Development of multi-group neutron activation cross-section library from JENDL/AD-2017

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JENDL Activation Cross Section File for Nuclear Decommissioning 2017 (JENDL/AD-2017) was released in 2018. Then a multi-group neutron activation cross-section library (MAXS/AD-2017) with the same format as MAXS-2015 by Dr. Okumura has been developed from JENDL/AD-2017 with PREPRO 2018 for activation calculations in nuclear facility decommissioning. MAXS/AD-2017 will be converted to ORIGEN libraries and be tested with the JPDR decommissioning data. In future MAXS/AD-2017 will be released.

1. Introduction

JENDL Activation Cross Section File for Nuclear Decommissioning 2017 (JENDL/AD-2017) [1] was released in 2018. This file includes the data of neutron-induced nuclear reactions for 311 nuclides from 10^{-5} eV to 20 MeV. Dr. Okumura et al. developed a multi-group neutron activation cross-section library (MAXS2015) based on the nuclear data libraries JENDL-4.0 and JEFF-3.0/A for activation calculations in nuclear facility decommissioning [2]. A multi-group neutron activation cross-section library (MAXS/AD-2017) with the same format as MAXS-2015 has been developed from JENDL/AD-2017 in order to make it possible to use the new JENDL file for activation calculations in nuclear facility decommissioning.

2. How to make MAXS/AD-2017

JENDL/AD-2017 includes total production cross sections (MF3) of radioactive and stable nuclides, branching ratios (MF9) and partial production cross sections (MF10) for the ground and isomer states of nuclides. JENDL/AD-2017 has the following four versions;

- MF3, MF9 and MF10 at 0 K,
- MF3, MF9 and MF10 at 293.6 K,
- MF3 and MF10 at 0 K (for NJOY processing),
- MF3 and MF10 at 293.6 K (for NJOY processing).

MAXS-2015 was produced with the NJOY2012 [3] code. However it was found that GENDF files produced with the groupr module in NJOY2012 did not include production cross sections to isomer states. Then the PREPRO 2018 [4] code was adopted for producing a group-wise file of JENDL/AD-2017 (MF3, MF9 and MF10 at 0 K). The following modules in PREPRO 2018 were used; ENDF2C, LINEAR, RECENT, SIGMA1, ACTIVATE, FIXUP, DICTIN, GROUPIE. The calculation conditions are as follows;

- Temperature : 300 K,

- Group structure : 199 groups (VITAMIN-B6),
- Weighting spectrum : Maxwell + 1/E + Fission,
- Infinite dilution cross section.

The produced group-wise file of JENDL/AD-2017 was converted to MAXS/AD-2017 of the MAXS format [2] with a small program. Figure 1 shows the data of ^{59}Co in MAXS/AD-2017 as an example. Figure 2 plots the capture cross section (red line) of ^{59}Co in MAXS/AD-2017 with the continuous energy one (blue line), where the red line represents the blue line well.

The following issues were pointed out in this processing.

- No information of decay data (MF8) in the capture reaction of ^{187}W and ^{193}Os → Add
- The MT number of the (n,t) reaction of ^{6}Li is changed from 105 to 107 for ORIGEN-S because ORIGEN-S cannot treat the (n,t) reaction.
- The MAXS format includes no data for the (n,n') reaction→ MAXS/AD-2017 includes the data for the (n,n') reaction, though ORIGEN-S cannot treat the (n,n') reaction.

A similar procedure for a DCHAIN-SP library was also established, and was provided to the PHITS group. Users can use the DCHAIN-SP library of JENDL/AD-2017 in the latest PHITS (PHITS3.16).

3. Summary

A multi-group neutron activation cross-section library (MAXS/AD-2017) with the MAXS format was developed from JENDL/AD-2017 for activation calculations in nuclear facility decommissioning. Next MAXS/AD-2017 will be converted to ORIGEN libraries and be tested with the JPDR decommissioning data [5]. Then MAXS/AD-2017 will be released.

