INTERNATIONAL ATOMIC ENERGY AGENCY

# NUCLEAR DATA SERVICES

DOCUMENTATION SERIES OF THE IAEA NUCLEAR DATA SECTION

**IAEA-NDS-051**
Rev. 3

JNDC-FP2
The JNDC Nuclear Data Library of Fission Products
Version 2

Contents and Documentation

Abstract: This library contains nuclear decay data and fission
yield data for 1080 unstable and 147 stable fission product
nuclides, as well as neutron capture cross sections for 166
nuclides. The fission yield data are given for 20 fissioning
systems and include independent yields, cumulative yields and
mass yields. The library is in a free format and is available,
costfree, on magnetic tape or on diskette.

O. Schwerer (ed.)
March 1991

**Citation guidelines:**

JNDC-FP2

The JNDC Nuclear Data Library of Fission

Products

Version 2

1990


    This library contains nuclear decay data and fission yield data for 1080 unstable and 147 stable fission product nuclides, as well as neutron capture cross sections for 166 nuclides. The library is in a free format, i.e. not ENDF/B-format, and has 13301 records. It supersedes earlier versions which had been published in the reports JAERI-H-9715 (Sept. 1981) and JAERI-M-9357 (Feb. 1981).


    The decay data include half-life, branching ratio, and total β- and γ-ray energies released per decay of each unstable nuclide.


    The fission-yield data, consisting of independent yields, cumulative yields and mass yields, are given for the following 20 fissioning systems:


| Nuclide | Incident neutron energies: | | | |
|---|---|---|---|---|
| | thermal | fission spectrum average | 14MeV | spontaneous |
| Th-232 | | + | + | |
| U-233 | + | + | + | |
| U-235 | + | + | + | |
| U-236 | | + | | |
| U-238 | | + | + | |
| Np-237 | | + | | |
| Pu-239 | + | + | + | |
| Pu-240 | | + | | |
| Pu-241 | + | + | | |
| Pu-242 | | + | | |
| Cf-252 | | | | + |

The neutron capture cross sections are given in a 27 group structure and are derived from the JENDL-3 library. Groups 1 to 25 cover the energy range 10 Rev to lO~5eV. The resonance integral, thermal cross section and g-factor are given as group 26 to 28.

Table 2.6.2 Structure of group constants of neutron capture cross section

| Group | ----- Energy Range ------- | | | | Lethargy width |
|---|---|---|---|---|---|
| | Upper | | Lower | | |
| 1 | 10.0 | (MeV) | 6.0653 | (MeV) | 0.500 |
| 2 | 6.0653 | | 3.6788 | | 0.500 |
| 3 | 3.6788 | | 2.2313 | | 0.500 |
| 4 | 2.2313 | | 1.3534 | | 0.500 |
| 5 | 1.3534 | | 0.82085 | | 0.500 |
| 6 | 0.82085 | | 0.38774 | | 0.750 |
| 7 | 0.38774 | | 0.18316 | | 0.750 |
| 8 | 0.18316 | | 0.086517 | | 0.750 |
| 9 | 86.517 | (KeV) | 40.863 | (KeV) | 0 750 |
| 10 | 40.863 | | 19.305 | | 0.750 |
| 11 | 19.305 | | 9.1188 | | 0.750 |
| 12 | 9.1188 | | 4.3074 | | 0.750 |
| 13 | 4.3074 | | 2.0347 | | 0.750 |
| 14 | 2.0347 | | 0.96112 | | 0.750 |
| 15 | 961.12 | (eV) | 454 0 | (eV) | 0.750 |
| 16 | 454.0 | | 214.45 | | 0.750 |
| 17 | 214.45 | | 101.30 | | 0.750 |
| 18 | 101.30 | | 47.851 | | 0.750 |
| 19 | 47.851 | | 22.603 | | 0.750 |
| 20 | 22.603 | | 10.677 | | 0.750 |
| 21 | 10.677 | | 5.0435 | | 0.750 |
| 22 | 5.0435 | | 2.3824 | | 0.750 |
| 23 | 2.3824 | | 1.1254 | | 0.750 |
| 24 | 1.1254 | | 0.53158 | | 0.750 |
| 25 | 0.53158 | | 0.00001 | | 10.881 |
| RI | | > | 0.625 | | — |
| T | | 0.0252 | | | — |

<u>Documentation</u>

The contents of the library - but not the format of the computer file is documented in the two reports given below. The two reports are similar in so far as both contain the full numerical content of the library in a convenient tabular form. In addition, each of them contains some other useful information and graphs.

