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KAOS-V CODE: AN EVALUATION TOOL FOR NEUTRON
KERMA FACTORS AND OTHER NUCLEAR RESPONSES

by

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ABSTRACT

The accurate evaluation of neutron fluence-to-kerma (Kinetic Energy Released in Materials) factors from microscopic nuclear data is the subject of this report. The algorithms developed for this purpose combine in a consistent manner the two basic methods for computing kerma factors, namely kinematics and direct energy balance. These algorithms are implemented in the code KAOS-V which was used as the main evaluation tool to construct the response function library KAOS/LIB-V. KAOS-V uses data from the evaluated nuclear data files ENDF/B-V. Auxiliary nuclear data bases, e.g., the Japanese evaluated nuclear data library JENDL-2 can be used as a source of isotopic cross sections when these data are not provided in ENDF/B-V files for a natural element. These are needed mainly to estimate average quantities such as effective Q-values for the natural element. The code has the ability to compare the different results which aids in the choice of a consistent set of algorithms to evaluate kerma factors. Data in ENDF/B-IV and ENDF/B-V format can be processed. For resonance treatment, the code has the ability to access NJOY and NPTXS interface files in formatted or binary forms. No input instructions are necessary to run the code interactively. The user can simply respond to the interactive messages sent by the code if an INPUT file is absent. An INPUT file is automatically generated following an interactive run, and can be edited and used to rerun or produce

different results. Grouped and point output data can be produced along with graphic representation. These features are instrumental in detecting and understanding energy balance deficiencies and other problems in the nuclear data files.

The KAOS-V code is written in standard FORTRAN-77. It runs on the CRAY machines of the National Magnetic Fusion Energy Computer Center NMFECC at Lawrence Livermore National Laboratory.

CHAPTER I

GENERAL BACKGROUND

I.1 Introduction

Heating is an important parameter of any nuclear system. The energy multiplication in a fusion reactor blanket is a very important factor in the assessment of the economic performance of the power plant. More important are the severe design constraints due to the undesirable heating effects. This is particularly true for the plasma facing components where excessive heating is expected. Other fusion reactor components where heating effects are important are the superconducting magnets. The accurate calculation of heating in the magnet due to the leaking neutrons and photons is a key factor in the design of the cryogenic heat removal systems. Accurate neutron kerma factors are necessary for heating calculations and are also in increasing demand for biomedical applications. Direct experimental measurements of kerma factors are limited to a few isotopes and cover the small range of high incident neutron energies of particular importance to radiation therapy. The microscopic cross section data bases, especially the ENDF/B files [1] provide the means for computing kerma factors for many materials and for incident neutron energies up to 20 MeV. The computed kerma factors from microscopic cross sections however, suffer from large uncertainties that cruelly manifest themselves sometimes as negative values [2,3]. It is generally

believed - justifiably so - that the fundamental improvement in the computed kerma factors rests on improving the microscopic cross section data. The huge inertia which is involved in adjusting the massive nuclear data files and the fact that these are generally satisfactory for their main purpose - transport calculations - make it difficult to take this route.

Thus, there is a need for improved methods in calculating nuclear heating parameters from the existing data pool. The work presented in this report is a response to this need, especially for the fusion community.

I.2 Formulation of the Nuclear Heating Problem

Two methods for calculating nuclear heating are discussed in this section. The key characteristics of each method are presented.

I.2.1 Macroscopic Method

Heating is calculated by performing the energy balance in a macroscopic volume [4]. The net current of neutrons (and gammas) across the zone boundaries multiplied by their respective energies is calculated using transport codes. Adding this term to the sum of the different reaction rates inside the zone multiplied by their respective Q-values along with a similar term for the decay energy will result in the total heating rate in the macroscopic volume. This method has no fundamental advantage as compared with kerma methods. A

modified energy balance method was introduced by G. Woodruff and D. Quimby [5]. Macroscopic energy balance was used earlier by Abdou [6] as an integral test of the validity of transport cross sections and kerma factors generated by the computer program MACK [7].

I.2.2 Kerma Method

The calculation of many nuclear responses takes the form

$$R = \int_P F_R(P) \phi(P) dP \quad (I.1)$$

where $\phi(P)$ is the neutron (or gamma) flux at the point P in the space-energy phase space, and $F_R(P)$ is the response function corresponding to the response R .

Heating as a nuclear response at the spatial point r is the sum of the neutron heating $H_n(r)$ and the gamma heating $H_\gamma(r)$, where

$$H_n(r) = \sum_j N_j(r) \int \phi_n(r, E) K_{nj}(E) dE \quad (I.2)$$

and

$$H_\gamma(r) = \sum_j N_j(r) \int \phi_\gamma(r, E) K_{\gamma j}(E) dE. \quad (I.3)$$

$N_j(r)$ is the number density of nuclide j at r , $\phi_n(r, E)$ is the neutron flux at point r and energy E , and $\phi_\gamma(r, E)$ is the gamma flux at point r and energy E . Both neutron and gamma fluxes for a nuclear system are obtained by performing transport calculations. The heating response functions $K_{nj}(E)$ and $K_{\gamma j}(E)$ are known as "fluence-to-kerma factors" or "kerma factors" for short. The term "kerma" is an acronym for the

Kinetic Energy Released in Materials. The neutron kerma factor is the rate of energy released locally in the material on collision with a nuclide j with unit number density due to a unit flux of neutrons of energy E . Thus,

$$K_{nj}(E) = \sum_i K_{nij}(E) \quad (I.4)$$

where

$$K_{nij}(E) = \sigma_{ij}(E) h_{nij}(E) . \quad (I.5)$$

The K_{nij} are known as partial kerma factors, $\sigma_{ij}(E)$ is the microscopic cross section for reaction i and nuclide j at neutron energy E , and $h_{nij}(E)$ is the energy released locally due to that reaction.

The mechanisms of gamma interactions with materials are well-understood and present no problem. Thus, our problem reduces to improving the accuracy of the neutron kerma factors, or rather the energy release parameters $h_{nij}(E)$.

I.3 Kerma Factor Computing Methods

A general outline of the methods used to compute neutron kerma factors is presented here. There are basically two methods, direct energy balance and reaction kinematics. The following description of the methods and their limitations will help provide a background for understanding previous works.

1.3.1 Direct Energy Balance Method

The prompt energy released locally due to the reaction of type i , a neutron of energy E , and a nuclide j is obtained from:

$$h_{ij} = E + Q_{ij} - \bar{E}_{n_{ij}} - \bar{E}_{\gamma_{ij}} \quad (I.6)$$

Q_{ij} is the Q-value of the reaction $\bar{E}_{n_{ij}}$ is the average energy carried away by all secondary neutrons produced in the reaction. $\bar{E}_{\gamma_{ij}}$ is the same for gamma photons. A decay energy term should be added to h_{ij} to get the total energy release to account for local energy deposition due to the decay of excited reaction products. Equation (I.6) is an accurate expression for the energy release and implies no simplifying approximations.

However, using equation (I.6) to compute h_{ij} does not always produce good results for several reasons explained below.

The Precision Problem

The magnitude of h_{ij} is usually much smaller than that of the individual terms on the right hand side of equation (I.6). A small uncertainty in E_{γ} which might be accepted on its own accord, will result in a large uncertainty in h . Quite often, negative values of h are calculated [2] which is physically prohibited, of course. The negative numbers are, by no means, small in absolute value [3]. There is no assurance that the positive numbers are correct either. The required precision of the secondary neutron and gamma energies

required to apply equation (I.6) - at least for some reactions - can be impractical, especially where these data are produced mainly for transport calculations and they are sufficiently accurate for that purpose. A common argument in favor of using equation (I.6) with bad or even negative kerma is that the error will be compensated for by the subsequent heating contribution of secondary neutrons and photons. This can be true only for a large system which has dimensions much larger than transport mean distances.

The Nature of Some Data

For reactions, e.g., radiative capture, where the component of recoil energy due to gamma photons emission is important, another difficulty arises with using equation (I.6). If secondary photon energies are calculated from the ENDF/B-V data files by analyzing excitation energy levels and decay schemes, no account of the small effect of the recoil energy on the energy available for the photons is considered. Even if excitation energy levels and transition probabilities are absolutely accurate, equation (I.6) will underestimate heating by missing the important component of recoil energy due to photon emission. This is particularly true for low incident neutron energies (below ~10 keV) as explained later in section II.2.1.

Data Violating Energy Balance

The use of equation (I.6) readily implies the assumption that the required data are consistent and they conform to the energy conservation principle. In fact, this is not always the case as pointed out by MacFarlane in his prize-winning paper "Energy Balance of ENDF/B-V" [8], with more details in reference [9].

Natural Element Problem

For natural elements and isotope mixtures, the single Q-value [10] for each reaction provided in ENDF/B data is not sufficient. The effective Q-value for an element is the average of Q-values of its constituent isotopes weighted by their cross sections as:

$$Q_{\text{eff}}(E) = \frac{\sum_j N_j \sigma_j(E) Q_j}{\sum_j N_j \sigma_j(E)} \quad (I.7)$$

where N_j is the number density of the isotope j . Q_{eff} is energy-dependent and the lack of individual isotopic cross sections destroys the usefulness of equation (I.6).

I.3.2 Reaction Kinematics Methods

The prompt heat released locally is the sum of the energies of the secondary charged particles and the recoil nucleus.

$$h_{ij} = E_{rij} + E_{xij} \quad (I.8)$$

where $E_{r ij}$ is the average kinetic energy of the recoil nucleus following a reaction of type i with the target nuclide j . $E_{x ij}$ is the sum of the average secondary charged particles' kinetic energies. The secondary particles' average energies are calculated by solving energy and momentum balance equations. Usually, but not always, the expressions for recoil and secondary energies, unlike equation (I.6), do not suffer the precision problems. This is the main advantage of the kinematics methods. On the other hand, the accuracy is affected by simplifying assumptions whose severity varies from one reaction type to another. This statement can be equivalently posed in terms of the unsatisfied demand for certain data. For many-body reactions, for example, correlation data are required for exact kinematics calculations [11]. At present, there is no angular correlation data in the ENDF/B files, while energy correlation data are limited to the class of reactions that proceed via the formation of an excited intermediate state [1]. Another problem that limits the accuracy of kinematics methods is the absence of detailed partial cross sections for the various residual nuclei excited states following charged particle emission for most materials. Simple models used to estimate the residual excitation affect the accuracy for some calculations of neutron kerma factors, e.g., the energy-averaging model [12]. The need for residual excitation is equivalent to the need for secondary particles' spectra. Except for natural silicon, secondary charged particles' spectra are not provided in ENDF/B-V.

Like the direct method, but to a lesser extent, the unavailability of isotopic data for some natural elements would limit the accuracy of kinematics analysis for some reaction types.

I.4 History of Previous Kerma Factor Calculations

The early efforts to calculate fluence-to-kerma factors start with J. Ritts, M. Solomito, and P. Stevens. They calculated kerma for some nuclides composing the human body [13]. That work was extended by Ritts, Solomito, and Steiner [14] to include a few materials of interest in thermonuclear blanket assemblies. The new elements considered were Li-6, Li-7, Be, C, P, Cu, and Nb, making a total of 19 materials in their kerma factor library. Ritts et al., used simplified reaction kinematics. However, the methodology they developed passed to subsequent works. For his thesis dissertation [6], M. Abdou made an important contribution in the subject development. His analysis considered more reaction details and also included decay energy. To avoid duplicating the effort for new materials or new evaluation data for a material, he developed the computer code MACK [7]. The code MACK was designed to process ENDF/B-III data files and create a nuclear response function library; MACKLIB [15]. With the introduction of ENDF/B-IV, which contained secondary photon production files for some materials, an alternative direct energy balance became available. The new version, MACK-IV [10] was developed by Abdou, Gohar, and Wright. MACK-IV computes kerma factors

following either a direct photon production path, i.e., a direct energy balance method, or the old traditional kinematics carried over from the previous version. It is interesting to note that the residual excitation in the indirect gamma path, i.e., kinematics method was recognized as an important problem by the authors of MACK-IV. The code MACK-IV was used to generate the library MACKLIB-IV [16], which contains - among other response functions - energy release parameters (kerma factors) for 49 materials. The use of the indirect-gamma path to generate kerma factors for MACKLIB-IV was limited to materials for which the direct gamma production path produces any negative values for kerma. In this case, the indirect gamma production path was used to compute kerma factors for this material in the entire incident neutron energy range covered by ENDF/B-IV. The "elemental" Q-value problem was also discussed, but not considered for the actual generation of kerma factor results for the obvious reason of the lack of data.

Extending the earlier work by R. Bach and Caswell [17], R. Caswell, J. Coyne, and M. Randolph [12] developed a model for calculating neutron kerma factors for biological and medical applications. They produced values for 19 light elements or nuclides: H, Li-6, Li-7, B, C, N, O, F, Na, Mg, Al, Si, P, S, Cl, Ar, K, Ca, and Fe. The chief source of nuclear data was ENDF/B-IV. For carbon, they used a preliminary ENDF/B-V evaluation, while ENDL files supplied these data for S, P, and Ar, and the UKAEA file was used for B-11. To

extend their work to the incident neutron energy of 30 MeV, they used an optical model calculation normalized to the available experimental data or fitted to the other data they used at 20 MeV. The elemental Q-value problem was not considered in the model. Reaction kinematics was used to calculate recoil energies for reactions that do not produce charged particles. A simple energy averaging algorithm was developed to account for residual nucleus excitation by estimating the probability of exciting different levels which lead to gamma ray emission. The emitted gamma energy does not contribute to kerma. In the energy averaging algorithm, an effective Coulomb threshold above the energetic threshold of $0.1 \text{ } zZ \text{ MeV}$ for exciting each level was assumed, where z is the atomic number of the lighter emitted particle and Z is the atomic number of the residual nucleus after particle emission. The probability of exciting levels with energy above the threshold rises linearly from zero at the Coulomb threshold up to 2 MeV above that energy. For higher energies, the excitation of all accessible levels is assumed equally probable. Caswell et al., however, suggested that the energy-averaging procedure should be replaced as soon as possible with experimentally measured or more reliably calculated final state cross sections. Compared to ENDF/B-IV data where excitation energies could be obtained, the authors reported an agreement within 40%.

The quantities closely related to fluence-to-kerma factors, namely the total and local energy deposits per collision were included

in the tabulated 175 neutron group constants derived from the ENDL library [18]. The local energy deposits are equivalent to neutron kerma factors divided by the total microscopic cross section, while the total energy deposits include the produced gamma energy. For local energy deposits arising from reactions in which charged particles are emitted, it was necessary to estimate the average energy of the final-state charged particles. Because of the absence of experimental determination of these data for most reactions, a gross estimate was made of the average charged-particle energy [19] such that:

$$\begin{aligned}
 E_{ch} &= \frac{1}{2} \sqrt{E^*} \quad , \quad Z \geq 29 \\
 &= \frac{1}{4} E^* \quad , \quad 7 < Z < 29 \\
 &= E^* \quad , \quad Z \leq 7
 \end{aligned} \tag{I.9}$$

where E^* is the available energy after the average neutron energy has been removed, and Z is the target nucleus atomic number. The difference between the local and total energy deposits represents the energy carried away by gamma rays. To ensure the calculational consistency between these quantities, energy deposits are used by R. Howerton et al. [20] as input data for calculating the photon production cross sections and spectra. In other words, the reason why the energy deposits calculated from the ENDL library by direct energy balance are never negative is that they reflect the way photon production was evaluated and the approximate estimation of energy

deposits reappears. Kerma factors from the December 1985 version of the ENDL data file were calculated by R. Howerton [20] and tabulated in the 175 neutron group structure used by the TART Monte Carlo neutron transport code. The materials included are those with atomic number $Z < 30$ and the incident neutron energy ranges between 1.307×10^{-9} and 20.0 MeV. As in the previous set, and for the same reasons, no negative kerma factors were found.

