3.2418E-03	-2.7498E-03	-8.7915E-03	-1.0353E-02
22	2.0005E-03	4.1871E-03	2.3022E-03	-2.7641E-03	-7.4449E-03	-7.9513E-03
23	1.7895E-03	3.6666E-03	1.7869E-03	-2.8548E-03	-6.7977E-03	-6.6245E-03
24	3.4815E-03	6.9074E-03	2.7188E-03	-6.5428E-03	-1.3398E-02	-1.1268E-02
25	4.5039E-03	8.4586E-03	1.9835E-03	-1.0245E-02	-1.7093E-02	-1.0667E-02
26	7.9823E-03	1.3495E-02	-9.0546E-04	-2.2087E-02	-2.6716E-02	-5.4626E-03
27	9.7799E-03	1.3442E-02	-8.9527E-03	-3.0267E-02	-1.9100E-02	1.8338E-02
28	5.8643E-03	6.1101E-03	-9.3247E-03	-1.7011E-02	-7.4335E-04	2.0665E-02
29	6.7864E-03	4.9143E-03	-1.3937E-02	-1.5423E-02	1.0298E-02	2.4599E-02
30	7.0067E-03	2.2640E-03	-1.6813E-02	-7.7002E-03	2.0552E-02	1.4358E-02
31	1.7327E-04	1.5945E-05	-4.3193E-04	-5.5717E-05	5.7921E-04	1.0912E-04
32	5.5514E-03	-6.1169E-04	-1.3760E-02	2.1198E-03	1.8207E-02	-4.0903E-03
33	5.2447E-03	-2.7457E-03	-1.1847E-02	9.0408E-03	1.2268E-02	-1.5863E-02
34	1.2914E-02	-1.7434E-02	-1.1210E-02	3.5777E-02	-2.0702E-02	-1.8522E-02
35	7.9814E-03	-1.9917E-02	2.1997E-02	-1.3793E-02	3.4287E-04	1.0862E-02
SUM	2.0782E-01	3.7222E-01	4.3562E-01	4.5441E-01	4.4968E-01	4.3169E-01

Group 2

To	P 0	P 1	P 2	P 3	P 4	P 5
2	8.6639E-03	2.5940E-02	4.3063E-02	5.9931E-02	7.6444E-02	9.2507E-02
3	1.8436E-02	5.4919E-02	9.0236E-02	1.2364E-01	1.5442E-01	1.8193E-01
4	1.9249E-02	5.6733E-02	9.1220E-02	1.2087E-01	1.4406E-01	1.5947E-01
5	5.0691E-03	1.4805E-02	2.3360E-02	3.0036E-02	3.4258E-02	3.5619E-02
6	5.2241E-03	1.5186E-02	2.3727E-02	3.0025E-02	3.3430E-02	3.3524E-02
7	5.4152E-03	1.5657E-02	2.4184E-02	3.0031E-02	3.2473E-02	3.1102E-02
8	5.6513E-03	1.6236E-02	2.4744E-02	3.0037E-02	3.1321E-02	2.8234E-02
9	5.9441E-03	1.6950E-02	2.5419E-02	3.0012E-02	2.9874E-02	2.4750E-02
10	6.3100E-03	1.7833E-02	2.6223E-02	2.9902E-02	2.7983E-02	2.0413E-02
11	6.7723E-03	1.8930E-02	2.7164E-02	2.9612E-02	2.5414E-02	1.4881E-02
12	7.3652E-03	2.0306E-02	2.8240E-02	2.8974E-02	2.1798E-02	7.6655E-03
13	8.1415E-03	2.2051E-02	2.9420E-02	2.7683E-02	1.6536E-02	-1.9270E-03
14	9.1871E-03	2.4300E-02	3.0596E-02	2.5157E-02	8.6339E-03	-1.4828E-02
15	1.0651E-02	2.7246E-02	3.1459E-02	2.0239E-02	-3.5746E-03	-3.2056E-02
16	3.5746E-03	8.8859E-03	9.4714E-03	4.4916E-03	-4.4088E-03	-1.3679E-02
17	5.1105E-03	1.2450E-02	1.2501E-02	4.3254E-03	-8.9628E-03	-2.1438E-02
18	4.1345E-03	9.8358E-03	9.1639E-03	1.6593E-03	-9.4045E-03	-1.8486E-02
19	5.9962E-03	1.3852E-02	1.1678E-02	-5.5322E-04	-1.6704E-02	-2.7584E-02
20	4.5903E-03	1.0242E-02	7.5709E-03	-2.7876E-03	-1.4876E-02	-2.0829E-02
21	3.1529E-03	6.8377E-03	4.4756E-03	-3.0849E-03	-1.1056E-02	-1.3703E-02
22	2.5762E-03	5.4599E-03	3.2027E-03	-3.2125E-03	-9.4279E-03	-1.0613E-02

