Generation of a multigroup gamma production and photon transport library for STREAM

Kyeongwon Kim, Matthieu Lemaire, Nhan Nguyen Trong Mai, Wonkyeong Kim, Deokjung Lee*
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology 50 UNIST-gil, Ulsan 44919, Republic of Korea
**Corresponding author: deokjung@unist.ac.kr*

1. Introduction

STREAM is a deterministic neutron transport analysis code developed by the Computational Reactor Physics and Experiment Laboratory (CORE) at the Ulsan National Institute Science and Technology (UNIST) [1]. Ongoing development efforts include the generation of a multigroup gamma library for STREAM and the implementation of a photon transport capability by solving the multigroup photon transport equation.

The procedure to generate a multigroup gamma library with the nuclear data processing code NJOY2016 is first detailed in this paper. A preliminary comparison study of the STREAM multigroup gamma against the MCS continuous energy gamma library is then performed for ^{235}U (production of gamma photons through neutron capture, fission, inelastic and nonelastic scattering) and photo atomic cross sections (for photoelectric absorption, coherent and incoherent scattering, and pair production).

2. Methods and Results

A multigroup gamma library based on the library ENDF/B-VII.1 has been generated for STREAM with the nuclear data processing code NJOY2016 [2]. The library includes the gamma production cross section matrices for 425 nuclei processed at 7 different temperatures (293.6K, 600K, 900K, 1200K, 1500K, 1800K, 2100K) and photo atomic cross sections and matrices for 100 elements (Z=1 to Z=100).

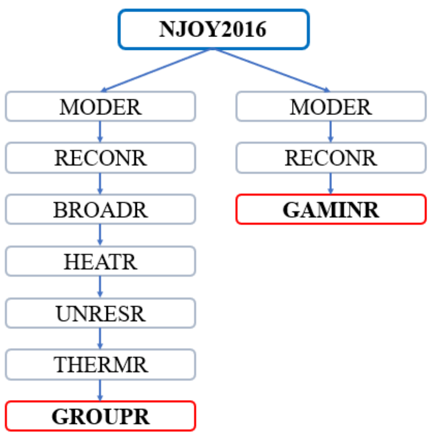

Fig. 1. Workflow of gamma library generation with NJOY2016 [2]

2.1. Neutron induced gamma production

The neutron induced gamma production cross section matrices are output by the GROUPR module of NJOY. The GROUPR module can generates multigroup cross sections, group to group scattering matrices and gamma production matrices [2]. The generation workflow of NJOY is shown in Fig. 1. 72 neutron energy groups and 18 photon energy groups are adopted for the multigroup library. The structure of the 18 photon energy groups is given in Table I. A standard light water reactor spectrum is used in GROUPR for the cross section weighting. The spectrum is shown in Fig. 2.

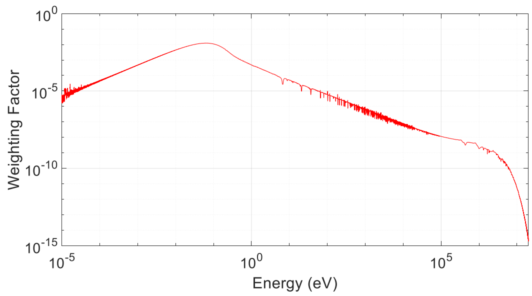

Fig. 2. Standard light water reactor spectrum

For some nuclei and/or neutron reactions, NJOY does not output the neutron induced gamma production cross section matrix directly, but outputs separately the neutron cross section, the gamma yield and the outgoing gamma spectrum. Therefore, the neutron induced gamma production cross section matrix is calculated from this data as in Eq. (1):

$$\sigma_{gp,n\to p} = \chi_{n\to p} Y_n \sigma_n \tag{1}$$

where $\sigma_{gp,n\to p}$ is the matrix element from neutron energy group *n* to photon energy group *p*, $\chi_{n\to p}$ is the spectrum from neutron energy group *n* to photon energy group *p*, Y_n is the photon yield for the neutron energy group *n*, and σ_n is the neutron cross section.

The generated cross section matrices include the nonelastic production matrix (MF16, MT3), the inelastic production matrix (MF16, MT4 and MT50-91), the fission production matrix (MF16, MT18-21 and MT38), the capture production matrix (MF16, MT102) [3] and a “other production” matrix which represents all the other, rarer neutron reactions that can produce gamma photons, such as (n, 2n) and (n, pα). The total matrix is simplex calculated as the sum of the partial matrices as in Eq. (2) [4].

$$\sigma^{total}_{gp,n\to p} = \sigma^{capture}_{gp,n\to p} + \sigma^{fission}_{gp,n\to p} + \sigma^{inelastic}_{gp,n\to p} + \sigma^{nonelastic}_{gp,n\to p} + \sigma^{others}_{gp,n\to p} \qquad (2)$$