The local neutron heating in the units of MeV/collision is listed as a continuous function of incident neutron energy in the Recommended Monte Carlo Cross Section library RMCCS [21] and the Alternate Monte Carlo Cross Section library AMCCS [22] supplying cross section data to the Monte Carlo Neutron-Photon transport code MCNP [23]. The local neutron heating was calculated by direct energy balance from data supplied by ENDF/B-III, ENDF/B-IV, ENDL-73, ENDL-75, and LASL data libraries. There are 26 out of 68 evaluation on RMCCS which have at least one negative heating number. However, of the 26 sets, with negative heating numbers, only 8 have photon production and neutron cross sections from different sources. This problem is much more pronounced on the AMCCS library. The latest version of the RMCCS library to be used with the MCNP code version 3A depends on data from the ENDF/B-V. The MCNP code, however, does not use the negative heating numbers, but sets them to zero when they are encountered - the matter that casts some doubt on the accuracy of final heating results for some applications and leads to their overestimation in the limiting case of large, uniform, nuclear systems.

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The nuclear data processing system NJOY [24,25] produces almost all types of nuclear data and provides for a convenient interface format for linking data to application codes. One of its modules, HEATR [25], computes neutron kerma factors. HEATR utilizes the direct energy balance method for all reactions except for radiative capture, where the kinematics method was used in its last update. The neutron kerma factors generated by NJOY are stored in MATXS libraries [26] with different group structures. The negative kerma factors are kept and not zeroed out. It has been argued that this will preserve the global energy balance and produce the correct heating source in large uniform media. Elemental Q-values for some materials were derived by R. MacFarlane [27] from microscopic cross sections found in the activation file ACTL [28].

I.5 Status of Present Nuclear Data

The nuclear data files used in this work are briefly mentioned in this section. The key characteristics required to improve the kerma factor evaluation based on ENDF/B are discussed.

I.5.1 The ENDF Library

The most important data source is the "Evaluated Nuclear Data Files" ENDF [1]. The ENDF system was designed for the storage and retrieval of the evaluated nuclear data that are required for neutronics, photonics, and decay calculations. This system is

composed of several parts that include neutron cross sections, photon cross sections, and nuclear structure data with the associated processing codes (appendix I in reference [1]). The evaluated data in ENDF are related to the experimental data library CSISRS (Cross Section Information Storage and Retrieval System). The experimental cross section data are supplemented by other nuclear data, such as spins, energies, and parities of excited states. The experimental data are then analyzed, and in some cases, the results are combined with predictions from nuclear model calculations [29] to obtain the recommended cross sections. The recommended cross sections are then encoded in the ENDF format for subsequent incorporation into either ENDF/A or ENDF/B libraries. ENDF/A and ENDF/B are two different evaluated data libraries maintained at the NNDC (National Nuclear Data Center). The ENDF/A library contains either complete or partial data sets and may contain more than one evaluation for a particular nuclide. The ENDF/B library, on the other hand, contains only one complete evaluation of the cross sections for each material in the library. The data set selected for the ENDF/B library is the set recommended by the CSEWG (Cross Section Evaluation Working Group); a cooperative effort of many laboratories coordinated by the NNDC. The last version of ENDF/B is ENDF/B-V. Two revisions [29] have been made to update ENDF/B-V for important materials in advance of the more comprehensive changes that will accompany a complete version change to ENDF/B-VI.

Energy Balance of ENDF/B-V

ENDF/B-V contains photon production data for 24 fissionable and 36 nonfissionable materials. MacFarlane [8] pointed out that kinematics limits of secondary photon energies (or the combined neutron and photon energies) are often violated. The tests he made showed surprisingly large energy balance errors. For example, as reported by R. MacFarlane [9], the energy of secondary photons produced by thermal neutron radiative capture by Mn-55 is overestimated by 53.7%; while for Co-59, inelastic scattering photon energy was reported to be in error by 250% in the neutron energy range of 1.2-3 MeV.

In reference [2], 25 materials, for which at least one of the heating numbers is negative, are listed, indicating energy balance problems. On the top of their priority list, we find P-31, Cr, Mn-55, Fe, Co-59, Ni, Cu, Nb-93, Mo, W-182, W-183, W-184 and W-186. In the new evaluation for iron [30] made by Arthur and Young of Los Alamos, energy balance considerations are taken into account. MacFarlane and Foster [31] reported dramatic improvements in iron heating numbers. They used the preequilibrium statistical model code GNASH to calculate the isotopic cross sections for iron and coupled the results to the code RECOIL.

Other problems with ENDF/B data, as far as heating calculations are concerned, can be summarized as:

- * There are no isotopic data for some natural elements.

- * Only a few secondary charged particle spectra are given.
- * No angular correlations and incomplete energy correlations between secondary particles are found in the data files.
- * Reactions with small cross sections are ignored. These reactions are usually endoergic with high Q-values. The product of the cross section and the Q-value is not always small. The direct energy balance should produce overestimated neutron kerma factors. The neglect of the Na-23(n,np) reaction [32] is a good example where such an effect could account for energy imbalance rather than the well-studied gamma production.
- * Reactions leading to the emission of a secondary neutron and charged particle sometimes misrepresent the secondary charged particle. This error, in particular, does not affect transport calculations and would not be detected without attention to energy balance considerations. A minor energy imbalance was noticed, for example, when the ENDF/B-V flag LR was set to indicate the decay of the 6.029 MeV excited inelastic level of B-10 by alpha emission without regard to the competition by photon emission [32], while the emission of such photons are considered in the gamma file. The need for branching information in the LR flag is therefore recognized. A large energy imbalance, however, was detected in F-19 [32] because the LR flag was set to zero for the ENDF/B-V reactions MT=64-91, indicating deexcitation by gamma emission, while the dominant mode of

deexcitation is actually alpha particle emission. The comparison between excitation energy and gamma energy production [9] falsely indicated underestimated gamma production.

* The gamma spectra are sometimes truncated where the low-energy photons are not represented. The truncation of photons below 0.3 MeV in Mn-55 and Cr [33] are two examples. Most experimental techniques do not permit measurement of the photon spectra for photon energies less than ~ 0.5 MeV because of interferences such as the 0.51 MeV annihilation radiation [34]. The truncation of low-energy photons obviously produces an underestimated gamma energy production. On the other hand, the normalization of the truncated spectrum to produce an accurate gamma multiplicity, should however overestimate the average gamma energy because the spectrum will be shifted to higher gamma energies.

* Interpolation problems arise when photon production data as a function of the incident neutron energy are widely-spaced [35]. When a linear interpolation is prescribed, for instance, for both gamma multiplicity and average energy, the resulting gamma production between data points will take a presumably unintended parabolic shape. This scalloped behavior appears also in linear-linear plots and is not merely due to plotting in log-log scale, as pointed out in reference [21]. An illustration of this interpolation problem is given in Figure (I.1). Another interpolation problem arises when the gamma production is represented by a coarse histogram. Energy balance

Figure I.1
Sketch of the linear-parabolic interpolation problem.

problems and negative kerma factors arise when the width of a histogram-interpolated gamma production encompasses many neutron groups, resulting in the loss of resolution. An illustration of this problem is given in Figure (I.2).

I.5.2 The ENDL Library

The Evaluated Nuclear Data Library ENDL is maintained at the Lawrence Livermore Laboratory [19, 36-43]. The last version, ENDL82 [41] contains complete evaluations for 94 isotopes and natural elements. Explicit representations are given for energy and/or angular distributions for all secondary particles from neutron-induced reactions. The average energies of the secondary particles are also given. With the above features added, it has been claimed possible to insure energy conservation among secondary neutrons, charged particles, photons, and residual nuclei to within 10% or 100 keV. As pointed out earlier, the energies of the secondary charged particles were derived from approximate nuclear systematics. These secondary particles' average energies were used as an input in evaluating photon production cross sections. The use of direct energy balance to compute kerma factors from these data will actually reproduce the kerma factors already estimated by the approximate models.

I.5.3 The JENDL Library

The Japanese Evaluated Nuclear Data Library (JENDL) has been compiled since 1974. For the last available edition; JENDL-2, evalua-

I.2
Gamma production histogram representation problem.

tions [44] were prepared by JNDC (Japanese Nuclear Data Committee) for 181 nuclides. These data are compiled in the ENDF/B-IV format. No gamma production files were included in JENDL. The emphasis on isotopic evaluations was very helpful in providing for the needed isotopic cross sections to prepare estimates for variable elemental Q-values used in this work. It should be noted however, that cross section data from the JENDL-2 and the ENDF/B-V libraries were sometimes not in good agreement.

I.5.4 Other Data Sources

Other nuclear data libraries include INDL/V [45] which is the IAEA (International Atomic Energy Agency) Nuclear Data Library for various nuclear data evaluations in ENDF-5 format, and KEDAK-4 [46] the Karlsruhe evaluated neutron data library. For nuclear structure and decay data, the two editions of the Table of Isotopes [47,48] were extensively used throughout this work.

CHAPTER II

THE KERMA FACTOR EVALUATION MODEL

II.1 Introduction

As outlined in the previous sections, the calculated values of neutron kerma factors need to be improved. The models developed in this work aim at reducing the numerical errors and maximizing the utilization of the available data. These models are able to tolerate some data uncertainties and minimize the effect of simplifying assumptions.

In this algorithm development, direct energy balance and kinematics methods are combined. The better method is used to obtain the final result after analyzing the error associated with each method in a particular incident neutron energy range.

Natural elements, for which isotopic nuclear data are not yet evaluated, deserve and are given special attention. Some of the expressions are not directly applicable in the case of natural elements, and suffer larger uncertainties.

The proposed model will be described in the next sections. The direct energy balance method will be presented first. Following that, the kinematics method will be described. For each nuclear interaction, one or more kinematic expressions for the local energy release will be included. A section on error analysis and sensitivity to the required data will follow each expression as appropriate. Finally, the procedure for evaluating kerma factors from the available data and methods will be described.

The ENDF/B-V file numbers MF and reaction numbers MT referred to in this chapter are described in detail in reference [1].

II.2 Prompt Kerma Factor - Direct Energy Balance

The kerma factor K for a given nuclide at incident neutron energy E can be expressed as:

$$K = \sum_i \sigma_i h_i \quad (\text{II.1})$$

where σ_i is the microscopic cross section for reaction i , and h_i is the corresponding energy of secondary charged particles and residual nuclei. The sum is taken over all possible reactions with the incident neutron. The local energy deposition, or the h -factor h_i , can be obtained by subtracting the energy carried away with neutral particles from the total available energy. Therefore,

$$K = \sum_i \sigma_i (E + Q_i - E_{ni} - E_{\gamma i}) \quad (\text{II.2})$$

where Q_i is the Q -value of reaction i , E_{ni} is the average energy of all secondary neutrons, and $E_{\gamma i}$ is the average gamma energy of reaction i . Equation (II.2) gives the basic algorithm for computing kerma factors by direct energy balance. The detailed gamma production for every gamma-producing reaction is not always given in ENDF/B-V. Instead, gamma production data are usually lumped together for several reactions, e.g., non-elastic lump plus radiative capture gammas. In this case, partial kerma factors for individual reactions are not properly defined and Equation (II.2) has to be rearranged. Thus,

$$K = \sum \sigma_i (E + Q_i - E_{ni}) - \sum \sigma_{\gamma j} E_{\gamma j} \quad (II.3)$$

where $\sigma_{\gamma j}$ is the gamma production cross section for the simple or composite reaction j , and $E_{\gamma j}$ is the corresponding gamma energy. A special treatment for radiative capture kerma factor is needed. This will be elaborated in section II.2.1. With radiative capture singled out, Equation (II.3) becomes:

$$K = \sum_{i, i \neq (n, \gamma)} \sigma_i (E + Q_i - E_{ni}) - \sum_{j, j \neq (n, \gamma)} \sigma_{\gamma j} E_{\gamma j} + K_{(n, \gamma)}. \quad (II.4)$$

II.2.1 Radiative Capture Kerma Factor - $K_{(n, \gamma)}$

The local energy deposited due to an (n, γ) reaction is carried by the recoil nucleus. Following the general direct energy balance approach, the average recoil energy can be obtained from

$$E_r = E + Q - E_\gamma \quad (II.5)$$

where E_γ is the average energy of all emitted photons. The recoil energy is usually less than 1% of the capture photons' energy, resulting in a large relative error in E_r as computed by equation (II.5), even for a small error in E_γ . If only the simpler discrete photon representation is found in the ENDF/B file 12, taking primary photons into account, the total gamma energy should match the available energy and, according to ENDF/B rules, we get

$$E_\gamma = Q + E \frac{A}{A+1} \quad (II.6)$$

where A is the ratio of target nucleus mass to the neutron mass. Using this value with equation (II.5) we get

$$E_r = \frac{E}{A+1} \quad (\text{II.7})$$

which is only the first term in the kinematically obtained relation, namely:

$$E_r = \frac{1}{A+1} E + \frac{1}{2(A+1)m_n c^2} E_\gamma^2 \quad (\text{II.8})$$

where $m_n c^2$ is the neutron rest mass energy.

The reason for this is that calculating photon energies from energy level data of the product nucleus overlooks the slight reduction of photon energies due to the product nucleus recoil energy which shares the excitation energy with the emitted photons. Although this component is in the order of 0.1% of total photon energy and may be justifiably ignored for many other applications, it makes considerable contributions to the recoil energy, and actually dominates the total prompt kerma for small incident neutron energies. The above argument justifies the exclusion of radiative capture from the direct energy balance general equation (II.3) to be singled out for kinematics treatment. There remains one more difficulty. It was observed for some materials in ENDF/B-V that capture photon files do not cover the entire energy range of the cross section. Instead, the capture photon data are given up to a certain cutoff incident neutron energy, while the rest is lumped together with other photon-producing reactions: MF=13, MT=3. Formwise, the above non-redundant gamma representation is suitable for using the direct energy balance method

to compute total neutron kerma factors for all reactions in the entire energy range of ENDF/B. The direct energy balance method however is not acceptable to compute capture kerma factors for low incident neutron energies (below ~ 10 keV), because no information could be recovered about the dominant component of the recoil energy due to the momentum of the emitted photons. If the kinematics method is used to compute capture recoil energy in the entire incident neutron energy range, with the gamma cutoff energy high enough (in the keV region), the error produced by using zero gamma energy in the momentum equation above the cutoff is negligible. The problem arises though when direct energy balance is used with other reactions where the gamma energy containing capture photons is subtracted. The unjustified subtraction of capture photon energy above the cutoff results in underestimated or possibly negative neutron kerma factors. The problem is solved by using the kinematics equation (II.7) below the cutoff, and the direct energy balance method of equation (II.5) above the cutoff provided that the photon lump in MT=3 is subtracted, i.e., the other gamma-producing reactions whose photons are represented in the same ENDF/B section are treated by the direct energy balance method.