Table 17 (continued)

23	2.3034E-03	4.7800E-03	2.5083E-03	-3.3859E-03	-8.6546E-03	-8.9138E-03
24	4.4784E-03	9.0032E-03	3.8896E-03	-7.9153E-03	-1.7176E-02	-1.5369E-02
25	5.7891E-03	1.1025E-02	3.0298E-03	-1.2643E-02	-2.2128E-02	-1.5013E-02
26	1.0252E-02	1.7602E-02	-3.9642E-04	-2.7786E-02	-3.5153E-02	-9.4561E-03
27	1.2554E-02	1.7586E-02	-1.0728E-02	-3.8765E-02	-2.6209E-02	2.1233E-02
28	7.5281E-03	8.0418E-03	-1.1622E-02	-2.2100E-02	-2.0618E-03	2.5974E-02
29	8.7154E-03	6.5404E-03	-1.7617E-02	-2.0363E-02	1.2148E-02	3.1946E-02
30	9.0049E-03	3.1464E-03	-2.1475E-02	-1.0666E-02	2.5843E-02	1.9765E-02
31	2.2279E-04	2.6367E-05	-5.5436E-04	-9.2044E-05	7.4020E-04	1.7995E-04
32	7.1415E-03	-5.9914E-04	-1.7732E-02	2.0783E-03	2.3558E-02	-4.0171E-03
33	6.7538E-03	-3.3582E-03	-1.5408E-02	1.1116E-02	1.6425E-02	-1.9703E-02
34	1.6667E-02	-2.2067E-02	-1.5431E-02	4.6279E-02	-2.4857E-02	-2.6193E-02
35	1.0588E-02	-2.6277E-02	2.8622E-02	-1.7268E-02	-6.7448E-04	1.4985E-02
SUM	2.5321E-01	4.4010E-01	5.0421E-01	5.1547E-01	5.0003E-01	4.7036E-01

Group 40

To	P 0	P 1	P 2	P 3	P 4	P 5
40	1.1823E+00	2.2915E+00	1.3827E+00	1.1176E-01	-2.5975E-01	-1.8825E-01
41	1.9669E+00	-1.6837E+00	1.9793E-02	-2.1197E-01	-4.5765E-02	-1.1496E-01
SUM	3.1492E+00	6.0781E-01	1.4025E+00	-1.0021E-01	-3.0552E-01	-3.0320E-01

Group 41

To	P 0	P 1	P 2	P 3	P 4	P 5
41	2.2774E+00	1.5056E+00	9.1779E-01	6.0592E-02	-3.4815E-01	-3.2743E-01
42	9.3827E-01	-1.0944E+00	4.1912E-01	-2.7172E-01	-9.2046E-03	3.8334E-03
SUM	3.2157E+00	4.1117E-01	1.3369E+00	-2.1112E-01	-3.5736E-01	-3.2360E-01

Group 42

To	P 0	P 1	P 2	P 3	P 4	P 5
42	2.8553E+00	5.9378E-01	1.0297E+00	-2.3100E-01	-4.0509E-01	-3.4090E-01
43	4.2603E-01	-5.1354E-01	1.9998E-01	-1.2257E-01	9.3568E-04	3.4703E-03
SUM	3.2813E+00	8.0245E-02	1.2296E+00	-3.5357E-01	-4.0415E-01	-3.3743E-01