Table I: Energy structure of the 18 photon groups

	18 photon energy groups	
	Lower bound (eV)	Upper bound (eV)
1	8.00E+06	1.00E+07
2	6.50E+06	8.00E+06
3	5.00E+06	6.50E+06
4	4.00E+06	5.00E+06
5	3.00E+06	4.00E+06
6	2.50E+06	3.00E+06
7	2.00E+06	2.50E+06
8	1.67E+06	2.00E+06
9	1.33E+06	1.67E+06
10	1.00E+06	1.33E+06
11	8.00E+05	1.00E+06
12	6.00E+05	8.00E+05
13	4.00E+05	6.00E+05
14	3.00E+05	4.00E+05
15	2.00E+05	3.00E+05
16	1.00E+05	2.00E+05
17	5.00E+04	1.00E+05
18	1.00E+03	5.00E+04

2.2. Photo atomic cross sections

The photo atomic cross sections and scattering matrices are output by the GAMINR module of NJOY. The GAMINR module calculates multigroup photo atomic cross section and group to group photon scattering matrices with the workflow shown in Fig. 1 [2]. Like for GROUPR, the same 18 photon energy groups are employed in GAMINR.

The generated photo atomic data include the total cross section (MF23, MT501), the coherent scattering cross section (MF23, MT502), the incoherent scattering cross section (MF23, MT504), the pair production scattering cross section (MF23, MT516), the photoelectric absorption cross section (MF23, MT522), the heat production cross section (MF23, MT525), the coherent scattering matrix (MF26, MT502), the incoherent scattering matrix (MF26, MT504) and the pair production scattering matrix (MF26, MT516) [3].

2.3. Comparison of multigroup vs continuous energy libraries for ^{235}U

The multigroup gamma production cross sections of ^{235}U at 900K for capture, fission, inelastic, nonelastic, are compared against the continuous energy cross sections of MCS library for illustration in Figs. 3-4. The multigroup nonelastic cross section seems inaccurate at the lower energy group. The capture and fission cross sections are suddenly cut off at the higher energy group. This is because any photon that cannot be assigned to a particular level or particle distribution are given in a special called the nonelastic summation reaction [3]. Therefore, the average of the group wise cross section is lower when some part of continuous energy has some values and others have zeros at higher and lower energy. Similarly, due to cross section weighting factor, there are some differences in the lowest energy group. Globally, the multigroup cross sections follow well the continuous energy cross sections.

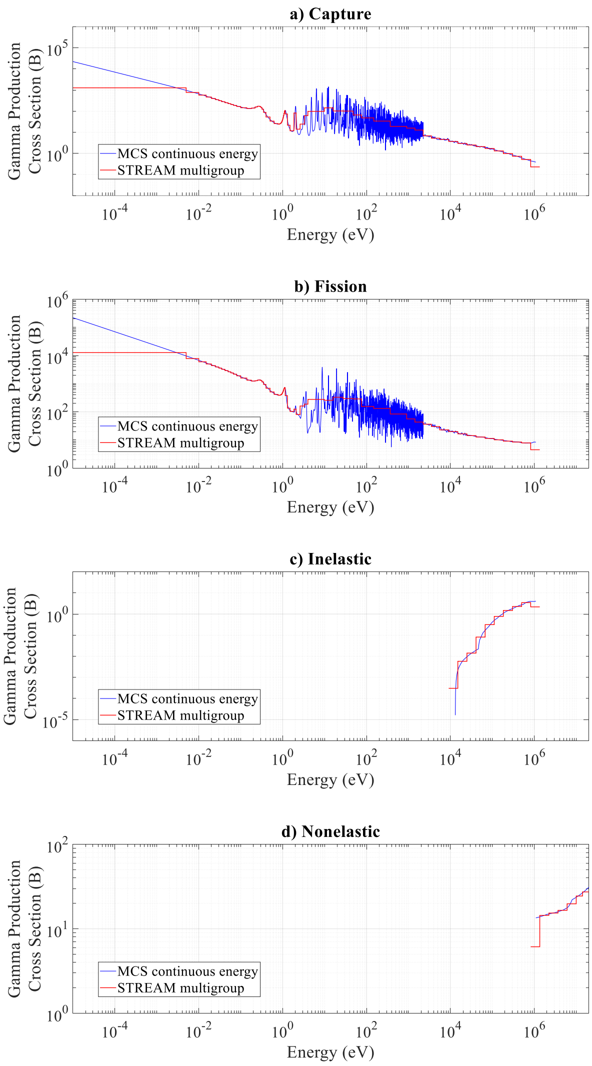

Fig. 3. Comparison of gamma production cross section of ^{235}U for each reaction at 900K. a) capture b) fission c) inelastic d) nonelastic

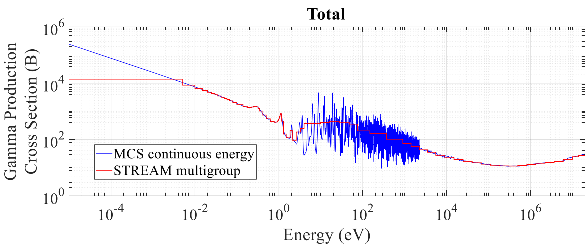

Fig. 4. Comparison of total gamma production cross section of ^{235}U at 900K.