II.3 Prompt Kerma Factor - Kinematics Method

In the following sections, the kinetic energy of charged recoils are computed for each class of reactions. It should be noted that some reactions can not be treated by the kinematics method,

namely the reactions producing charged particles where no direct information is provided for their energy. The equation of momentum balance can not be used in this case and the kerma factor is derived from the energy balance relation. On the other hand, the secondary neutron energy from elastic and level inelastic scattering which are needed in the direct energy balance approach can be obtained only by kinematics.

II.3.1 Elastic Scattering, (MT = 2)

The h-factor for elastic scattering is the average energy of the recoil nucleus E_r . Solving energy and linear momentum balance equations, we get

$$h = E_r = \frac{2AE}{(A+1)^2} (1 - \bar{\mu}_{cm}) \quad (II.9)$$

where A is the ratio of target nucleus mass to neutron mass. The average cosine of the secondary neutron scattering angle in the center of mass system $\bar{\mu}_{cm}$ is obtained from the angular distribution data in ENDF/B file 4.

The energy of the secondary neutron can be obtained only by kinematics, and elastic scattering partial kerma is identical for both the kinematics and direct energy balance approaches.

The relative error in the average recoil energy due to the uncertainty in $\bar{\mu}_{cm}$ is:

$$\delta E_r / E_r = - \delta \bar{\mu}_{cm} / (1 - \bar{\mu}_{cm}). \quad (II.10)$$

This error increases with forward scattering where the value of $\bar{\mu}_{cm}$ is close to 1. In this case the contribution of elastic scattering to total kerma is less and the uncertainty in $\bar{\mu}_{cm}$ becomes less important.

II.3.2 Level Inelastic Scattering, (MT = 51-90)

The recoiling target nucleus is left in the λ^{th} excited state with excitation energy E_{λ}^* . The average recoil kinetic energy is obtained from:

$$E_r = \frac{2AE}{(A+1)^2} (1 - \bar{\mu}_{cm} \sqrt{1 - (A+1)E_{\lambda}^* / (AE)}) - E_{\lambda}^* / (A+1). \quad (II.11)$$

Deexcitation by photon emission (LR flag = 0 or 31) results in an h-factor $h = E_r$. If direct energy balance is chosen for a particular material as the general approach, the λ^{th} level scattering contribution to the first term of Equation (II.4) will be $\sigma_{s\lambda} (E_r + E_{\lambda}^*)$, where $\sigma_{s\lambda}$ is the inelastic scattering cross section to the λ^{th} excitation level. The addition of the excitation energy E_{λ}^* will be offset by the subtraction of the equivalent contribution of inelastic gammas in the second term of Equation (II.4). Deexcitation by charged particle emission results in an h-factor of:

$$h = E_r + E_{\lambda}^* + Q \quad (II.12)$$

which is identical for both kinematics and direct energy balance general approaches.

The effect of the uncertainty in the average cosine of the scattering angle, $\bar{\mu}_{cm}$, is less than its effect in the earlier case of elastic scattering. The reason is that in the case of inelastic scattering, $\bar{\mu}_{cm}$ is multiplied by a factor which is less than 1. The effect of the error in the excitation energy E_{λ}^* on the recoil energy can be approximated assuming isotropic scattering and large mass ratio as:

$$\delta E_r / E_r \simeq \delta E_{\lambda}^* / (2E - E_{\lambda}^*). \quad (II.13)$$

A relative error in the recoil energy will be nearly of the same magnitude as the error in the excitation energy near the reaction threshold where the cross section is small. For higher incident neutron energies, the effect of this error is reduced. Usually E_{λ}^* is precisely known for well-defined levels. On the other hand, some closely-spaced levels are lumped together as one level in ENDF/B-V representation, or the continuum is divided into excitation energy bands and presented as pseudo levels.

II.3.3 Inelastic Scattering to Continuum, (MT = 91)

Angular distribution data for the secondary neutron in the laboratory system is provided in ENDF/B file 4. The average recoil energy can be calculated from:

$$E_r = \frac{1}{A} (E + E_{n'} - 2\bar{\mu}_L \sqrt{E} \langle \sqrt{E_{n'}} \rangle) \quad (II.14)$$

where $\bar{\mu}_L$ is the average cosine of the secondary neutron scattering angle in the laboratory system, and $\langle \sqrt{E_n} \rangle$ is the average of the square root of the secondary neutron energy weighted by its energy spectrum given in ENDF/B-V file 5. To derive Equation (II.14) from energy and linear momentum balance, angular and energy distribution data were assumed independent. The ENDF/B-V file 6 coupled angular and energy distribution data was never activated for any material. If $\bar{\mu}_L = 0$ is found in file 4, it is not substituted directly in Equation (II.14), but the more reasonable assumption of center-of-mass isotropy is made. The average recoil energy is thus obtained from:

$$E_r = \frac{(A-1)E}{A(A+1)} + \frac{E_n}{A} . \quad (\text{II.15})$$

In the case the deexcitation flag LR indicates charged particle emission, kinematics equations (II.14) and (II.15) cannot be used and only the direct energy balance method is applicable for this reaction.

The average excitation energy is always larger than the last discrete level excitation, and the secondary neutron energy is expected to be much smaller than the incident neutron energy. The recoil energy is therefore dominated by the incident neutron kick and is less sensitive to the uncertainty in the secondary neutron energy.

II.3.4 Radiative Capture, (MT=102)

The kinematic relation presented by Equation (II.8) is used to compute radiative capture kerma factors when capture gamma files are

given in ENDF/B-V data. If gamma production cross section and spectra are not provided in ENDF/B-V, an alternative kinematics equation can be used. Solving the momentum and energy balance equations, the average recoil energy can be obtained as:

$$E_r = E + Q + \eta(A+1)m_n c^2 \left(1 - \sqrt{1 + \frac{2}{\eta(A+1)m_n c^2} \left(Q + \frac{AE}{A+1} \right)} \right) \quad (\text{II.16})$$

where $m_n c^2$ is neutron rest mass energy. The η -factor contains the photon spectral information and represents their effect on the recoil nucleus energy. For isotropic uncorrelated photons, the η -factor is obtained from:

$$\eta = E_\gamma^2 / \overline{E_\gamma^2} \quad (\text{II.17})$$

For a single isotropic photon, we get $\eta=1$. For an estimate of the η -factor, other data sources for capture gamma spectra are needed. The capture kerma factor is sensitive to the η -factor at low incident neutron energies where $\delta E_r/E_r \simeq -\delta\eta/\eta$.

II.3.5 Sequential (n,2n) Reaction, (MT = 6-9, 46-49)

In the time-sequential (n,2n) reaction $A(n,n1)A^*(n2)(A-1)^*$, the first neutron $n1$ is essentially coming from an inelastic scattering event that may leave the nucleus A^* in one of several definite excited states. The subsequent decay of the intermediate nucleus A^* by the emission of the second neutron $n2$ will possibly

leave the final product nucleus $(A-1)^*$ in an excited state. The average energy of the recoiling intermediate nucleus is given by equation (II.11). The additional component of recoil energy due to the emission of the second neutron will be considered next. The reaction mechanism suggests that the second neutron is isotropically emitted in the center-of-mass moving with the intermediate nucleus A^* . The angular distribution data of the second neutron in the laboratory system given in ENDF/B file 4 can be interpreted in a qualitative way. If the second neutron is not isotropic in the laboratory system, it will be understood that the anisotropy is due to the momentum component of the moving excited intermediate nucleus, and therefore, the net recoil energy is computed from:

$$\begin{aligned} E_r = & \frac{2A(A-2)E}{(A-1)(A+1)^2} (1 - \bar{\mu}_{cm,1} \sqrt{1 - (A+1)E_\lambda^* / (AE)}) \\ & - (A-2)E_\lambda^* / (A^2-1) + E_{n2} / (A-1) \end{aligned} \quad (II.18)$$

where $\mu_{cm,1}$ is the cosine of the first neutron emission angle.

On the other hand, an isotropic second neutron is understood to be emitted after a time delay long enough to allow slowing down of the intermediate nucleus A^* by the surrounding medium, therefore,

$$\begin{aligned} h = & \frac{2AE}{(A+1)^2} (1 - \bar{\mu}_{cm,1} \sqrt{1 - (A+1)E_\lambda^* / (AE)}) \\ & - E_\lambda^* / (A+1) + E_{n2} / (A-1) . \end{aligned} \quad (II.19)$$

In the incident neutron energy range for which the excitation of the residual nucleus is not energetically possible, or not observed; and if only a few data points are provided for the second neutron energy description, we can use the binding energy of the last neutron in the target nucleus, B , to replace the secondary neutron energy data and write:

$$h = \frac{2AE}{(A+1)^2} (1 - \bar{\mu}_{cm,1} \sqrt{1 - (A+1)E_{\lambda}^* / (AE)}) - \frac{B}{A} + E_{\lambda}^* / (A(A+1)) . \quad (\text{II.20})$$

II.3.6 Direct (n,2n) Reaction, (MT=16)

The two neutrons are indistinguishable in the ENDF/B data. The reaction proceeds without going through an intermediate state. The kinetic energy of the recoiling residual nucleus following an (n,2n) reaction is obtained by solving momentum and energy balance as:

$$E_r = \frac{1}{A-1} (E + E_{n1} + E_{n2} - 2\mu_1 \sqrt{EE_{n1}} - 2\mu_2 \sqrt{EE_{n2}} + \mu_1 \mu_2 \sqrt{E_{n1}E_{n2}} + 2\sqrt{E_{n1}E_{n2}} \sqrt{(1-\mu_1^2)(1-\mu_2^2)} \cos(\phi_1 - \phi_2)) \quad (\text{II.21})$$

where E_{n1} , and E_{n2} are the two secondary neutron energies, μ_1 , and μ_2 are the scattering angle cosines in the laboratory system, and ϕ_1 and ϕ_2 are the azimuthal angles. Detailed information about secondary neutron spectra and angular distributions required to apply equation

(II.21) are not all available in the ENDF/B-V data and, therefore, simplifying assumptions are necessary. Different levels of approximations can be applied depending on the availability and the user's confidence in the accuracy of different required data. For example, it was traditionally assumed that the two secondary neutrons are emitted isotropically in the center-of-mass (cm) system. In this algorithmic development, it is possible to use the secondary neutron angular distribution data given in ENDF/B-V file 4. Consider the following arguments to reduce equation (II.21):

- * No azimuthal information is available. The azimuthal angles of the two neutrons are assumed independent and, therefore, the average $\langle \cos(\phi_1 - \phi_2) \rangle = 0$. The effect of a non-zero $\langle \cos(\phi_1 - \phi_2) \rangle$ is less significant for highly anisotropic neutrons.

- * Individual secondary neutron angular distributions are not given. We, therefore, assume that the distribution of the given average neutron is the same for each individual neutron with no correlation. Hence, μ_1 and μ_2 are replaced by the combined μ_L and, therefore,

$$\langle \mu_1 \mu_2 \rangle = \mu_L^2. \quad (\text{II.22})$$

- * Individual secondary neutron energy distributions are not given. Only the distribution of an average neutron is usually given in file 5. The energy of the average neutron is

$$E_{n'} = (E_{n1} + E_{n2}) / 2 \quad (\text{II.23})$$

The assumption that $\sqrt{E_{n1}} + \sqrt{E_{n2}} = 2\sqrt{E_{n'}}$ is accurate to 0.1%,

provided that the deviation of either E_{n1} or E_{n2} from the average does not exceed 10%.

* The average quantity $\langle \sqrt{E_{n1} E_{n2}} \rangle$ was computed assuming the triangular distribution shown in Fig. (II.1) for the average neutron, and the correlation $E_{n1} + E_{n2} = 2 E_n$, resulting in:

$$\langle \sqrt{E_{n1} E_{n2}} \rangle = 0.905 E_n . \quad (\text{II.24})$$

The variation of this result with the postulated distribution is not significant and the application of more elaborate secondary neutron distributions [49,50] is not necessary. Using the above argument, the recoil energy can be computed from:

$$E_r = \frac{1}{A-1} (E + 2E_n (1 + 0.905 \bar{\mu}_L^2) - 4 \bar{\mu}_L \sqrt{E E_n}) . \quad (\text{II.25})$$

For small μ_L or isotropic neutron emission, the algorithm developed next for (n,mn) reactions can be used.

II.3.7 Isotropic (n,mn) Reactions

With $m = 2, 3$, or 4 , the energy of the m neutrons can be found in ENDF/B file 5 without distinguishing individual neutrons. With the center-of-mass isotropy assumption, the average recoil energy can be obtained from:

$$E_r = \frac{A-m}{A(A+1)} E + \frac{E_{mn}}{A} \quad (\text{II.26})$$

II.1

Triangular approximation of the secondary neutrons energy distribution.

where E_{mn} is the average of the kinetic energy of all the m secondary neutrons. For an $(n,2n)$ reaction, the error resulting from the indistinguishability between the energies of the two neutrons in the above expression is $\approx E_{2n}/(A(A-1))$ which is small for large mass ratios. For the $(n,3n)$ and $(n,4n)$ reactions, assuming that the mass ratio is much larger than 1, the indistinguishability error can be approximated to $(m-1) E_{mn} / A^2$.

In the incident neutron range where the energy available for the reaction (n,mn) is not sufficient (or not observed) to excite the residual nucleus, the method described below can be used to compute the average recoil energy.

Assuming that the m neutrons are emitted isotropically in the center-of-mass, the momentum and energy conservation lead to:

$$E_r = \frac{2A+1-m}{(A+1)^2} E + \frac{Q}{A+1} \quad . \quad (II.27)$$

In this alternative method, the secondary neutron data in the ENDF/B file 5 are not required and are replaced by the more accurately-known Q -value. The recoil energy produced by this method lies always between the values produced by the direct energy balance method and that of the kinematics method. Figure (II.2) shows the dependence of the average recoil energy on the average secondary neutron energy obtained from the data files at a specific incident

Figure II.2
The relation between different methods for calculating the average recoil energy for (n,mn) reactions.

neutron energy. The intersection between the two lines representing direct balance and kinematics produces the recoil energy of equation (II.27) and the corrected value for the average secondary neutron energy. The latter can be compared with the value obtained from the data files and provide a useful feedback to nuclear data evaluators. The use of equation (II.27) is recommended when the description of secondary neutrons energy is given only roughly by a few points in the ENDF/B-V file 5. The user has to supply the upper limit of the incident neutron energy above which residual excitation is possible.

II.3.8 Level (n,x) Reactions

A single charged particle x is emitted leaving the product nucleus in a definite excitation level. The particle x may represent p , d , t , He-3 , or alpha particle. Data for this class of reactions may be found in ENDF/B-V file 3, MT = 700 Series. The partition of energy between the recoiling product nucleus and the charged particle is not necessary for the purpose of computing kerma factors. A kinematic relation for the local energy deposition h using momentum and energy balance can not be written for this reaction. Such kinematic relation will be equivalent to the direct energy balance equation with the residual excitation energy E^* replacing the secondary gamma energy, i.e.,

$$h = E + Q_0 - E^* \quad (\text{II.28})$$

where Q_0 is the Q-value to the ground state. Equation (II.28) however, does not utilize ENDF/B-V gamma files and fits the general framework of the kinematic option.