Group 43

To	P 0	P 1	P 2	P 3	P 4	P 5
43	2.9662E+00	3.7993E-02	9.7223E-01	-3.9311E-01	-4.5126E-01	-3.8588E-01
44	2.7559E-01	-3.4436E-01	1.4029E-01	-8.0187E-02	2.5267E-03	1.7382E-03
SUM	3.2418E+00	-3.0637E-01	1.1125E+00	-4.7330E-01	-4.4873E-01	-3.8414E-01

Group 44

To	P 0	P 1	P 2	P 3	P 4	P 5
44	2.9072E+00	-5.8317E-01	8.6382E-01	-6.3363E-01	-5.5618E-01	-4.5372E-01
SUM	2.9072E+00	-5.8317E-01	8.6382E-01	-6.3363E-01	-5.5618E-01	-4.5372E-01

4. SUMMARY

The new AMPX does not contain independent codes to replace the LAPHNGAS and SMUG modules that are part of all prior releases of the AMPX system. LAPHNGAS was used to produce photon production data arising from neutron interactions, and the SMUG module was used to produce photon interaction data.

In the new coding, the gamma production and gamma interaction kinematics data are produced in the Y12 module, just as is the neutron interaction data. The group-averaged parameters and transfer matrices are also produced in the X10 modules for all situations.

This report describes the physics and procedures associated with processing photon data. Several important observations derive from the study:

1. AMPX and NJOY produce equivalent results for all cases studied, except for low energy coherent photon scattering, where AMPX is known to have some numerical difficulties that should be corrected when the codes are converted to double precision.

2. The calculational times for the two code systems are roughly the same. (It is noted that none of the photon processing is complicated enough to require large time expenditures).

3. The SMUG module contains an error in its expression for the Klein-Nishina equation. Brian Broadhead,[9] who has worked with the SCALE system for several years and has performed photon transport studies using both MCNP and SCALE, was interviewed to determine if he had noted anything that suggest the error could have adversely affected his results. A review of his comparisons indicates excellent agreement between MCNP and SCALE for some fairly deep penetration studies involving transport through iron. This would suggest that the error does not manifest itself in causing SCALE to improperly calculate photon transport; however, eliminating the error should not change the results by any significant amount, since MCNP photon transport is based on NJOY processed results.

5. REFERENCES

1. N. M. Greene, J. L. Lucius, L. M. Petrie, W. E. Ford, III, J. E. White and R. Q. Wright, *AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B*, ORNL/TM-3706, Union Carbide Corp., Nucl. Div., Oak Ridge National Laboratory, March 1976.

2. *AMPX-II, Modular Code System for Generating Coupled Multigroup Neutron Gamma-Ray Cross-Section Libraries From Data in ENDF Format*, RSIC documentation for PSR-63, November 30, 1978.

3. N. M. Greene, W. E. Ford, III, L. M. Petrie, J. W. Arwood, *AMPX-77: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Cross-Section Libraries from ENDF/B-IV and/or ENDF/B-V*, ORNL/CSD/TM-283, October 1992.

4. V. McLane, C. L. Danford, and P. F. Rose, *ENDF-102: Data Formats and Procedures for the Evaluated Nuclear Data File ENDF-6*, Brookhaven National Laboratory, July 1990. (Revised 1998).

5. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, III, May 2000. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.

6. R. E. MacFarlane and D. W. Muir, *The NJOY Nuclear Data Processing System—Version 91*, LA-12740-M, October 1994.

7. C. Shao-Hui, *A New Interpolation Method for Two-dimensional Functions in Evaluated Nuclear Data Library*, 1990.

8. J. R. Knight and F. R. Mynatt, *MUG: A Program for Generating Multi-Group Photon Cross-sections*, CTC-17, Union Carbide Corp., Nucl. Div., Oak Ridge National Laboratory, January 1970.

9. B. L. Broadhead, personal communication, Oak Ridge National Laboratory, Oak Ridge, Tennessee, February 2001.

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