The multigroup photo atomic cross sections of U for photoelectric absorption, incoherent scattering, coherent scattering and pair production are compared with the continuous energy cross sections from MCS library in Figs. 5-6. Small differences can be observed in the lowest energy group due to the cross section weighting.

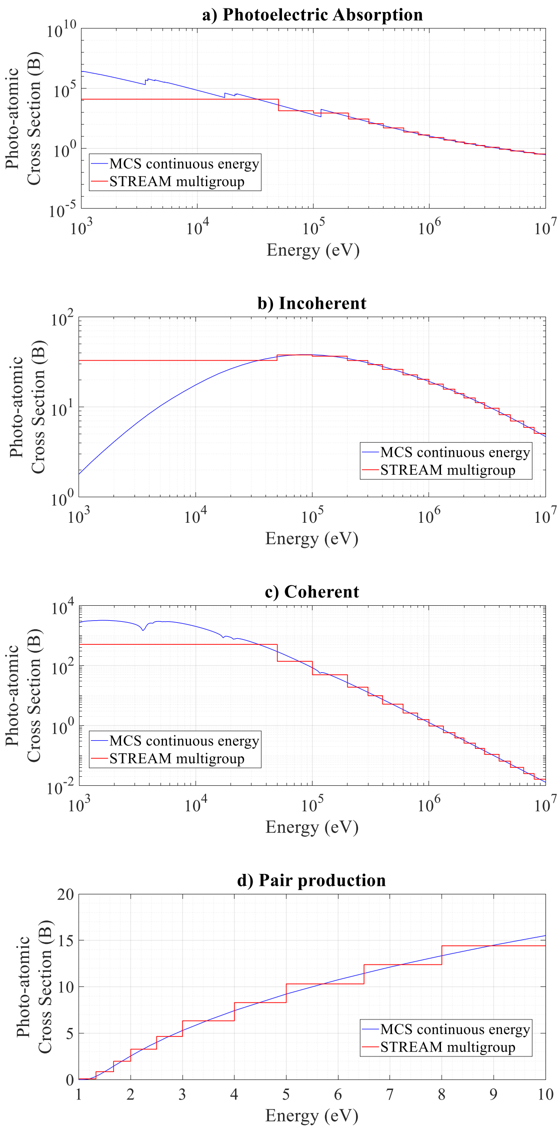

Fig. 5. Comparison of photo-atomic cross section of U for each reaction. a) photoelectric-absorption b) incoherent c) coherent d) pair production

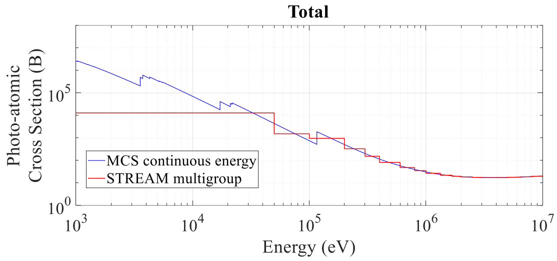

Fig. 6. Comparison of total photo-atomic cross section of U.

3. Conclusions

A multigroup library for gamma photon production from neutron interactions and for photon transport has been generated for the STREAM deterministic code. A preliminary comparison study is multigroup and continuous-energy data is shown for ^{235}U and shows good agreement between the two data types. Future work will involve solving the photon transport equation in STREAM using the generated multigroup gamma library and proceed to verification studies of STREAM results against reference MCS calculations for several photon benchmarks.

Acknowledgement

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korea government (MSIT). (No. NRF-2019M2D2A1A03058371).

REFERENCES

[1] Sooyoung Choi, et. al., Recent Development Status of Neutron Transport Code STREAM, Korean Nuclear Society, May 23-24, 2019, Jeju, Korea
[2] R. E. MacFarlane, D. W. Muir, R. M Boicourt, A. C. Kahler and J. L. Conlin, The NJOY Nuclear Data Processing System, Version 2016, Los Alamos National Laboratory, 2016
[3] A. Trkov, M. Herman and D. A. Brown, ENDF-6 Formats Manual: Data Formats and Procedures for the Evaluated Nuclear Data Files ENDF/B-VI and ENDF/B-VII, CSEWG Document ENDF-102 Report, BNL-903656-2009, Rev.2
[4] Kang-Seog Kim and Ser Gi Hong, Gamma transport and diffusion calculation capability coupled with neutron transport simulation in KARMA 1.2, Annals of Nuclear Energy, Vol.57, pp.59-67, 2013