II.3.9 Continuum (n,x) Reactions

The kinematic treatment of (n,x) reactions is limited to the materials for which direct information about the secondary charged particle is given. The only encountered cases in ENDF/B-V where such data are available are that of Si(n,p) and Si(n, α) reactions. The proton and alpha particle energies are given in file 5, MT = 718 and 798 respectively. Assuming that the charged particle is isotropically emitted in the center-of-mass, we obtain:

$$h = \frac{A+1-2A_x}{(A+1)(A+1-A_x)} E + \frac{A+1}{A+1-A_x} \bar{E}_x \quad (\text{II.29})$$

where A_x and E_x are the charged particle's mass ratio and average energy respectively.

II.3.10 Continuum (n,mn';ix) Reactions

The kinematic treatment of this class of reactions is possible only if direct information about the secondary charged particle energy is available. The only encountered cases in ENDF/B-V where such information is given are that of Si(n,n'p) and Si(n,n' α) reactions. The proton and alpha particle energies were given in file 5, MT = 719 and 799 respectively. The ratio between (n,n'x) and (n,xn') is not

given, and the secondary neutron is assumed to be emitted first. It is assumed that the secondary neutron is emitted isotropically in the center-of-mass, and the charged particle is isotropically emitted in the coordinate system moving with its precursor nucleus. The h-factor can be obtained from:

$$h = \frac{(A-1)(A-2A_x)E}{A(A+1)(A-A_x)} + \frac{(A-2A_x)E_n}{A(A-A_x)} + \frac{AE_x}{A-A_x} \quad (\text{II.30})$$

where A_x and E_x are the charged particle's mass ratio and average energy respectively.

II.4 Heat Release from Fission

For calculations of heating in fission and hybrid fusion reactors, it is important to know the energy release per fission and its various components: fragment kinetic energies and prompt and delayed radiation. The total energy per fission can be determined either by mass balance after fission product decay or by adding the individual components.

Fission product yields and the subsequent decay data are given in ENDF/B-V, file 8. Reaction numbers MT=454 and 459 specify the incident energy-dependent fission product yield data. 40 sets of direct and cumulative yields(44000 yields) are given [51].

The previous version of ENDF/B, namely ENDF/B-IV contains only 10 sets of direct yields. However, LaBauve et al. [52] reported that

ENDF/B-IV data compare better with experiment. Similar criticism was reported in reference [53].

Using the energy-dependent yields, the energy-dependent Q-value is calculated from:

$$Q(E) = c^2 [M - \sum_j Y_j(E) M_j - (\bar{\nu}(E) - 1) m_n] \quad (\text{II.31})$$

where $Y_j(E)$ is the yield and M_j is the mass of the fission product j , M is the fissionable nucleus mass, and $\bar{\nu}(E)$ is the average number of fission neutrons. The sensitivity of the Q-value to adjustments in the yield sets was investigated by R. Sher et al. [54]. It was found that, although some yields were changed under the adjustment by substantial amounts, the Q-value calculated from any of the yield sets varied by less than 0.1%.

The local energy deposition can be obtained from the energy balance as:

$$h = E + Q(E) - E_n - E_\gamma - E_\nu \quad (\text{II.32})$$

where E_ν is the unrecoverable energy carried away by neutrinos and antineutrinos.

The values of fission energy release and its different components calculated by Sher et al. [54] are adopted for the ENDF/B-V.2 (second revision) and are used for calculating fission kerma factors in this work. More recent measurements [55] of the kinetic

energy of fission fragments are in good agreement (to 1 MeV) with those presented by Sher. The variation of different energy release components suggested by Walker [54] takes the form:

$$E_i(0) = E_i(E_{inc}) + \delta E_i \quad (II.33)$$

where E_i is any of the energy release components, $E_i(0)$ is its value at $E_{inc}=0$, and $E_i(E_{inc})$ is the value at an incident neutron energy E_{inc} . It should be recognized that the values at $E_{inc}=0$ are strictly fictitious and represent an artifice by which it is possible to recover the values at any E_{inc} .

The prompt fission kerma factor K_f becomes:

$$K_f(E_{inc}) = \sigma_f(E_{inc}) * EFR(E_{inc}) \quad (II.34)$$

where EFR is the kinetic energy of the fission fragments including betas. In the case where no gamma files are provided in the ENDF/B-V evaluation, local deposition of prompt gamma energy is assumed. Thus,

$$K_f(E_{inc}) = \sigma_f(E_{inc}) * (EFR(E_{inc}) + EGP(E_{inc})) \quad (II.35)$$

where EGP is the prompt gamma energy per fission. It should be noted that both $EFR(E_{inc})$ and $EGP(E_{inc})$ are taken to be constants independent of the incident neutron energy. This indicates that the

increase in the incident neutron energy does not result in increasing fission energy release, but is rather used up in producing more fission neutrons [56].

Charged particle and gamma decay heat K_{fd}^{ch} and K_{fd}^{γ} are obtained from:

$$K_{fd}^{ch} = \sigma_f(E_{inc}) * EB(E_{inc}) \quad (II.36)$$

and

$$K_{fd}^{\gamma} = \sigma_f(E_{inc}) * EGD(E_{inc}) \quad (II.37)$$

where EB and EGD are the the energy released by delayed beta and gamma rays, respectively. The change of these quantities with the incident neutron energy is obtained from:

$$EB(E_{inc}) = EB(0) - 0.075 E_{inc} \quad (II.38)$$

and,

$$EGD(E_{inc}) = EGD(0) - 0.075 E_{inc} . \quad (II.39)$$

II.5 Natural Elements

The description of the model so far applies only to materials with a single isotope. In the case of natural elements with more than one isotope, a few modifications need to be introduced to the model and additional data are needed to supplement the ENDF/B files. The need for energy-dependent effective Q-values is of particular importance. The effective Q-value for a specific reaction is defined as:

$$Q_{\text{eff}}(E) = \sum_j W_j(E) Q_j \quad (\text{II.40})$$

where the sum is taken over all the constituent isotopes. Q_j is the Q -value of the j^{th} isotope for the specific reaction, and the weighting function $W_j(E)$ is the ratio of the j^{th} isotope macroscopic cross section to the elemental macroscopic cross section for that particular reaction. The supplemental isotopic cross sections needed for constructing the weighting function are searched for in the literature and other data files. The same treatment applies also to computing the energy-dependent decay energy term for a natural element.

Other less significant improvements can be also introduced. For instance, the abundance-weighted mass ratio A is used with kinematics equations. An error $\delta A/A$ of magnitude usually less than 10% is associated with the particular partial kerma factor, where δA is the spread of mass ratios of the constituent isotopes. The replacement of the abundance-weighted mass ratio given in ENDF/B-V by cross section weighting like that of equation (II.40) is formally incorrect because the kerma factor dependence on mass ratio is not linear. The situation, however, can be improved with some reactions for which only one isotope can be identified to undergo the reaction. Non-pseudo levels in the ENDF/B-V reactions MT=51-90 series can be used along with nuclear structure data to identify a single isotope whose mass ratio will be substituted in the kinematics equation (II.11).

II.6 Energy Deposition due to Radioactive Decay

The contribution of the radioactive decay of the activated residual nuclei to the total kerma factor was considered in this work. Decay energy is obtained separately for charged particle decay and gamma decay. Charged particle decay heat adds to the local energy deposition, while decay gamma rays with a longer mean free path will be transported and deposit their energy elsewhere away from the reaction site or may leak from the system. Only charged particle decay heat was considered in previous works. An attempt was made to separate decay gamma energy production for a few materials in the Vitamin-E [57] group structure. These group gamma production matrices could be used with transport codes to calculate the distribution of gamma decay heat. The magnitude of decay gamma heat relative to the prompt gamma heat is usually small, and the available prompt gamma production data still suffer large uncertainties. The idea of transporting decay gamma rays was therefore abandoned for the present. A separate response function for the sum of prompt kerma factor, charged particle decay heat production, and gamma decay heat production is also computed.

Charged particle decay kerma factor K_d^{ch} , and gamma decay kerma factor K_d^γ are obtained from:

$$K_d^{ch}(E) = \sum_i \sigma_i(E) * E_d^{ch} \quad (II.41)$$

and,

$$K_d^\gamma(E) = \sum_i \sigma_i(E) * E_d^\gamma \quad (II.42)$$

where $\sigma_i(E)$ is the microscopic cross section of the i^{th} reaction leading to the formation of a radioactive residual. E_d^{ch} , and E_d^γ are the resulting charged particle and gamma energies respectively. For natural elements, isotopic cross sections are taken from ENDF/B-V dosimetry files or other data files or evaluated using the methods developed for computing the energy dependence of elemental Q-values.

Decay energy is considered only if the half-life of the parent nucleus is less or equal to one day. In case the radioactive nucleus does not reach stability after one decay event, subsequent decay is considered only if the effective half-life T_{eff} meets this cutoff criterion of one day. The effective half-life is taken to be:

$$T_{eff} = \sum_i T_i \quad (II.43)$$

where T_i is the half-life of the i^{th} decay event.

For beta decay, the maximum energy is obtained either from The Table of Isotopes [47,48] or calculated from the mass difference. The average is then computed by multiplying by the average-to-maximum ratio computed by Abdou using Fermi theory and tabulated in reference [7]. For β^+ decay, two 0.511 MeV annihilation gammas are added to the decay gamma contribution. Electron capture X-ray energy is small and therefore its contribution was neglected.

The contribution of fission delayed beta and gamma rays are taken from ENDF/B-V.2 according to Sher [54]. No half life cutoff was applied in this case.

The choice of the half-life cutoff is arbitrary. The decay energy saturates rapidly with increasing half-life, and the choice of one day cutoff can be supported by studies of fusion blankets afterheat [58,59]. For applications where the time dependence of decay heat is important, other radioactivity analysis codes like RACC [60] can be used.

The charged particle decay energy and gamma decay energy are tabulated in Appendix (A) for the contributing reactions for all the materials considered in this study.

II.7 Kerma Factor Evaluation Procedure

The code KAOS-V (Kerma And Other Stuff) was written to be used as the main kerma factor evaluation tool. A brief description of the code and a user's guide are given in chapter III. The evaluation of the recommended neutron kerma factors for a material depends largely on the individual characteristics of that material and the particularities of its evaluated nuclear data. The kerma factor evaluation procedure for each of the 58 materials covered in this work is briefly described in reference [32]. There is however a general procedure that applies to most of these evaluations. This general procedure is described as follows:

* The ENDF/B-V data for the material under study is reviewed. The nuclear structure, excitation levels, and decay modes for the target and product nuclei are reviewed using the Table of Isotopes. The reaction Q-values are also checked. This review helps primarily in planning the next steps.

* For materials where resonance parameters are given in ENDF/B-V file 2, an NJOY run [24,25] was made to reconstruct the cross sections and perform Doppler broadening at 300°K. The NJOY module HEATR is also run [25] to make comparisons with subsequent KAOS-V runs. The NJOY PENDF file is kept for these purposes.

* A KAOS-V run is made in the Direct Energy Balance mode. The partial and total kerma factors and gamma energy production cross section are printed and plotted. The intermediate files in which these results are stored are saved for subsequent comparison tests. The total kerma factors produced in this stage are in good agreement with the NJOY (HEATR) results for most materials.

* A KAOS-V run is made in the kinematics mode. Again, the partial and total kerma factors are printed, plotted, and saved for the subsequent comparisons.

* The total kerma factors produced by Direct Energy Balance and Kinematics are compared. Good agreement is expected for materials for which the ENDF/B-V files include detailed gamma production for each reaction. On the other hand, if substantial disagreement was found, further testing and comparison is made in the partial kerma factor

level. The interactive mode of the KAOS-V code is frequently used in this stage which facilitates detailed analysis of partial kerma factors and provides the means for performing non-standard tests.

- * Partial kerma factors for each reaction produced by the Direct Energy Balance and Kinematics methods are compared. The gamma energy production is also compared with the excitation energy of the residual nuclei, a common test to identify energy balance problems.

- * For natural elements, the energy-dependent Q-values are estimated using additional data sources as described earlier. The Direct Energy Balance and Kinematics runs are repeated using the variable Q-values. The sensitivity of the total kerma factors to the introduction of the variable Q-values is studied. Depending on this sensitivity, more effort was made in estimating the more important Q-values, and the use of variable Q-values was dropped altogether for some other reactions.

- * After making the necessary data corrections (if any), the information collected through the above testing and comparisons is used to make the final decision on the recommended set of prompt kerma factors. In general, at low incident neutron energies, kerma factors are dominated by radiative capture and elastic scattering. The methods for calculating kerma from these two processes are identical for both Direct Energy Balance and Kinematics approaches. At higher incident neutron energies, the inelastic scattering is turned on. If the ENDF/B-V file 12 is used to represent inelastic gammas in a manner consistent with file 3 cross sections, the agreement between the two

methods will be extended to cover the range of incident neutron energies below the thresholds of other nonelastic reactions. In this incident energy range, the total kerma factors using Direct Energy Balance for many materials experiences a minimum and becomes more sensitive to gamma energy imbalance or representation problems and is often negative, and the Kinematics method is therefore used. For incident neutron energies high enough to produce excited residuals following the emission of charged particles, Direct Energy Balance is generally preferred. The reasons for this preference are the fact that the Kinematics method is not equipped to account for residual excitation and generally overestimates kerma in this range, and also because the magnitude of kerma factors is higher and the sensitivity to gamma errors is therefore reduced. To preserve the continuity of the final recommended set of kerma factors, the connection between the results from the two methods is made at an incident neutron energy at which the two results are in agreement.

* The decay of the unstable products is then studied. The decay heat is included to produce additional kerma response functions as detailed in later sections.

CHAPTER III

KAOS-V CODE DESCRIPTION AND USER'S GUIDE

III.1 Introduction

The code KAOS-V (Kerma And Other Stuff) is designed to compute kerma factors using a user-controlled combination of the available methods described earlier. This code has the ability to compare the different results which aids in the choice of a consistent set of algorithms to evaluate kerma factors. This "evaluation" requires testing data consistency and the sensitivity of the results to using alternative methods which require in general a different set of data and assumptions. The code can process data in ENDF/B-IV and ENDF/B-V format. For resonance treatment, the code has the ability to access NJOY and NPTXS interface files in formatted or binary forms. No input instructions are necessary to run the code interactively. The user can simply respond to the interactive messages sent by the code if an INPUT file is absent. This feature was chosen to suit the expanding nature of this type of analysis with experience with more materials. It is also important for the user to monitor the run through the diagnostic messages and suggestions dictated by the particulars of the material's nuclear data. However, an INPUT file is automatically generated following an interactive run, and can be edited and used to rerun or produce different results. Grouped and point output data can be produced along with graphic representation.

The KAOS-V code is written in standard FORTRAN-77. It runs on the CRAY machines of the National Magnetic Fusion Energy Computer Center at Lawrence Livermore National Laboratory. In the following sections, a brief user-oriented description of KAOS-V code will be presented.

III.2 The Connected Files

The following is a description of input, output and intermediate files used or produced by the code.

III.2.1 INPUT file

The file INPUT should be prepared according to the instructions given below. If this file is not provided, the code will automatically enter the interactive mode as mentioned above. Concurrently, the code prepares an INPUT file in which the data provided interactively by the user are stored. The user may use the created INPUT file for a rerun or edit it to produce new results. To understand the created input file format, the user may refer to the Interactive Input Guide.