4. References

[1] https://wwwndc.jaea.go.jp/ftpnd/jendl/jendl-ad-2017.html

[2] K. Okumura, K. Kojima, K. Tanaka, "Development of multi-group neutron activation cross-section library for decommissioning of nuclear facilities," Proc. of 2014 Symposium on Nuclear Data, p. 43, JAEA-Conf 2015-003(2016).

[3] R. E. MacFarlane, D. W. Muir, R. M. Boicourt, A. C. Kahler, "The NJOY Nuclear Data Processing System, Version 2012," LA-UR-12-27079, Los Alamos National Laboratory (2012).

[4] https://www-nds.iaea.org/public/endf/prepro2018/

[5] N. P. Kocherov (Ed.), "International benchmark calculations of radioactive inventory for fission reactor decommissioning", INDC(NDS)-355 (1996).

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#MAXS-xs Library
# Nuclide ID & Name
270590
Co059
# background XS (sigz<0:effective XS)
 1.000000E+10
# Temperature (K)
 3.000000E+02
#Number of Energy groups (NGN)
199
#Number of Reaction Types (NMT)
12
#
# No. Energy       MT numbers
                    4             16            22            28            32            41            102           103           104           105           107           112
#                   inel  Co059   2n    Co058   na    Mn055   np    Fe058   nd    Fe057   2np   Fe057   g     Co060   p     Fe059   d     Fe058   t     Fe057   a     Mn056   pa    Cr055
   1  1.964000E+07  3.152500E-01  7.882728E-01  2.628246E-02  1.512986E-01  1.256140E-04  3.566710E-06  5.124860E-04  3.529885E-02  2.690402E-02  1.156921E-03  1.648094E-02  3.545590E-06
   2  1.733200E+07  3.673397E-01  7.809816E-01  1.699300E-02  1.299114E-01  2.515740E-07  0.000000E+00  6.795890E-04  3.812621E-02  2.143026E-02  5.896660E-04  2.089150E-02  1.429740E-07
   3  1.690500E+07  3.890478E-01  7.732319E-01  1.426557E-02  1.234424E-01  1.600430E-08  0.000000E+00  7.415650E-04  3.930724E-02  1.954275E-02  4.555710E-04  2.226716E-02  5.172710E-08
   4  1.648700E+07  4.253916E-01  7.582623E-01  1.012589E-02  1.132496E-01  8.194510E-09  0.000000E+00  8.278450E-04  4.112929E-02  1.694981E-02  3.063550E-04  2.451590E-02  1.102070E-08
   5  1.568300E+07  4.820551E-01  7.293188E-01  5.720482E-03  1.003492E-01  4.776000E-10  0.000000E+00  9.315310E-04  4.374212E-02  1.378198E-02  1.617830E-04  2.679604E-02  8.841500E-10
   6  1.491800E+07  5.337580E-01  6.982927E-01  3.159143E-03  9.045664E-02  0.000000E+00  0.000000E+00  9.896230E-04  4.610335E-02  1.162698E-02  1.021890E-04  2.791561E-02  1.284200E-10
   7  1.455000E+07  5.748928E-01  6.695753E-01  1.993162E-03  8.511011E-02  0.000000E+00  0.000000E+00  1.021886E-03  4.771146E-02  1.023066E-02  7.657830E-05  2.822381E-02  9.060600E-11
   8  1.419100E+07  6.217596E-01  6.339462E-01  1.204300E-03  8.112476E-02  0.000000E+00  0.000000E+00  1.047230E-03  4.926534E-02  8.625081E-03  5.944260E-05  2.824301E-02  5.367800E-11
   9  1.384000E+07  6.762964E-01  5.898754E-01  7.151310E-04  7.773346E-02  0.000000E+00  0.000000E+00  1.065554E-03  5.066195E-02  7.215683E-03  4.702250E-05  2.799245E-02  1.768500E-11
  10  1.349900E+07  7.683994E-01  5.122041E-01  3.144430E-04  7.301934E-02  0.000000E+00  0.000000E+00  1.082507E-03  5.280492E-02  5.680091E-03  3.161490E-05  2.724098E-02  2.075600E-14
  11  1.284000E+07  8.716533E-01  4.229124E-01  1.193130E-04  6.911899E-02  0.000000E+00  0.000000E+00  1.089096E-03  5.423351E-02  4.085699E-03  1.818330E-05  2.630491E-02  4.575700E-16
                                                                                   .
                                                                                   .
                                                                                   .
                                                                                   .
 189  1.000000E-01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  2.077658E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 190  7.000000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  2.456301E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 191  5.000000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  2.814737E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 192  4.000000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  3.194631E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 193  3.000000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  3.737702E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 194  2.100000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  4.481709E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 195  1.450000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  5.386065E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 196  1.000000E-02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  6.874970E+01  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 197  5.000000E-03  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  1.002084E+02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 198  2.000000E-03  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  1.660360E+02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 199  5.000000E-04  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  3.544969E+02  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
 200  1.000000E-05  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00  0.000000E+00
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Fig. 1 Example of MAXS/AD-2017 (^{59}Co).

