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ON EVALUATED NUCLEAR DATA FOR BETA-DELAYED GAMMA RAYS FOLLOWING FISSION OF SPECIAL NUCLEAR MATERIALS

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ABSTRACT

In this paper, a new type of information available in ENDF is discussed. During a consistency check of the evaluated nuclear data library ENDF/B-VII.0 performed at the Nuclear Data Subdivision of the Institute for Advanced Studies, the size of the files for some materials drew the attention of one of the authors. Almost 94 % of all available information for these special nuclear materials is used to represent the beta-delayed gamma rays following fission. This is the first time this information is included in an ENDF version.

1. INTRODUCTION

Brazil is committed – as per the Federal Constitution and the Nuclear Non-Proliferation Treaty – to a strictly peaceful use of nuclear energy. However, Brazil also asserts its strategic need to develop and master nuclear technology. According to the current Brazilian National Strategy of Defense [1], the nuclear sector is of strategic value.

So, the necessary connection between prevention of terrorist acts and measures of chemical, biological and nuclear defense requires an effective control of the spread of nuclear materials.

The Evaluated Nuclear Data Files (ENDFs) have been developed to store the nuclear data mainly required for transport calculations with neutrons and gammas and they contain information on neutron, proton, deuteron, tritium and photonuclear reactions, angular and energy distributions etc. Each time a new ENDF version is released, several verification steps are performed at the Nuclear Data Subdivision of the Institute for Advanced Studies.

During a consistency check of the ENDF/B-VII.0 [2], the size (in Megabytes) of the files for some materials, presented in Table 1, drew the attention of one of the authors. As can be seen from the table, the file sizes for the materials ${}^{235}_{92}U$ and ${}^{239}_{94}Pu$ are much larger than the others and arouse a certain curiosity.

Then, some general information was extracted from the ENDF/B-VII.0 files for these two materials and is shown in Table 2. The MAT number is a material identifier and the MF and MT numbers represent information and reaction types, respectively.

It is known that in MF = 1 general information is presented for the materials and that in MF = 12 multiplicities are available for the production of gamma rays, but MT = 460 was not, until the most recent version of ENDF/B was released, a number used for reaction. Thus, considering that these combinations of MFs/MT numbers use almost 94 % of the space used to store all available information for both materials and they were unknown, even for the most experienced researcher of the Subdivision, further investigation needed to be done to obtain more information about these combinations of MFs/MT.

Table 1. Size of files for some ENDF/B-VII.0 materials.

Material	Size (MB)	Material	Size (MB)
$^{233}_{92}U$	2.1	$^{239}_{94}Pu$	22.4
$^{234}_{92}U$	2.1	$^{240}_{94}Pu$	0.2
$^{235}_{92}U$	23.3	$^{241}_{94}Pu$	0.3
$^{238}_{92}U$	1.9	$^{242}_{94}Pu$	0.2

Table 2. Some $^{235}_{92}U$ and $^{239}_{94}Pu$ general information.

Material	MAT number	MF/MT	# of records	Total # of records	%	# of γ lines
$^{235}_{92}U$	9228	1/460	253,745	283,877	93.94	3,262
		12/460	12,928			
$^{239}_{94}Pu$	9437	1/460	244,001	272,966	93.94	3,129
		12/460	12,413			

2. EVALUATED NUCLEAR DATA FOR BETA-DELAYED GAMMA RAYS

An important point to bear in mind is that no mention of these combinations of MFs/MT have been done both in Appendix A, "ENDF-6 format, abbreviations", of the ENDF/B-VII.0 manual and in Appendices B, "Definition of reaction types", and G, "Maximum dimensions of ENDF parameters", of the previous version, ENDF/B-VI.8 [3]. However, in Chapter III, "Neutron reaction sublibrary", Section C.3, "Delayed photons", of the ENDF/B-VII.0 manual brief descriptions of these combinations are given.

The detection of kilogram quantities of fissionable material in cargo containers became a major issue for world security [4]. Both radiations from prompt gamma rays and from delayed gamma rays can be detected as a signature indicating the presence of a special nuclear material. Thus, the great interest in the development of active systems to detect this type of material encouraged the Monte Carlo neutron-photon transport community to improve the representation of photon- and neutron-induced fission processes [2].