Standard INPUT instructions

The following is a prescription for preparing an INPUT file for fast standard runs. Decisions are internally taken, and diagnostic messages will be written in the file DIAGNOSE. All input data are in free format.

Card (1):

ISTAND Positive value to specify direct energy balance method.
Negative value to use kinematics whenever possible. The
zero is reserved for automatically-prepared INPUT. The
absolute value indicates the output intensity as follows:

- 1 Print and plot grouped total prompt kerma factors, total
kerma factors, and response functions.
- 2 Also print and plot grouped partial and decay kerma
factors.
- 3 Also print point total prompt and total kerma factors.
- 4 Also print point partial kerma factors.

Card (2)

MAT ENDF/B material number. Positive value for version V, and
negative for version IV.

Card (3)

IPK 1 Compute prompt kerma factors for all reactions.
0 No effect.

Card (4)

IDECAF 1 Radioactive decay is considered, and cards 4.1 through 4.3
are needed.
0 No effect.

Card (4.1)

KIND 1 Charged particle decay only.
2 Gamma decay only.

- 3 Both charged particles and gamma entered separately. In this case, input cards 4.2 are required twice.
- 4 Both combined.

Card (4.2)

MT ENDF/B reaction number resulting in radioactive product.

EDECAY Corresponding decay energy in electron volts.

Card (4.2) is repeated for all participating reactions.

The set is terminated by setting MT=0.

Card (5)

IRES 1 Compute response functions as specified in the file RESPONSE. No action if the file is missing.

0 No effect.

Card (6)

IPRO 1 Produce grouped values of the computed kerma factors and other responses.

0 No effect.

III.2.2 ENDF file

An ENDF file containing the ENDF/B nuclear data for the particular material under study should be present. The file might include data for other materials as well. The ENDF file should be in text form.

III.2.3 PENDF file

A PENDF file is needed for materials where ENDF/B file 2 resonance parameters are given. The PENDF file is produced by running the NJOY module RECONR and used to supply reconstructed point cross sections for total, elastic, radiative capture, and fission reactions. If Doppler-broadened cross sections are needed, the PENDF file used to supply these data to KAOS-V code. It is produced by running NJOY modules RECONR and BROADR. A warning message will be sent by KAOS-V code if resonance parameters are given and a PENDF file is not provided. The code will continue in case this is intentionally left out for a particular task. The PENDF file should be in text form.

III.2.4 FT31 file

The FT31 file is produced by NPTXS [**], the AMPX module to produce point data for resonance nuclides. The FT31 file is equivalent to PENDF file, and either one is sufficient. Notice that FT31 is a binary file.

III.2.5 RESPONSE file

The file RESPONSE is needed to compute any non-kerma responses. If response computation is requested by setting IRES=1 the file RESPONSE will be opened to get the detailed response request. If the file is absent, the request will be skipped and diagnostic warning is sent.

The following set of cards is required for each response. The specified cross sections multiplied by a constant are summed up to construct the response function. Reactions will be ignored if not found on ENDF/B tape, and the whole response will be skipped if all its reactions are missing.

Card (1) Format (A72)

TITLE Response function title.

Card (2)

MT Reaction number. Interpreted as an LR flag if the reaction does not have an explicit MT number.

XMULT Reaction multiplier.

Card (2) is repeated for all reactions participating in making the response. The set is terminated by setting MT=0 (and any XMULT).

III.2.6 NATURAL file

The file NATURAL is needed to enter data superseding those of ENDF/B. The code will send a warning message if this file was not provided while running a natural element problem in case variable Q-values and other data are needed. The format of this file is as follows: Pairs of MT numbers and a modifying variable should be written each pair on a line, and terminated by setting MT=0. The meaning of the modifying variable depends on the value of MT. For MT=50-90, the modifying variable will be taken to replace the mass

ratio of the material to represent that of a specific isotope. For other MT reaction numbers, a positive entry of MT means that the modifying variable will be interpreted as a new Q-value. For a negative entry of MT, the modifying variable should indicate the number n of the file TAPE_n used to supply the variable Q-value data for that reaction. The code will send a diagnostic warning if the specified file TAPE(n) was not found and will proceed using the original ENDF/B Q-value.

III.2.7 GROUP file

If grouped kerma factors and other responses are requested, the file GROUP must be provided. The file contains group structure and weighting spectrum data. If the tabulated weighting function is not provided, the code will assume a flat weighting spectrum and send a diagnostic message. The GROUP file used in this work contains the Vitamin-E [57] 174 neutron group structure and weighting spectrum.

III.2.8 TAPE(n) files

Text files TAPE1, TAPE2,...up to 50 files are created by the code to store relevant kerma factors and response point tables. These files are used to produce grouped results. They can also be used to construct new responses. The TAPES produced by an earlier partially successful run can be used for restart runs to save computer time. Most TAPE(n) files are internally used the user does not normally need

to look at their contents. However, the user needs to enter data, e.g. energy-dependent Q-value tables for natural elements as TAPE(n) files. The TAPE(n) files are organized in the following format:

Card (1): General title, maximum 72 characters.

Card (2): Name of the first variable X in the data table, e.g.

'INCIDENT NEUTRON ENERGY eV.'

Card (3): Name of the second variable Y, again maximum 72 characters.

Card (4): A string of 20 integers representing interpolation data as described in ENDF/B-V format.

Card (5): Number of data points.

Card (6): There is one card per data point X_i, Y_i . These are arranged in ascending order, i.e. smaller values of X_i are listed first.

III.2.9 DIAGNOSE file

In the interactive mode, diagnostic messages are sent to the terminal. In the standard mode, the code creates the file DIAGNOSE to store these messages. The information sent to this file is a detailed follow up of the run, warnings, decisions taken by the code, definition of intermediate files, contents, cpu time information, etc.

III.2.10 OUTPUT file

The requested point and grouped results and comparisons are written in the OUTPUT file created by the code. Every set of output

results is furnished with the proper identifying title. The grouped kerma factors and other responses obtained from the OUTPUT file are used to construct the KAOS/LIB-V library.

III.3 INTERACTIVE INPUT GUIDE

The following is a description of how to respond to the code's messages in the interactive mode:

Section 1

Message: Enter ENDF/B material number.

User: A positive number means version V, negative number for the version IV data file.

Message: Compute prompt kerma factors? 0 or 1.

User: Input 1 to enter prompt kerma factors computation mode, enter 0 otherwise. If 1 is entered, refer to section 2.

Message: Radioactive decay kerma factors? 0 or 1.

User: Input 1 to enter decay kerma factor mode, refer to section 3 for detailed instructions.

Message: Compute more kerma factors? 0 or 1.

User: Input 1 to compute kerma factors using other algorithms with the additional help of the previously-prepared tapes. Refer to section 4 for details.

Message: Response functions? 0 or 1.

User: Input 1 to compute response functions. Refer to section 5 for detailed instructions.

Message: Process point tapes? 0 or 1.

User: To enter the process mode where the kerma factors and response TAPES can be plotted, compared, and/or grouped, the user should enter 1. Refer to section 6 for details.

Section 2 : Prompt Kerma Factor Mode

Message: Input reaction number:

User: Enter ENDF/B reaction number for which prompt kerma factors are requested. A negative number for computing gamma production from the reaction (-1 to compute total gamma production). Or enter 0 to terminate the prompt kerma factor computation mode. In this case, the code will send a diagnostic message and request permission to print total prompt kerma factors. The user's response is a 0 or 1.

The subsequent input varies with the different reaction numbers. For detailed instructions refer to subsections 2.0 through 2.9.

Message: print (a)/store (a)/add (a) to (z)? 3: 0 or 1.

User: Enter three numbers, the ones being the affirmative to do the following:

- * Print array (a) containing the partial prompt kerma factors or gamma production in the OUTPUT file.
- * Store array (a) in a text file TAPE(n) for further processing and editing. The user will get a diagnostic message relating to the tape's name and title of contents.
- * Add array (a) to the total prompt kerma factors stored in

array (z). Enter -1 to subtract (a) from (z) which is usually required to subtract gammas or recover from adding the wrong partial kerma factor.

Section 2.0

For a wrong reaction number, the diagnostic message: (reaction number** is absent, try again) and the first message of section 2 prompts.

Section 2.1

For MT=-1, requesting the calculation of total gamma production. Some details are required about the inclusion of capture and fission gammas to assure consistency with the methods used to compute kerma factors for these two reactions.

Message: ALLGAM: Include capture, fission? 2: 0 or 1.

User: Two numbers are required. Enter 1 for the first number to include gammas from radiative capture MT=102 or the part of it which is represented separately in ENDF/B gamma files, enter 0 otherwise. The second number is similar for fission reaction.

Section 2.2

For MT=2 (elastic scattering) or mt < -1 (requesting gamma production from a specific reaction), no further input is needed. Back to the first message in section 2.

Section 2.3

For mt=51, inelastic level reactions MT=51-90 are processed.

Message: Subtracting MT=3 or 4 photon lump contaminated with level inelastic contribution? 0 or 1.

User: Input 0 to compute inelastic level kerma factors, the next message will not appear. Input 1 to deposit the excitation energy locally, since that will be corrected if you intend to subtract gamma lumps containing the inelastic contribution. In this case, for every level that decays by gamma emission, the following message prompts:

Message: Input energy (eV) above which contamination exists for mt=***

User: Input the incident neutron energy above which the inelastic gamma (excitation energy) of the specific level will be added to the recoil energy.

Section 2.4

For MT=700, 720, 740, 760, 780 level charged particle reactions, notice that the user specifies only the first reaction (ground level) and the rest will automatically follow.

Message: MT=**, LR=** gammas subtracted later? 0 or 1.

User: This message appears only for LR=0 or 31. Enter 0 to subtract the excitation energy now. Enter 1 to deposit that energy locally in the energy range specified in the next message, with the understanding that it can be subtracted later on.

Message: Input 2 energy values [*,*] of gamma file.

User: Enter two incident neutron energies (eV) enclosing the range mentioned above.

Message: MT=**, LR=40, subtract 2*.51 MeV? 0 or 1.

User: This message prompts only if the pair production decay flag LR=40 is detected. The energy of the two 0.511 MeV photons will be subtracted from the local energy deposition if 1 is entered

Section 2.5

For MT=102 (radiative capture)

Message: Ignore gamma files? 0 or 1, input eta:

User: Enter two numbers, the first; one for affirmative to use only the Q-value rather than gamma files. In this case, no other messages will prompt. The second number is used only if the first number is 1. The η -factor is defined earlier in the model description. Typical values of η range between 2 and 3, while $\eta=1$ is equivalent to assuming only one isotropic photon.

Message: Input energy (eV) above which capture photons are included in the MT=3 lump. Automatic detection if negative value is entered.

User: Enter the incident neutron energy above which all the available energy including gammas is deposited locally. Input any negative number to activate automatic detection of truncated capture gamma file.

Section 2.6

For MT=91, 16, 17, and 37

Message: Direct energy balance? 0 or 1.

User: Enter 1 to use direct energy balance method, and refer to section 2.9. Enter 0 to use kinematic methods, and refer to the appropriate section 2.6.1 through 2.6.3.

Section 2.6.1

For MT=91 kinematic method.

Message: mt=91, LR=**: force cm isotropy? 0 or 1.

User: Enter 1 to use kinematics with the secondary neutron assumed to be isotropically emitted in the center-of-mass system. Enter 0 to consider secondary neutron angular distribution data.

Section 2.6.2

For MT=16 (n,2n) reaction.

Message: MT=16 kinematics: Input incident neutron energy (eV) above which residual excitation is possible, and below which the Q-value is used instead of the secondary neutron data.

User: Enter the requested energy having in mind that the Q-method will impose center-of-mass isotropy in its range.

Message: Force cm isotropy? 0 or 1.

User: Enter 1 to impose center-of-mass isotropy on the secondary neutrons distribution in the entire reaction range, and ignore the rest of messages in this subsection. Enter 0 to consider the angular distribution data.

Message: Input the absolute value of the average lab scattering cosine below which cm isotropy is assumed, or the negative of the absolute value of the average cm scattering cosine below which cm isotropy is used.

User: A recommended value is 0.05. If negative value is used, an approximate transformation to the center-of-mass system is activated and the input of -0.1 is a possible alternative.

Section 2.6.3

For MT=17, 37 kinematic.

Message: MT=17 [37] (n,3[4]n) reaction kinematics. Input incident energy (eV) below which residual excitation is not possible and Q-value is used.

User: Enter the requested energy cutoff.

Section 2.7

For MT=22 and 28, the code detects the presence of the secondary alpha or proton energy spectra files. If found, the kinematics option will be available and the following message prompts:

Message: Direct energy balance? 0 or 1.

User: Enter 1 to chose direct energy balance method, and refer to section 2.9. Enter 0 to select the kinematics method.

Section 2.8

For MT=18, to compute prompt and delayed kerma factors and the number of fission neutrons $\nu\sigma_f$.

Message: FISSION: Include prompt gamma energy? 0 or 1

User: Enter 1 to deposit fission gamma energy locally by including it in the prompt kerma factor.

Section 2.9

For all other reactions, or if specified earlier, direct energy balance is used.

Message: DEBM: Local gamma deposition, mt=**? 0 or 1.

User: Enter 0 to subtract gamma energy in the ENDF/B file for this specific reaction if it exists, enter 1 otherwise.

Section 3 (Radioactive Decay Kerma Factors)

The radioactive decay kerma factor is computed from:
decay kerma factor = sum (cross section * corresponding decay energy).

Message: Input decay kind: 1 charged particles only

2 gamma only

3 both separately

4 both combined

User: If distinction between gamma and charged particle decay input is desired, enter 3. In this case the following messages will appear twice; once for each decay kind.

Message: DECAY: Input reaction number, decay energy (eV)

User: Input a pair of numbers for each ENDF/B-V reaction producing radioactive nuclide with half-life below an arbitrary cutoff: MT, and the corresponding decay energy in eV. This message will be repeated until a zero is entered for the reaction number MT.

Message: Print decay/store it/add to prompt? 3: 0 or 1.

User: Enter three numbers, the ones being the affirmative to do the following:

- * Print total decay kerma factors in OUTPUT file.
- * Store it on a labeled text file for further processing.
- * Add it to the total prompt kerma factors to get the total kerma factors. In the case adding is requested, the following message will be sent.

Message: Print/store total kerma factors? 2: 0 or 1.

User: Enter two numbers, ones for affirmative to:

- * Print total kerma factors in OUTPUT file.
- * Store it in a labeled text file for further processing.

Section 4 (alternative kerma factors computation)

New responses can be calculated in this mode. This is actually an "array calculator" where the basic arithmetic operations are performed between arrays and TAPE contents where mesh representation and interpolation are taken care of.

Message: Use old (z)/need pkerma/need decay? 3: 0 or 1.

Input number of needed old tapes.

User: Input 4 numbers, the first three with ones for affirmative will decide on:

- * Using old array (z) contents as a background for the new kerma factors. Array (z) might be holding total prompt or total kerma factors with decay heat. Otherwise, it will be zeroed out.

* Enter prompt kerma factor computation mode.

* Enter decay heating computation mode.

For the fourth number, enter the number of old tapes that will be used (or a larger number). A zero will skip tape processing, and the next message will not appear.

