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

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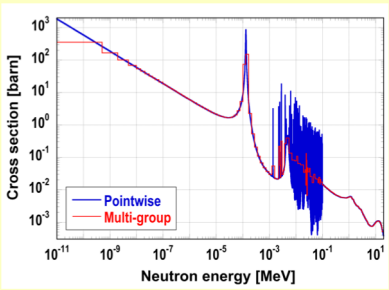
Introduction

- JENDL Activation Cross Section File for Nuclear Decommissioning 2017 (JENDL/AD-2017) [1] was released in 2018.
- This file includes neutron-induced nuclear reaction cross-sections of 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].
- Thus a multi-group neutron activation cross-section library (MAXS/AD-2017) with the same format as MAXS2015 has been developed from JENDL/AD-2017 for validation of JENDL/AD-2017 and activation calculations.

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 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(-2)
  - 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.
- 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.
  - The MAXS format includes no data for the (n,n') reaction→ MAXS files with data for the (n,n') reaction are also produced (Extended MAXS/AD-2017), 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).



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

```
#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)
11
#
# No. Energy      MT numbers
```

#	16 (n,2n)	22 (n,na)	28 (n,np)	32 (n,nd)	41 (n,2np)	102 (n,g)	103 (n,p)	104 (n,d)	105 (n,t)	107 (n,a)	112 (n,pa)
	Co058	Mn055	Fe058	Fe057	Fe057	Co060	Fe059	Fe058	Fe057	Mn056	Cr055
1	1.964000E+07	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	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	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	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
⋮												
197	5.000000E-03	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	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	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

Example of MAXS/AD-2017 (^{59}Co)

Concluding Remarks

- 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. Then MAXS/AD-2017 will be released.

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/