K. Tasaka et al., JAERI-1320 (September 1990)
H. Ihara (ed.), JAERI-N-89-204 (November 1989)

<u>Format</u> (described in detail on the following pages)

The library consists of <u>one</u> file in a free format which is defined by 11 card types. The sequence of data in the file corresponds to that of the sample listing given on pages 7-11 of the present document.

# Format of JNDC FP Decay Data File Verison 2

```
Card ID
   #1  Title & Date card (18A4)
         Comment card and generation date of file
   #2  Option card (12I6)
          NMAX,NPAR,NFIS,NGS ,NEUT,KSC ,NAN ,IMW ,IERR
           NMAX : Number of nuclides
           NPAR : Maximum number of parent nuclides
           NFlS : Maximum number of fission yields
           NGS  : Maximum number of energy group of neutron
                    capture cross section
           NEUT : Kind of neutron cross sections
           KSC  : Number of capture cross section nuclides
           NAN  : Number of mass number
           IMW  : Words of comment card for library file
                    1 card (18 words)
                    2 cards(36 words)
           IERR : Kind of nuclear data library
                    = 0 nuclear data library
                    = 1 nuclear uncertainty data library
   #3  File title card (18A4)
         Comment of nuclear data library file
   #4  Fission yield code ID card (9I8)
         (NFTYP(I),I=1,NFIS) : Fission yield code
         = 9O2322(232Th(F))    = 9O2323(232Th(HE))
         = 922331(233U(T))     = 922332(233U(F))
         = 922333(233U(HE))    = 922351(235U(T))
         = 922352(235U(F))     = 922353(235U(HE))
         = 922362(236U(F))     = 922382(238U(F))
         = 922383(238U(HE))    = 932372(237Np(F))
         = 942391(239Pu(T))    = 942392(239Pu(F))
         = 942393(239Pu(HE))   = 9424O2(240Pu(F))
         = 942411(241Pu(T))    = 942412(241Pu(F)
         = 942422(242Pu(F))    = 98252O(252Cf(S))
   #5  Mass number card (9I8)
         (MASSN(I),I=1,NAN)    : Mass number
         [Mass number 66 to 172]
   #6  Mass yield card (6E12.5)
         ((ANY(I,J),I=1,NFIS),J=1,NAN)    : Mass yield
   #7  Nuclear data card (I6, E11.4,1X,A1,4E11.4,F5.1,2I2)
         NUCL,RAM,NU,Q,EB,EG,EA,SO,ISGC,NCH
          NUCL : Nuclear ID number
                  NUCL = Z*10000 + A*10 + IS
```

```
                              Z : Atomic number
                              A : Mass number
                              IS: State identifier
                                     0 = Grand state
                                     1 = Meta-stable state(M)
                                     2 = Meta-stable statet(N)
              RAM   : Decay constant or half-life
              NU    : Unit of half-life
                     = blank : Decay constant
                     = S     : Second
                     = M     : Minutes
                     = H     : Hour
                     = D     : Day
                     = Y     : Year
              Q     : Q-value(Mev)
              EB    : Average beta and conversion electron energy(Mev)
              EG    : Average gamma energy(Mev)
              EA    : Average alpha energy(Mev)
              SP    : Spin data(example -2.5 : 5/2-)
              ISGC  : Flag for neutron capture cross section
                       = 0 No capture cross section
                       = 1 Capture cross section
              NCH   : Number of parent nuclides
   #8  Independent fission yield card (6E12.5)
        (YI(I),I=1,NFlS)     : Independent fission yield
   #9  Cumulative fission yield card (6E12.5)
        (YC(I),I=1,NFIS)     : Cumulative fission yield

  #10  Decay types and branching ratios card (6(I3,F8.5,1X))
        (NBIC(I),PBIC(I),I=1,NCH)