Message: Tape number? /multiplier?/add(1) sub(-1) mult(2) div(-2)

User: Enter the appropriate tape number as referred to by previous diagnostic messages. Also enter a number to be multiplied by the contents of the tape before performing the array arithmetic. Enter 1 to add, -1 to subtract, 2 to multiply, -2 to divide the contents of array(z) by the contents of the designated TAPE. We are therefore calculating:
array(z)=array(z) [+,-,*, or /] (multiplier * TAPE)
The message will be repeated for the requested number of TAPES or until a zero is entered for the tape number, then prompt kerma and/or decay modes will begin if specified.

Message: Store new (z)? 0 or 1

User: Enter 1 to store the new array (z) and respond to the next message.

Message: Input title:

User: Enter the title of a new TAPE with no more than 72 characters.

Message: More new combinations? 0 or 1

User: Enter 1 to get the first message of section 4. Enter 0 to exit this mode.

Section 5 (Response Functions)

Response functions are computed in the form:

$$\text{response} = \text{sum} (\text{cross section} * \text{multiplier}).$$

The response input will be obtained from the file RESPONSE if it exists. If the file RESPONSE is not provided, interactive response to the following messages is required.

Message: Input response title.

User: Type a title of maximum 72 characters. Only 60 characters may appear on plots.

Message: Input mt [LR] and multiplier.

User: Input ENDF/B reaction number and a multiplier. In case reaction number is not found, it will be taken as the corresponding LR flag and will be searched for in the 51-91 reaction series. This message will be repeated until a zero is entered for reaction number. In this case the following message is sent.

Message: Print/store this response/more? 3: 0 or 1.

User: Enter three numbers, ones being affirmative to do the following:

- * Print the response in OUTPUT file.
- * Store it in a labeled text file for further processing.
- * Compute more responses. In this case, the first message in this section will be sent. If 0 is entered a diagnostic message indicating the number of responses will be sent, and the response mode will terminate.

Section 6 (Editing Point Tapes)

Message: Point editing? 0 or 1.

User: Enter 1 if plots or comparison between tapes holding point data is required. If 0 is entered, the next message will be skipped.

Message: Input 2 tape labels/plot(1)/plot(2)/plot(1,2).

Print comparison? 6: 0 or 1.

User: Six numbers are required. The first two are the labels of two tapes to be subject to the following actions. This message will be repeated until either tape label is zero. The next four numbers with ones for affirmative decide on the following actions:

- * Plot the contents of the first tape.
- * Plot the contents of the second tape.
- * Plot the contents of the two tapes along with the relative difference (if it spans a minimum range) on the same frame.
- * Print comparison table in OUTPUT file.

Message: Group editing? 0=no, 1=some, 2=all.

User: 0 no group processing.

1 group processing of some tapes to be specified in the following two messages.

2 group processing of all tapes.

Message: Input number of tapes for grouping.

User: Enter the number of tapes whose contents will be grouped and the group values will be printed in OUTPUT, also group plots will be produced.

Message: Input tape labels:

User: Enter a string of tape labels with the length specified by the above answer. A wrong tape label will be ignored.

Message: Input group kind, flat=1, weighted=2.

User: This message prompts for every requested grouping.

1 flat weighting.

2 the weighting spectrum provided in file GROUP will be used. If not provided, flat option is used.

Message: Group comparison? 0 or 1

User: 0 Code terminates.

1 Comparison between grouped data stored in GTAPE(n) files requested. Respond to the following messages.

Message: Input two GTAPE labels:

User: Enter n1 or n2 to compare the contents of the GTAPEn1 and GTAPEn2 files. If one of the specified GTAPES is not found, a warning message prompts and the request for GTAPE numbers will be repeated. This message will be repeated until either n1 or n2 = 0 is specified, and the code terminates.

III.4 Sample Problem

As a test problem, the neutron kerma factors and other nuclear response functions will be calculated for natural Titanium. The

actual evaluation effort including different data tests, comparisons and analysis can not be duplicated here. Only the final procedure will be shown.

Required Files

The user needs to have the following files on his MFE disk space: **KAOS** the KAOS-V code source file, **GROUP** file containing Vitamin-E group structure and weighting spectrum, and the **ENDF/B-V** file which is **T508** for Titanium. No **PENDF** file is needed for this material since no resonance parameters are given in **ENDF/B-V**. Files **TAPE1**, **TAPE2**, and **TAPE3** contain the energy-dependent Q-values for the $Ti(n,2n)$, $Ti(n,p)$, and $Ti(n,\alpha)$ reactions. These can be obtained from the **QTAPE** library which is available with the **KAOS** package. Standard **RESPONSE** file should be also present to request non-kerma responses.

The **INPUT** file controlling the run is listed below:

```
2 , Direct energy Balance
1322 , ENDF/B-V MAT number for Titanium
1 , Compute prompt kerma factors
0 , Do not compute decay heat
1 , Compute non-kerma responses
1 , Group and plot the Results
```

Notice that the **KAOS-V** code does not read the comments following each data entry.

NATURAL file Listing

-16 , 1 Q-value for MT=16 (n,2n) given on TAPE1
-103 , 2 Q-value for MT=103 (n,p) given on TAPE2
-107 , 3 Q-value for MT=107 (n,a) given on TAPE3

Again, the KAOS-V code does not read the comments following data entries.

RESPONSE File Listing

prompt gamma energy production

-1,1.

0,0.

total h-1 production cross section

28,1.

103,1.

111,2.

112,1.

0,0.

total d production cross section

32,1.

35,1.

104,1.

114,1.

0,0.

total t production cross section

33,1.

36,1.

105,1.

113,1.

0,0.

total he-3 production cross section
 34,1.
 106,1.
 0,0.
 total he-4 production cross section
 22,1.
 23,3.
 24,1.
 25,1.
 29,2.
 30,2.
 35,2.
 36,2.
 107,1.
 108,2.
 109,3.
 112,1.
 113,2.
 114,2.
 0,0.
 total cross section
 1,1.
 0,0
 elastic cross section
 2,1.
 0,0.
 total inelastic cross section
 4,1.
 0,0.
 radiative capture cross section
 102,1.
 0,0.

total (n,2n) reaction cross section

6,1.

7,1.

8,1.

9,1.

16,1.

24,1.

30,1.

0,0.

(n,3n) reaction cross section

17,1.

25,1.

0,0.

TAPE3 file listing

The TAPE(n) files can be quite lengthy. Only one of the needed three files will be listed here. This file TAPE3 represents the variable Q-value for $Ti(n,\alpha)$ reaction as a function of the incident neutron energy using 9 points and linear interpolation. The file is listed below:

Q-value for $Ti(n,\alpha)$ Reaction

Incident Neutron Energy eV

Q-value eV

9 2 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0 0

9

1.0e-5 , 1.357e6

3.0e6 , 1.357e6

4.0e6 , 7.9e5

5.0e6 , 7.0e5

6.0e6 , 0.0
7.0e6 , -1.0e6
7.5e6 , -1.2e6
1.0e7 , -1.3e6
2.0e7 , -1.35e6

Compilation and Loading Line

```
RCFT I=KAOS, LIB=(DISSPLA,FORTLIB) / T V
```

Notice that the graphics library DISSPLA is used. The executable file (controllee) name will be XKAOS.

Execute Line

```
XKAOS ENDF=T508 / T V
```

The expected CPU run time is less than 1.5 minutes.

Produced Files

The following files are produced:

DIAGNOSE file: contains run summary, titles of the produced TAPE(n) files, CPU time consumption details, etc.

TAPE4, TAPE5, ..., TAPE32 files: the titles describing the contents of these files are to be found in the DIAGNOSE file. Following a final successful run in general, the user do not need to keep any TAPE(n) file.

F3PLOTX file: is the graphics file produced by DISSPLA. To view or get a hard copy of these plots, use NETPLOT.

OUTPUT file: The output file contains the grouped partial and total kerma factors and response functions output. This file is listed below:

```
tape 1:
group q-value for ti-nat(n,2n)
```

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	-5.599e+06
-8.189e+06	-8.232e+06	-8.278e+06	-8.328e+06	-8.379e+06	-8.466e+06
-8.612e+06	-8.967e+06	-9.700e+06	-9.810e+06	-9.906e+06	-1.000e+07
-1.011e+07	-1.022e+07	-1.050e+07	-1.059e+07	-1.064e+07	-1.081e+07

```
tape 2:
group q-value fot ti-nat(n,p)
```

1.819e+05	1.819e+05	1.819e+05	1.819e+05	1.819e+05	1.819e+05
1.819e+05	1.819e+05	1.819e+05	1.819e+05	1.819e+05	1.819e+05

[illegible][illegible]

1.447e+04	1.217e+04	9.771e+03	3.389e+04	3.636e+04	2.272e+04
2.036e+04	1.895e+04	1.723e+04	2.173e+04	2.733e+04	1.676e+04
2.448e+04	3.093e+04	5.160e+04	2.854e+04	4.505e+04	2.937e+04
5.537e+04	6.304e+04	5.080e+04	9.692e+04	8.475e+04	8.306e+04
5.558e+04	6.380e+04	8.703e+04	7.588e+04	9.867e+04	7.528e+04
6.164e+04	8.408e+04	8.308e+04	1.029e+05	1.421e+05	1.232e+05
1.130e+05	1.243e+05	1.146e+05	1.284e+05	1.479e+05	1.631e+05
1.528e+05	1.523e+05	1.578e+05	1.643e+05	1.636e+05	1.882e+05
1.866e+05	1.650e+05	1.684e+05	1.419e+05	1.543e+05	1.383e+05
1.420e+05	1.379e+05	1.288e+05	1.455e+05	1.400e+05	1.355e+05
1.313e+05	1.250e+05	1.234e+05	1.164e+05	1.101e+05	1.031e+05
1.018e+05	1.016e+05	1.011e+05	1.010e+05	1.012e+05	1.008e+05
1.011e+05	1.018e+05	1.022e+05	1.024e+05	1.026e+05	1.042e+05
1.065e+05	1.073e+05	1.087e+05	1.096e+05	1.108e+05	1.125e+05
1.146e+05	1.171e+05	1.236e+05	1.273e+05	1.298e+05	1.388e+05

tape 5:

group prompt kerma, direct energy balance, mat= 1322 mt= 16

[illegible]

tape 6:
group prompt kerma, direct energy balance, mat= 1322 mt= 17

86

tape 8:

group prompt kerma, direct energy balance, mat= 1322 mt= 28

[illegible]

tape 9:

group level inelastic prompt kerma, mat= 1322 mt= 5162

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
6.191e+01	6.638e+02	1.421e+03	2.217e+03	3.031e+03	3.347e+03
3.447e+03	3.552e+03	3.663e+03	3.840e+03	4.030e+03	4.130e+03
4.177e+03	4.196e+03	4.218e+03	4.382e+03	4.696e+03	4.952e+03
5.129e+03	5.320e+03	5.578e+03	5.788e+03	5.937e+03	6.094e+03
6.255e+03	6.334e+03	6.383e+03	6.434e+03	6.487e+03	6.543e+03
6.602e+03	6.664e+03	6.730e+03	1.150e+04	2.006e+04	9.347e+04
2.429e+05	3.113e+05	4.354e+05	3.589e+05	3.439e+05	6.138e+05
5.142e+05	6.544e+05	6.855e+05	8.407e+05	8.379e+05	9.002e+05
8.903e+05	9.198e+05	9.212e+05	9.121e+05	9.191e+05	9.271e+05
9.698e+05	1.253e+06	1.384e+06	1.505e+06	1.681e+06	1.826e+06
1.898e+06	2.697e+06	2.947e+06	2.706e+06	2.362e+06	2.144e+06
1.933e+06	1.742e+06	1.542e+06	1.334e+06	1.173e+06	1.055e+06
9.811e+05	8.824e+05	7.434e+05	6.090e+05	4.777e+05	4.166e+05
3.690e+05	3.187e+05	2.660e+05	2.285e+05	2.103e+05	1.911e+05
1.709e+05	1.551e+05	1.213e+05	1.089e+05	9.964e+04	9.852e+04
9.859e+04	9.867e+04	9.888e+04	9.900e+04	9.909e+04	9.936e+04

tape10:

group prompt kerma, direct energy balance, mat= 1322 mt= 91

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	1.123e+04	1.046e+05	6.185e+05	1.227e+06	1.652e+06
2.104e+06	2.580e+06	3.080e+06	3.605e+06	4.050e+06	4.398e+06
4.614e+06	4.904e+06	5.352e+06	5.815e+06	6.284e+06	6.644e+06
6.993e+06	7.334e+06	7.673e+06	7.949e+06	8.141e+06	8.174e+06
8.059e+06	7.745e+06	6.845e+06	6.499e+06	6.184e+06	5.841e+06
5.486e+06	5.122e+06	4.388e+06	4.057e+06	3.841e+06	3.259e+06

tapell1:

group capture prompt kerma, mat= 1322

2.880e+03	1.385e+03	7.139e+02	6.278e+02	5.512e+02	4.831e+02
4.232e+02	3.693e+02	3.220e+02	2.792e+02	2.416e+02	2.067e+02
1.759e+02	1.497e+02	1.266e+02	1.047e+02	8.651e+01	7.118e+01
5.755e+01	4.624e+01	3.694e+01	2.953e+01	2.366e+01	1.916e+01
1.572e+01	1.308e+01	1.110e+01	9.503e+00	8.352e+00	7.359e+00
6.484e+00	5.809e+00	5.276e+00	4.943e+00	4.705e+00	4.545e+00
4.782e+00	5.235e+00	6.126e+00	7.722e+00	2.840e+01	8.566e+01
1.069e+01	3.641e+01	6.197e+00	7.026e+00	2.080e+01	1.657e+01
7.982e+01	5.324e+01	3.716e+01	6.525e+01	4.692e+01	1.392e+02
3.116e+01	1.252e+01	1.428e+01	1.440e+01	8.298e+00	1.385e+01
5.293e+01	2.403e+01	2.594e+01	2.296e+01	1.018e+01	1.719e+01
1.844e+01	4.840e+01	2.269e+01	2.783e+01	1.267e+01	3.156e+01
3.051e+01	8.549e+00	2.223e+01	2.261e+01	1.231e+01	3.995e+01
1.084e+01	2.306e+01	9.692e+00	3.264e+01	2.188e+01	1.816e+01
2.046e+01	2.289e+01	2.543e+01	2.768e+01	2.840e+01	2.876e+01