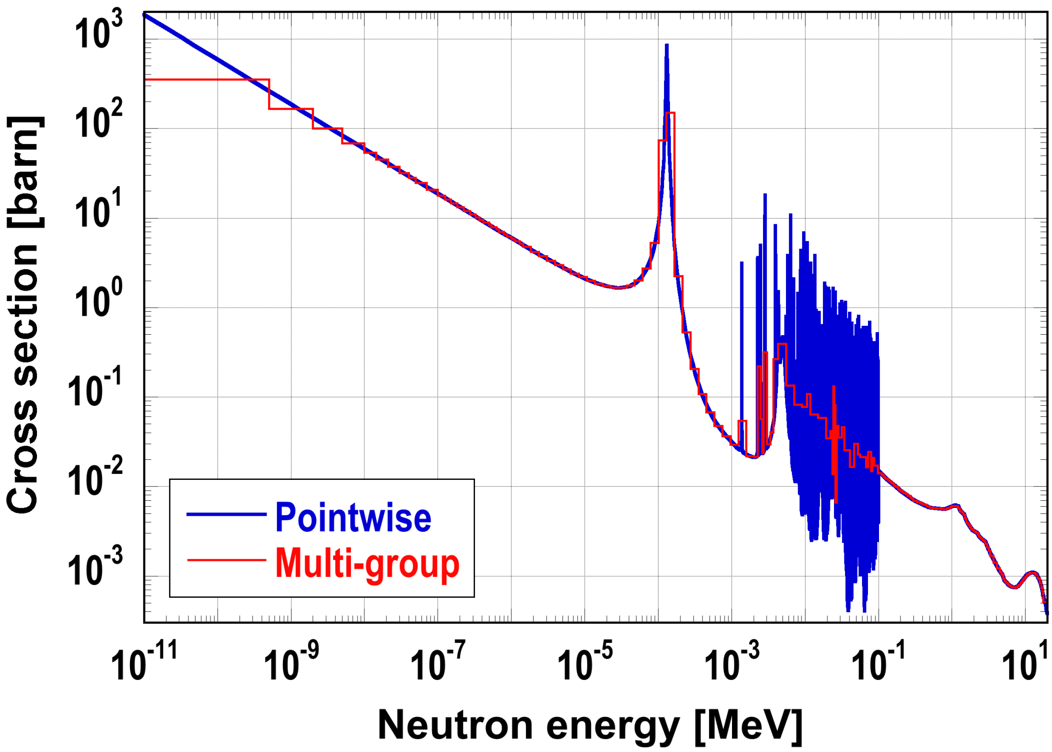

Fig. 2 Capture cross section of ^{59}Co in JENDL/AD-2017 (Red line : MAXS/AD-2017).

JENDL/AD-2017 の多群中性子放射化断面積ライブラリ開発

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