        NBIC : Decay type
              NMM   N : State identifier of parent
                         = 0 Grand state
                           1 Meta-stable state(M)
                           2 Meta-stable state(N)
                     MM : Decay type
                         = 1 ƒÃ- -decay
                           2 Isomeric transition
                           3 Neutron capture reaction
                           4 ƒÃ+ and/or electron capture dccay
                           5 ƒ¿-decay
                           6 Delayed neutron emission
        PBlC : Branching ratio
  #11  Neutron capture cross sections card (6E12.5)
        (SC(I),I=1,NGS)      : Neutron capture cross section
                 I = 1 •` 25  Cross sections of fast energy
```

```
                  26        Resonance integral of neutron capture reaction
                  27        Thermal neutron capture cross section for
                            2200 m/sec neutrons
                  28        g-factor


  [Example]

....+....1....+....2....+....3....+....4....+....5....+....6....+....7....+....8   ID  Description
LIBRARY   DATA : COPIED 90/07/26                                                  #1  Title & Date
 1227    6    20    28    1  166  107   18    0                                   #2  Option card
 JNDC FP DECAY FILE V.2 WITH CAPTURE SIG.                                         #3  File title
  902322   902323   922331   922332   922333   922351   922352   922353   922362   00010   #4  Fission yield code
  922382   922383   932372   942391   942392   942393   942402   942411   942412   00020       code ID
  942422   982520                                                          00000030
       66       67       68       69       70       71       72       73       74   00040   #5  Mass number
       75       76       77       78       79       80       81       82       83   00050
 .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
 .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
      156      157      158      159      160      161      162      163      164   00140
      165      166      167      168      169      170      171      172         00000150
 1.21673E-06 1.29096E-04 2.63109E-07 4.50630E-07 6.67707E-04 7.62318E-08    66010   #6  Mass yield
 8.81737E-07 3.01656E-04 7.47461E-07 1.74947E-08 8.50891E-05 1.90870E-07    66020
 1.84292E-07 8.80570E-07 6.33383E-05 5.41350E-06 1.35288E-07 1.84035E-07    66030
 1.89368E-07 2.32189E-09                                                    00066040
 4.19387E-06 2.19860E-04 1.19495E-06 1.79850E-06 1.51954E-03 3.91331E-07    67010
 2.83513E-06 6.74050E-04 1.86837E-06 6.06170E-08 1.39923E-04 3.81017E-07    67020
 3.68048E-07 2.90989E-06 9.99120E-05 8.12143E-06 2.51492E-07 1.29036E-06    67030
 3.78141E-07 1.15984E-08                                                    00067040
 .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
 .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .  .
 1.15693E-08 1.60145E-06 1.66753E-09 2.34121E-09 4.45554E-05 5.91680E-09   170010
 2.04193E-08 3.26341E-05 8.95452E-08 6.82498E-08 6.05987E-05 4.21956E-07   170020
 3.94048E-07 9.75259E-06 4.76571E-05 3.20348E-05 1.52873E-06 1.51942E-05   170030
 4.59435E-05 1.28134E-03                                                    00170040
 5.03625E-09 7.51679E-07 5.55350E-10 7.19219E-10 2.46406E-05 2.81776E-09   171010
 7.16668E-09 1.76804E-05 2.95122E-08 1.91105E-08 3.35607E-05 1.25203E-07   171020
 1.88836E-07 3.25882E-06 3.76282E-05 1.77539E-05 2.87110E-07 5.32270E-06   171030
 2.55167E-05 5.98813E-04                                                    00171040
 2.59343E-09 3.49650E-07 1.85244E-10 2.69983E-10 1.89245E-05 9.70653E-10   172010