2.894e+01	2.901e+01	2.910e+01	3.002e+01	3.175e+01	3.299e+01
3.386e+01	3.563e+01	3.818e+01	4.029e+01	4.205e+01	4.383e+01
4.566e+01	4.773e+01	4.997e+01	5.229e+01	5.487e+01	5.768e+01
6.060e+01	6.337e+01	6.615e+01	6.403e+01	5.949e+01	5.345e+01
4.957e+01	4.938e+01	4.986e+01	5.074e+01	5.213e+01	4.877e+03
1.722e+04	1.700e+04	1.683e+04	1.682e+04	1.689e+04	1.721e+04
1.711e+04	1.670e+04	1.634e+04	1.611e+04	1.599e+04	1.591e+04
1.572e+04	1.543e+04	1.540e+04	1.538e+04	1.537e+04	1.497e+04
1.424e+04	1.308e+04	1.140e+04	1.028e+04	9.265e+03	7.877e+03
6.747e+03	6.149e+03	5.630e+03	5.221e+03	4.879e+03	4.617e+03
4.471e+03	4.282e+03	4.013e+03	3.760e+03	3.525e+03	3.314e+03
3.130e+03	2.983e+03	2.891e+03	2.884e+03	3.022e+03	3.402e+03
4.197e+03	5.177e+03	9.738e+03	1.343e+04	1.813e+04	2.482e+04
3.390e+04	4.655e+04	1.039e+05	1.505e+05	1.837e+05	2.244e+05

tape12:

group prompt kerma, direct energy balance, mat= 1322 mt= 103

1.829e-12	8.327e-12	3.390e-11	4.398e-11	5.691e-11	7.352e-11
9.485e-11	1.222e-10	1.574e-10	2.026e-10	2.605e-10	3.350e-10
4.306e-10	5.533e-10	7.110e-10	9.134e-10	1.173e-09	1.507e-09
1.935e-09	2.485e-09	3.191e-09	4.098e-09	5.262e-09	6.758e-09
8.677e-09	1.114e-08	1.431e-08	1.837e-08	2.359e-08	3.029e-08
3.890e-08	4.994e-08	6.412e-08	8.234e-08	1.057e-07	1.358e-07
1.614e-07	1.784e-07	1.922e-07	2.020e-07	2.178e-07	2.408e-07
2.661e-07	3.017e-07	3.691e-07	4.739e-07	6.084e-07	7.419e-07
8.402e-07	1.003e-06	1.288e-06	1.551e-06	1.713e-06	1.801e-06
1.846e-06	1.917e-06	2.001e-06	2.092e-06	2.273e-06	2.493e-06
2.827e-06	3.283e-06	3.720e-06	4.109e-06	4.662e-06	5.254e-06
5.708e-06	6.107e-06	6.374e-06	6.951e-06	7.876e-06	8.593e-06
9.032e-06	9.496e-06	9.978e-06	1.049e-05	1.103e-05	1.160e-05
1.219e-05	1.281e-05	1.348e-05	1.416e-05	1.489e-05	1.565e-05
1.646e-05	1.730e-05	1.819e-05	1.961e-05	2.113e-05	2.194e-05
2.232e-05	2.247e-05	2.264e-05	2.395e-05	2.647e-05	2.853e-05
2.998e-05	3.233e-05	3.573e-05	3.851e-05	4.047e-05	4.254e-05
4.472e-05	4.703e-05	4.943e-05	5.198e-05	5.463e-05	5.744e-05
6.038e-05	1.019e-04	3.222e-04	5.745e-03	7.559e-02	4.379e+01
1.120e+02	2.053e+02	3.108e+02	3.667e+02	4.191e+02	4.743e+02
5.321e+02	6.575e+02	1.068e+03	1.540e+03	2.031e+03	2.549e+03

3.098e+03	3.669e+03	4.377e+03	5.243e+03	5.645e+03	5.740e+03
5.964e+03	6.126e+03	6.317e+03	7.081e+03	8.699e+03	1.058e+04
1.270e+04	1.622e+04	2.256e+04	3.096e+04	3.892e+04	4.509e+04
5.236e+04	6.240e+04	7.363e+04	8.814e+04	1.048e+05	1.203e+05
1.300e+05	1.432e+05	1.673e+05	1.958e+05	2.283e+05	2.635e+05
3.035e+05	3.506e+05	4.054e+05	4.704e+05	5.418e+05	6.194e+05
7.031e+05	7.623e+05	8.810e+05	9.115e+05	9.307e+05	9.435e+05
9.556e+05	9.662e+05	9.622e+05	9.519e+05	9.411e+05	8.964e+05

```

tapel3:
group prompt kerma, direct energy balance, mat= 1322 mt= 104

```

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	1.814e-01
1.649e+00	3.504e+00	6.929e+00	1.100e+02	3.795e+02	6.673e+02

1.116e+03	2.622e+03	6.890e+03	8.455e+03	1.087e+04	1.793e+04
2.583e+04	3.442e+04	6.481e+04	1.031e+05	1.312e+05	2.617e+05

tape14:

group prompt kerma, direct energy balance, mat= 1322 mt= 105

[illegible]

tape15:

group prompt kerma, direct energy balance, mat= 1322 mt= 106

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	8.385e-07	1.868e-05	5.098e-05	9.397e-05
2.497e-02	2.450e-01	1.172e+00	1.623e+00	4.571e+00	1.985e+01
4.218e+01	6.742e+01	2.092e+02	5.205e+02	7.546e+02	4.937e+03

tape16:

group prompt kerma, direct energy balance, mat= 1322 mt= 107

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	9.582e-02	2.532e+00
6.136e+00	9.913e+00	1.389e+01	2.027e+01	2.709e+01	3.071e+01
3.242e+01	3.308e+01	3.385e+01	3.973e+01	5.101e+01	6.024e+01
6.679e+01	7.732e+01	9.257e+01	1.050e+02	1.138e+02	1.231e+02
1.329e+02	1.432e+02	1.540e+02	1.654e+02	1.773e+02	1.899e+02
2.031e+02	2.169e+02	2.316e+02	2.482e+02	2.644e+02	2.993e+02
3.425e+02	3.736e+02	4.059e+02	4.401e+02	4.761e+02	5.137e+02
5.533e+02	5.951e+02	6.390e+02	6.848e+02	7.335e+02	7.860e+02
9.126e+02	1.169e+03	1.410e+03	1.559e+03	1.637e+03	1.687e+03
1.814e+03	2.086e+03	2.419e+03	2.775e+03	3.151e+03	3.617e+03
4.175e+03	5.069e+03	6.466e+03	8.720e+03	1.091e+04	1.244e+04
1.408e+04	1.580e+04	1.762e+04	1.955e+04	2.174e+04	2.374e+04
2.499e+04	2.666e+04	2.984e+04	3.352e+04	3.784e+04	4.582e+04
5.493e+04	6.611e+04	7.903e+04	9.563e+04	1.158e+05	1.368e+05
1.610e+05	1.952e+05	2.821e+05	3.140e+05	3.467e+05	3.987e+05
4.547e+05	5.157e+05	6.889e+05	8.002e+05	8.770e+05	1.117e+06

tape17:

group prompt kerma, direct energy balance, mat= 1322 mt= 111

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	1.540e-07
2.787e-06	6.465e-06	1.032e-05	1.141e-03	3.723e-03	7.369e-03
1.729e+00	1.679e+01	7.000e+01	9.345e+01	1.525e+02	3.847e+02
6.479e+02	9.341e+02	1.974e+03	3.355e+03	4.370e+03	7.887e+03

tape18:

group prompt kerma, direct energy balance, mat= 1322 mt= 112

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	4.998e-09	2.527e-08	7.359e-08
3.501e-07	2.231e-06	7.759e-06	9.787e-06	1.118e-03	7.241e-03
1.494e-02	2.571e-02	6.081e-01	2.717e+00	4.324e+00	4.966e+01

tape19:

group gamma production less capture, mat=1322

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
6.009e+01	6.444e+02	1.379e+03	2.152e+03	2.942e+03	3.240e+03
3.324e+03	3.413e+03	3.507e+03	3.657e+03	3.817e+03	3.902e+03
3.942e+03	3.958e+03	3.977e+03	4.115e+03	4.380e+03	4.597e+03
4.745e+03	4.884e+03	5.063e+03	5.209e+03	5.313e+03	5.423e+03
5.534e+03	5.567e+03	5.569e+03	5.572e+03	5.575e+03	5.578e+03
5.581e+03	5.584e+03	5.588e+03	1.018e+04	1.847e+04	9.558e+04
2.492e+05	3.209e+05	4.466e+05	3.781e+05	3.696e+05	6.168e+05
5.413e+05	6.041e+05	6.701e+05	7.395e+05	8.125e+05	8.892e+05

9.681e+05	1.050e+06	1.121e+06	1.166e+06	1.188e+06	1.204e+06
1.242e+06	1.322e+06	1.422e+06	1.528e+06	1.638e+06	1.781e+06
1.950e+06	2.219e+06	2.611e+06	3.044e+06	3.403e+06	3.655e+06
3.914e+06	4.175e+06	4.449e+06	4.738e+06	5.040e+06	5.306e+06
5.471e+06	5.693e+06	6.011e+06	6.323e+06	6.652e+06	6.998e+06
7.359e+06	7.672e+06	7.948e+06	8.247e+06	8.552e+06	8.650e+06
8.594e+06	8.550e+06	8.343e+06	8.193e+06	8.046e+06	7.846e+06
7.634e+06	7.435e+06	7.229e+06	7.111e+06	7.029e+06	7.113e+06

tape20:

group nr=1, mat=1322 prompt (deb) kerma for titanium-nat

2.880e+03	1.385e+03	7.140e+02	6.279e+02	5.513e+02	4.834e+02
4.235e+02	3.696e+02	3.224e+02	2.797e+02	2.422e+02	2.075e+02
1.770e+02	1.511e+02	1.283e+02	1.069e+02	8.931e+01	7.479e+01
6.219e+01	5.221e+01	4.459e+01	3.936e+01	3.627e+01	3.533e+01
3.647e+01	3.967e+01	4.520e+01	5.322e+01	6.427e+01	7.901e+01
9.977e+01	1.283e+02	1.667e+02	2.219e+02	2.971e+02	3.991e+02
4.997e+02	5.743e+02	6.481e+02	7.238e+02	1.192e+03	2.596e+03
1.268e+03	2.297e+03	1.830e+03	2.950e+03	6.282e+03	9.881e+03
1.589e+04	3.138e+04	7.218e+04	5.329e+04	6.367e+04	4.957e+04
4.233e+04	3.592e+04	3.197e+04	2.825e+04	1.798e+04	1.124e+04
3.046e+04	1.445e+04	2.500e+04	4.314e+04	2.278e+04	1.545e+04
1.624e+04	1.443e+04	1.599e+04	1.218e+04	1.422e+04	1.023e+04
1.533e+04	7.975e+03	1.160e+04	1.109e+04	8.289e+03	1.164e+04
7.249e+03	6.550e+03	7.115e+03	2.408e+04	2.810e+04	1.644e+04
1.462e+04	1.234e+04	9.966e+03	3.412e+04	3.663e+04	2.301e+04
2.066e+04	1.925e+04	1.754e+04	2.207e+04	2.773e+04	1.721e+04
2.497e+04	3.147e+04	5.225e+04	2.927e+04	4.583e+04	3.021e+04
5.627e+04	6.400e+04	5.182e+04	9.800e+04	8.589e+04	8.427e+04
5.686e+04	6.516e+04	8.847e+04	7.752e+04	1.006e+05	7.356e+04
5.590e+04	7.508e+04	7.257e+04	8.455e+04	1.174e+05	1.261e+05
1.042e+05	1.929e+05	1.485e+05	2.487e+05	1.929e+05	1.947e+05
9.613e+04	4.362e+04	-2.036e+04	-6.648e+04	-8.247e+04	-6.504e+04
-6.245e+04	1.196e+05	1.544e+05	1.448e+05	2.239e+05	2.130e+05
1.216e+05	6.618e+05	6.106e+05	4.795e+05	3.866e+05	3.445e+05
3.283e+05	3.583e+05	3.964e+05	4.325e+05	4.253e+05	3.985e+05
3.853e+05	3.693e+05	3.866e+05	4.345e+05	4.810e+05	4.771e+05
4.663e+05	5.047e+05	5.852e+05	6.225e+05	6.071e+05	6.562e+05

7.418e+05	6.624e+05	4.368e+05	3.691e+05	3.058e+05	3.233e+05
3.471e+05	3.592e+05	3.511e+05	5.647e+05	7.432e+05	1.159e+06

tape21:

group total prompt gamma production, mat=1322

4.820e+07	2.318e+07	1.195e+07	1.050e+07	9.221e+06	8.084e+06
7.081e+06	6.179e+06	5.388e+06	4.672e+06	4.042e+06	3.458e+06
2.943e+06	2.505e+06	2.117e+06	1.751e+06	1.446e+06	1.190e+06
9.618e+05	7.727e+05	6.170e+05	4.930e+05	3.948e+05	3.193e+05
2.619e+05	2.175e+05	1.842e+05	1.574e+05	1.378e+05	1.210e+05
1.062e+05	9.460e+04	8.531e+04	7.926e+04	7.458e+04	7.088e+04
7.360e+04	7.990e+04	9.288e+04	1.165e+05	4.246e+05	1.272e+06
1.564e+05	5.234e+05	8.681e+04	9.376e+04	2.613e+05	1.961e+05
9.150e+05	5.812e+05	3.663e+05	5.915e+05	4.129e+05	1.184e+06
2.616e+05	1.032e+05	1.152e+05	1.140e+05	6.254e+04	9.904e+04
3.539e+05	1.470e+05	1.440e+05	1.196e+05	4.914e+04	7.513e+04
7.609e+04	1.919e+05	8.659e+04	9.929e+04	4.056e+04	9.516e+04
8.780e+04	2.365e+04	5.861e+04	5.724e+04	3.027e+04	9.323e+04
2.448e+04	5.080e+04	2.138e+04	6.691e+04	4.446e+04	3.586e+04
3.832e+04	4.090e+04	4.362e+04	4.491e+04	4.349e+04	4.270e+04
4.234e+04	4.219e+04	4.204e+04	4.152e+04	4.053e+04	3.972e+04
3.917e+04	3.873e+04	3.820e+04	3.790e+04	3.791e+04	3.790e+04
3.788e+04	3.789e+04	3.791e+04	3.792e+04	3.804e+04	3.821e+04
3.836e+04	3.838e+04	3.833e+04	4.053e+04	4.541e+04	1.183e+05
2.689e+05	3.397e+05	4.648e+05	3.958e+05	3.870e+05	6.254e+05
5.413e+05	6.041e+05	6.701e+05	7.395e+05	8.125e+05	8.892e+05
9.681e+05	1.050e+06	1.121e+06	1.166e+06	1.188e+06	1.204e+06
1.242e+06	1.322e+06	1.422e+06	1.528e+06	1.638e+06	1.781e+06
1.950e+06	2.219e+06	2.611e+06	3.044e+06	3.403e+06	3.655e+06
3.914e+06	4.175e+06	4.449e+06	4.738e+06	5.040e+06	5.306e+06
5.471e+06	5.693e+06	6.011e+06	6.323e+06	6.652e+06	6.998e+06
7.359e+06	7.672e+06	7.948e+06	8.247e+06	8.552e+06	8.650e+06
8.594e+06	8.550e+06	8.343e+06	8.193e+06	8.046e+06	7.846e+06
7.634e+06	7.435e+06	7.229e+06	7.111e+06	7.029e+06	7.113e+06

tape22:

group total h-1 production cross section

3.182e-18	9.776e-18	3.505e-17	4.501e-17	5.779e-17	7.420e-17
9.528e-17	1.224e-16	1.571e-16	2.017e-16	2.590e-16	3.326e-16
4.271e-16	5.484e-16	7.041e-16	9.041e-16	1.161e-15	1.491e-15
1.914e-15	2.457e-15	3.155e-15	4.051e-15	5.202e-15	6.680e-15
8.577e-15	1.101e-14	1.414e-14	1.816e-14	2.332e-14	2.994e-14
3.844e-14	4.936e-14	6.337e-14	8.138e-14	1.045e-13	1.342e-13
1.595e-13	1.763e-13	1.900e-13	1.997e-13	2.153e-13	2.380e-13
2.630e-13	2.982e-13	3.648e-13	4.684e-13	6.015e-13	7.334e-13
8.307e-13	9.917e-13	1.273e-12	1.533e-12	1.694e-12	1.780e-12
1.825e-12	1.895e-12	1.978e-12	2.068e-12	2.246e-12	2.464e-12
2.795e-12	3.245e-12	3.678e-12	4.063e-12	4.608e-12	5.194e-12
5.643e-12	6.038e-12	6.299e-12	6.871e-12	7.785e-12	8.493e-12
8.928e-12	9.385e-12	9.867e-12	1.037e-11	1.090e-11	1.146e-11
1.205e-11	1.267e-11	1.332e-11	1.400e-11	1.472e-11	1.547e-11
1.627e-11	1.710e-11	1.798e-11	1.939e-11	2.089e-11	2.168e-11
2.206e-11	2.221e-11	2.238e-11	2.368e-11	2.617e-11	2.820e-11
2.964e-11	3.196e-11	3.532e-11	3.806e-11	4.001e-11	4.206e-11
4.422e-11	4.649e-11	4.887e-11	5.138e-11	5.401e-11	5.678e-11
5.969e-11	9.823e-11	2.945e-10	4.895e-09	5.999e-08	3.303e-05
8.390e-05	1.476e-04	2.151e-04	2.403e-04	2.614e-04	2.836e-04
3.069e-04	3.617e-04	5.244e-04	7.044e-04	8.937e-04	1.093e-03
1.302e-03	1.522e-03	1.785e-03	2.092e-03	2.225e-03	2.245e-03
2.288e-03	2.261e-03	2.226e-03	2.390e-03	2.837e-03	3.355e-03
3.930e-03	4.905e-03	6.625e-03	8.591e-03	1.022e-02	1.138e-02
1.265e-02	1.437e-02	1.611e-02	1.843e-02	2.097e-02	2.314e-02
2.441e-02	2.609e-02	2.900e-02	3.224e-02	3.566e-02	3.906e-02
4.258e-02	4.664e-02	5.114e-02	5.611e-02	6.212e-02	6.944e-02
7.777e-02	8.908e-02	1.166e-01	1.253e-01	1.323e-01	1.391e-01
1.459e-01	1.532e-01	1.721e-01	1.845e-01	1.931e-01	2.090e-01

tape23:

group total d production cross section

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

[illegible]

tape25:

group total he-3 production cross section

[illegible]

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	1.754e-12	1.511e-11	3.202e-11	4.970e-11
8.510e-09	8.272e-08	3.021e-07	3.826e-07	9.760e-07	3.935e-06
7.312e-06	1.099e-05	2.903e-05	6.455e-05	9.115e-05	4.903e-04

tape26:

group total he-4 production cross section

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	4.096e-08	1.083e-06
2.608e-06	4.213e-06	5.901e-06	8.612e-06	1.150e-05	1.304e-05
1.376e-05	1.404e-05	1.438e-05	1.687e-05	2.167e-05	2.557e-05
2.835e-05	3.282e-05	3.929e-05	4.456e-05	4.832e-05	5.227e-05
5.642e-05	6.078e-05	6.537e-05	7.019e-05	7.526e-05	8.059e-05
8.620e-05	9.209e-05	9.828e-05	1.053e-04	1.122e-04	1.215e-04
1.317e-04	1.389e-04	1.465e-04	1.544e-04	1.628e-04	1.716e-04
1.809e-04	1.906e-04	2.008e-04	2.116e-04	2.229e-04	2.352e-04

2.648e-04	3.188e-04	3.685e-04	3.993e-04	4.150e-04	4.256e-04
4.525e-04	5.079e-04	5.774e-04	6.506e-04	7.275e-04	8.222e-04
9.321e-04	1.107e-03	1.363e-03	1.702e-03	2.007e-03	2.222e-03
2.455e-03	2.713e-03	2.984e-03	3.270e-03	3.624e-03	3.958e-03
4.164e-03	4.442e-03	4.815e-03	5.162e-03	5.571e-03	6.343e-03
7.224e-03	8.216e-03	9.302e-03	1.070e-02	1.238e-02	1.414e-02
1.610e-02	1.847e-02	2.427e-02	2.640e-02	2.854e-02	3.175e-02
3.520e-02	3.894e-02	4.996e-02	5.820e-02	6.402e-02	8.460e-02

tape27:

group total cross section

1.006e+01	7.128e+00	5.830e+00	5.659e+00	5.504e+00	5.374e+00
5.259e+00	5.162e+00	5.070e+00	4.993e+00	4.923e+00	4.865e+00
4.813e+00	4.762e+00	4.714e+00	4.675e+00	4.643e+00	4.613e+00
4.590e+00	4.569e+00	4.551e+00	4.534e+00	4.519e+00	4.505e+00
4.492e+00	4.480e+00	4.471e+00	4.461e+00	4.441e+00	4.429e+00
4.487e+00	4.586e+00	4.706e+00	4.867e+00	5.093e+00	5.402e+00
5.710e+00	5.957e+00	6.243e+00	6.627e+00	9.923e+00	1.991e+01
8.816e+00	1.424e+01	9.204e+00	1.152e+01	1.904e+01	2.474e+01
3.521e+01	5.760e+01	1.045e+02	6.528e+01	6.979e+01	5.163e+01
4.300e+01	3.519e+01	2.997e+01	2.541e+01	1.492e+01	8.533e+00
2.000e+01	8.386e+00	1.229e+01	1.992e+01	9.300e+00	5.567e+00
5.393e+00	4.503e+00	4.765e+00	3.335e+00	3.463e+00	2.290e+00
3.260e+00	1.624e+00	2.239e+00	2.049e+00	1.470e+00	1.964e+00
1.172e+00	1.016e+00	1.049e+00	3.380e+00	3.817e+00	2.128e+00
1.822e+00	1.478e+00	1.145e+00	3.705e+00	3.758e+00	2.293e+00
2.034e+00	1.887e+00	1.708e+00	2.061e+00	2.424e+00	1.401e+00
1.890e+00	2.188e+00	3.255e+00	1.727e+00	2.598e+00	1.594e+00
2.763e+00	2.849e+00	2.201e+00	4.117e+00	3.579e+00	3.443e+00
2.215e+00	2.425e+00	3.181e+00	2.699e+00	3.532e+00	2.807e+00
2.329e+00	2.948e+00	2.848e+00	2.898e+00	3.480e+00	3.322e+00
2.964e+00	3.177e+00	3.032e+00	3.106e+00	3.335e+00	3.691e+00
3.549e+00	3.606e+00	3.594e+00	3.593e+00	3.529e+00	3.906e+00
3.899e+00	3.826e+00	4.016e+00	3.541e+00	3.826e+00	3.641e+00
3.736e+00	3.756e+00	3.608e+00	3.685e+00	3.599e+00	3.572e+00
3.550e+00	3.466e+00	3.478e+00	3.412e+00	3.371e+00	3.307e+00
3.279e+00	3.246e+00	3.194e+00	3.140e+00	3.083e+00	3.018e+00
2.949e+00	2.877e+00	2.801e+00	2.719e+00	2.635e+00	2.558e+00

2.484e+00	2.427e+00	2.376e+00	2.362e+00	2.350e+00	2.338e+00
2.327e+00	2.315e+00	2.281e+00	2.260e+00	2.247e+00	2.216e+00

tape28:

group elastic cross section

4.393e+00	4.403e+00	4.415e+00	4.417e+00	4.420e+00	4.423e+00
4.425e+00	4.430e+00	4.435e+00	4.440e+00	4.446e+00	4.451e+00
4.455e+00	4.460e+00	4.464e+00	4.469e+00	4.471e+00	4.473e+00
4.475e+00	4.477e+00	4.478e+00	4.475e+00	4.472e+00	4.467e+00
4.461e+00	4.455e+00	4.450e+00	4.442e+00	4.425e+00	4.415e+00
4.475e+00	4.575e+00	4.696e+00	4.857e+00	5.085e+00	5.394e+00
5.702e+00	5.948e+00	6.233e+00	6.614e+00	9.873e+00	1.976e+01
8.798e+00	1.417e+01	9.193e+00	1.151e+01	1.901e+01	2.472e+01
3.511e+01	5.753e+01	1.044e+02	6.521e+01	6.974e+01	5.149e+01
4.297e+01	3.517e+01	2.996e+01	2.539e+01	1.492e+01	8.522e+00
1.996e+01	8.368e+00	1.227e+01	1.991e+01	9.294e+00	5.558e+00
5.384e+00	4.481e+00	4.755e+00	3.324e+00	3.458e+00	2.279e+00
3.250e+00	1.621e+00	2.232e+00	2.043e+00	1.467e+00	1.953e+00
1.169e+00	1.006e+00	1.038e+00	3.359e+00	3.794e+00	2.103e+00
1.797e+00	1.453e+00	1.118e+00	3.678e+00	3.730e+00	2.264e+00
2.005e+00	1.857e+00	1.679e+00	2.031e+00	2.392e+00	1.368e+00
1.856e+00	2.153e+00	3.219e+00	1.690e+00	2.561e+00	1.556e+00
2.724e+00	2.811e+00	2.162e+00	4.079e+00	3.540e+00	3.404e+00
2.177e+00	2.386e+00	3.142e+00	2.655e+00	3.479e+00	2.682e+00
2.056e+00	2.606e+00	2.385e+00	2.511e+00	3.111e+00	2.689e+00
2.432e+00	2.516e+00	2.355e+00	2.294e+00	2.533e+00	2.839e+00
2.718e+00	2.767e+00	2.769e+00	2.784e+00	2.719e+00	3.093e+00
3.059e+00	2.841e+00	2.995e+00	2.494e+00	2.694e+00	2.436e+00
2.523e+00	2.371e+00	2.201e+00	2.330e+00	2.287e+00	2.265e+00
2.245e+00	2.156e+00	2.164e+00	2.093e+00	2.051e+00	1.988e+00
1.961e+00	1.930e+00	1.888e+00	1.849e+00	1.807e+00	1.753e+00
1.695e+00	1.636e+00	1.573e+00	1.502e+00	1.427e+00	1.372e+00
1.330e+00	1.286e+00	1.200e+00	1.176e+00	1.159e+00	1.143e+00
1.130e+00	1.118e+00	1.085e+00	1.064e+00	1.050e+00	1.017e+00

tape29:

group total inelastic cross section

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
3.758e-04	4.006e-03	8.621e-03	1.350e-02	1.849e-02	2.037e-02
2.091e-02	2.147e-02	2.205e-02	2.300e-02	2.401e-02	2.454e-02
2.479e-02	2.489e-02	2.501e-02	2.587e-02	2.754e-02	2.890e-02
2.984e-02	3.071e-02	3.184e-02	3.276e-02	3.341e-02	3.410e-02
3.480e-02	3.501e-02	3.503e-02	3.504e-02	3.506e-02	3.508e-02
3.510e-02	3.512e-02	3.514e-02	4.031e-02	4.964e-02	1.225e-01
2.711e-01	3.390e-01	4.605e-01	3.843e-01	3.670e-01	6.305e-01
5.292e-01	6.588e-01	6.744e-01	8.091e-01	7.988e-01	8.479e-01
8.272e-01	8.357e-01	8.213e-01	8.051e-01	8.062e-01	8.084e-01
8.353e-01	9.812e-01	1.017e+00	1.043e+00	1.127e+00	1.199e+00
1.207e+00	1.377e+00	1.397e+00	1.343e+00	1.299e+00	1.292e+00
1.289e+00	1.292e+00	1.294e+00	1.297e+00	1.295e+00	1.291e+00
1.289e+00	1.285e+00	1.272e+00	1.254e+00	1.235e+00	1.219e+00
1.204e+00	1.186e+00	1.164e+00	1.140e+00	1.113e+00	1.071e+00
1.014e+00	9.490e-01	7.937e-01	7.390e-01	6.911e-01	6.466e-01
6.016e-01	5.557e-01	4.655e-01	4.260e-01	4.003e-01	3.319e-01

tape30:

group radiative capture cross section

5.658e+00	2.721e+00	1.402e+00	1.233e+00	1.083e+00	9.490e-01
8.313e-01	7.255e-01	6.326e-01	5.485e-01	4.745e-01	4.060e-01
3.455e-01	2.941e-01	2.486e-01	2.056e-01	1.698e-01	1.397e-01

1.129e-01	9.072e-02	7.243e-02	5.787e-02	4.635e-02	3.749e-02
3.074e-02	2.553e-02	2.163e-02	1.848e-02	1.618e-02	1.420e-02
1.247e-02	1.111e-02	1.001e-02	9.305e-03	8.755e-03	8.319e-03
8.639e-03	9.378e-03	1.090e-02	1.368e-02	4.981e-02	1.493e-01
1.836e-02	6.142e-02	1.019e-02	1.100e-02	3.065e-02	2.304e-02
1.073e-01	6.814e-02	4.293e-02	6.927e-02	4.838e-02	1.387e-01
3.064e-02	1.208e-02	1.349e-02	1.334e-02	7.319e-03	1.159e-02
4.138e-02	1.718e-02	1.682e-02	1.396e-02	5.733e-03	8.756e-03
8.863e-03	2.234e-02	1.008e-02	1.154e-02	4.710e-03	1.104e-02
1.018e-02	2.741e-03	6.785e-03	6.622e-03	3.500e-03	1.077e-02
2.819e-03	5.786e-03	2.305e-03	7.456e-03	4.775e-03	3.748e-03
4.015e-03	4.297e-03	4.593e-03	4.715e-03	4.525e-03	4.423e-03
4.373e-03	4.353e-03	4.333e-03	4.250e-03	4.094e-03	3.967e-03
3.880e-03	3.803e-03	3.707e-03	3.643e-03	3.623e-03	3.601e-03
3.576e-03	3.562e-03	3.552e-03	3.542e-03	3.542e-03	3.548e-03
3.551e-03	3.538e-03	3.518e-03	3.247e-03	2.869e-03	2.409e-03
2.069e-03	1.961e-03	1.886e-03	1.826e-03	1.785e-03	1.753e-03
1.719e-03	1.678e-03	1.648e-03	1.634e-03	1.626e-03	1.642e-03
1.617e-03	1.562e-03	1.517e-03	1.487e-03	1.472e-03	1.462e-03
1.438e-03	1.398e-03	1.379e-03	1.361e-03	1.342e-03	1.292e-03
1.214e-03	1.091e-03	9.229e-04	8.055e-04	7.078e-04	5.919e-04
4.966e-04	4.443e-04	3.987e-04	3.623e-04	3.316e-04	3.081e-04
2.952e-04	2.786e-04	2.553e-04	2.337e-04	2.139e-04	1.962e-04
1.807e-04	1.678e-04	1.584e-04	1.537e-04	1.566e-04	1.713e-04
2.052e-04	2.477e-04	4.448e-04	6.031e-04	8.032e-04	1.083e-03
1.458e-03	1.972e-03	4.213e-03	5.962e-03	7.162e-03	8.322e-03

tape31:

group total (n,2n) reaction cross section

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.

[illegible]

tape32:
group (n,3n) reaction cross section

[illegible]

0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	0.
0.	0.	0.	0.	0.	2.050e-06

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