Therefore, it was included with this MT, for the first time in an ENDF version, data for delayed gamma rays from beta decay of neutron-induced fission. The data were estimated using Monte Carlo models for the materials (reactions) $^{235}_{92}U(n_{th},f)$ and $^{239}_{94}Pu(n_{th},f)$ [5]. Time distributions for delayed photons are stored in MF = 1, MT = 460 and photon yields in MF = 12, MT = 460. A detailed description of the formats and procedures used to represent that data can be found in Ref. [6].

A basic quantity of interest to Monte Carlo users is the delayed γ-ray source function s_γ [6],

$$s_\gamma(E, E_\gamma, t) = \frac{d^2 n_\gamma}{dt\, dE_\gamma}(E, E_\gamma, t), \tag{1}$$

where: E is the energy of the fission-inducing projectile; E_γ is the energy of the emitted photon; t is the time following fission at which the photon is emitted; and $\frac{d^2 n_\gamma}{dt\, dE_\gamma}$ is the number of photons emitted per fission per second per eV. This function describes the production of photons following spontaneous or induced fission. In Fig. 1 some processes leading to emission of beta-delayed photons are reproduced [6].

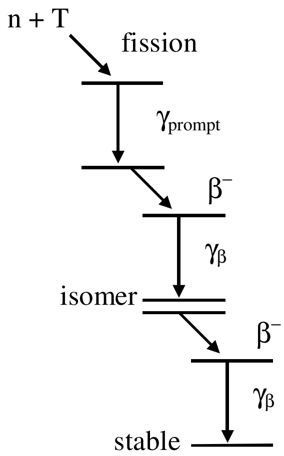

Figure 1. Some processes producing beta-delayed photons. The contributions to the source function s_γ are labeled γ_β.

Although s_γ can be represented in two different ways, the discrete and continuous representations, only the discrete representation was employed for materials ${}^{235}_{92}U$ and ${}^{239}_{94}Pu$ in ENDF/B-VII.0. In this representation [6],

$$s_\gamma(E, E_\gamma, t) = \sum_{i=1}^{NG} \delta(E_\gamma - E_i)\, y_i(E)\, T_i(t), \tag{2}$$

where: $\delta(E_\gamma - E_i)$ represents the Dirac's delta distribution; $y_i(E)$ is the $i^{\underline{th}}$ photon's multiplicity; $T_i(t)$ represents the time dependence of the $i^{\underline{th}}$ photon's multiplicity; and NG is the number of discrete photons or γ lines. Their values for ${}^{235}_{92}U$ and ${}^{239}_{94}Pu$ are 3,262 and 3,129, respectively, as shown in Table 1.

3. FINAL COMMENTS

The number of records used to represent the beta-delayed gamma-rays data from neutron-induced fission for $^{235}_{92}U$ and $^{239}_{94}Pu$ in ENDF/B-VII.0 is almost 94 % of total number of records used in the evaluations. Special care is necessary to handle computationally this large amount of data.

With the availability of these new evaluated nuclear data, some attitudes need to be taken to ensure that the Subdivision continues to provide services [7] of advisory / consultant nature in this matter. First of all, these new data, as well as their representation, need to be studied deeply. Next, it will be important to verify if the computer programs that currently prepare [8] the nuclear data for computer programs that use the Monte Carlo method [9] are still able to manage these new data. Subsequently, the entire detection procedure needs to be studied with the Monte Carlo method, namely, neutron transport in cargo containers up to 6 m or 12 m long, 2.5 m wide and 2.5 m high, subsequent fission of clandestine fissionable material that may be hidden, delayed gamma-ray production and transport of these gamma rays to an external detector. Finally, just to give an idea of the dimensions [10] involved in the problem, a sphere of enriched uranium has a critical radius of 8.741 cm and a sphere of plutonium 6.385 cm.

It is expected that this paper may stimulate theoretical and experimental researchers involved with inspection methods based on neutron and gamma radiation to address this national defense problem.

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