 2.04811E-09 1.60920E-05 7.93971E-09 9.98000E-09 2.16190E-05 4.84057E-07   172020
 5.55978E-08 9.65932E-07 1.88655E-05 1.03637E-05 9.58010E-08 1.52001E-06   172030
 1.53329E-05 3.76327E-04                                                    00172040
230660 1.8734E+02    1.6214E+01 5.4830E+00 4.7990E+00 0.0        0.0 0 023066011   #7  Nuclear data
 2.53931E-13 3.85499E-12 2.74046E-17 5.79769E-17 5.31656E-14 1.93964E-1623066013   #8  Independent
 9.72948E-15 1.26557E-13 1.75496E-14 6.77534E-15 1.21107E-11 3.74634E-1623066023       fission yield
 3.86294E-17 3.68364E-16 1.77382E-15 1.08924E-14 2.74879E-15 3.16130E-1523066033
```

```
    4.08893E-15 3.49802E-18                                                23066043
    2.53931E-13 3.85499E-12 2.74046E-17 5.79769E-17 5.31656E-14 1.93964E-1623066014   #9  Cumulative
    9.72948E-15 1.26557E-13 1.75496E-14 6.77534E-15 1.21107E-11 3.74634E-1623066024       fission yield
    3.86294E-17 3.68364E-16 1.77382E-15 1.08924E-14 2.74879E-15 3.16130E-1523066034
    4.08893E-15 3.49802E-18                                                23066044
  240660 2.5941E+00    9.8500E+00 3.8630E+00 1.6930E+00 0.0          0.0 0 124066011
     1 1.00000                                                        24066012   #10  Decay types and
    4.69088E-09 9.89976E-08 1.05956E-11 1.80001E-11 1.54000E-08 2.10951E-1124066013       branching ratios
    4.88771E-10 1.93009E-08 6.47975E-10 9.55051E-11 1.56014E-07 2.97000E-1124066023
    1.05034E-11 6.95207E-11 7.55040E-10 1.35012E-09 1.16030E-10 1.27991E-1024066033
    2.22999E-10 6.51650E-14                                                24066043
    4.69113E-09 9.90014E-08 1.05956E-11 1.80002E-11 1.54001E-08 2.10953E-1124066014
    4.88781E-10 1.93010E-08 6.47993E-10 9.55119E-11 1.56026E-07 2.97004E-1124066024
    1.05034E-11 6.95211E-11 7.55042E-10 1.35013E-09 1.16033E-10 1.27994E-1024066034
    2.23003E-10 6.51685E-14                                                24066044
    . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . .
    . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . .
    . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . .
  340740 0.0          0.0          0.0          0.0          0.0          0.0 1 134074011
     1 0.34900                                                        34074012
    5.93826E-18 1.28018E-15 3.67871E-14 2.23840E-13 1.91002E-12 1.14398E-1634074013
    5.79323E-16 1.49474E-13 2.82312E-16 1.85193E-19 1.80175E-16 2.17128E-1534074023
    1.14361E-14 2.45811E-14 1.07006E-12 3.15588E-15 2.60578E-17 2.45130E-1634074033
    3.82615E-18 1.80396E-16                                                34074043
    2.62539E-13 1.47281E-10 8.42509E-10 1.31938E-09 1.70647E-08 8.91849E-1234074014
    1.27611E-11 3.12032E-09 6.74282E-12 9.99338E-15 8.81037E-12 3.76938E-1134074024
    1.09210E-10 2.43342E-10 7.36539E-09 2.47137E-11 6.10301E-13 6.74250E-1234074034
    8.08859E-14 2.13058E-12                                                34074044
    3.20850E-03 1.00916E-02 1.87879E-02 2.93373E-02 4.29705E-02 7.28995E-0234074015   #11  Neutron capture
    7.59055E-02 9.24381E-02 1.39332E-01 2.22931E-01 3.39608E-01 4.80488E-0134074025       cross sections
    6.54952E-01 2.81436E+00 4.17653E-02 2.10211E+01 4.71615E-01 7.26950E-0134074035
    7.04932E+02 1.31955E+01 5.21381E+00 5.54257E+00 7.15141E+00 9.87754E+0034074045
    2.71939E+01 5.77770E+02 5.19980E+01 1.00520E+00                        34074055
    . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . .
    . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . . .
  ....+....1....+....2....+....3....+....4....+....5....+....6....+....7....+....8
```

-