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INGDB-90

The International Neutron Nuclear Data Base for Geophysics Applications

Assembled by
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Abstract

This document describes the contents of the International Neutron Nuclear Data Base for applications in nuclear geophysics, such as borehole logging and mineral analysis. It contains neutron cross-section data for 19 elements and their isotopes of primary importance in geophysics, plus a data file with neutron spectra of three frequently used neutron sources. The INGDB-90 file is available, costfree, from the IAEA Nuclear Data Section on PC diskettes or on magnetic tape.

July 1991

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Introduction

The Nuclear Data Section of the IAEA convened a Consultants' Meeting on Nuclear Data for Applied Nuclear Geophysics in 1986 [1] in Vienna. The participants of the meeting discussed the data requirements needed for such applications as borehole logging and mineral analysis. They have recommended to the Nuclear Data Section to create an international nuclear data library for this application and worked out a list of elements and reactions to be included into the file.

The present data file was assembled in response to this request and contains all the data requested by the specialists during this meeting.

The library contains data for the following basic elements H, B, C, O, Na, Mg, Al, Si, S, Cl, Ca, Ti, V, Cr, Mn, Fe, Ni, Cu, W, Au and also for P-31 and K-41 which could interfere with the determination of the quantity of the basic elements. In addition to data for natural elements, data for nuclear reactions on particular isotopes of these elements are often needed. Therefore this library contains also data for these isotopes. In total data for 43 materials (both elements and isotopes) are in the database.

2. File 1. Pointwise Cross-Section Data

The File 1 contains neutron nuclear cross-section data for incident neutron energies from 10E-6eV to 20 MeV. For each basic element the total and elastic cross-sections are given and also cross-sections for neutron induced reactions important for element determination. The data were taken from two major libraries recently released by the US National Nuclear Data Centre at Brookhaven (ENDF/B-6 library) [2] and by the Nuclear Data Centre of the Japanese Atomic Energy Research Institute (JAERI) (JENDL-3 library) [3]. The ENDF-6 format [4] of data presentation is used for data extracted from these two libraries. Data for a few reactions were taken from the ADL-90 library [5] which are in the ENDF-5 format (see Table 1). These data are presented here in the original ENDF-5 format [6]. The initial data were processed to pointwise evaluated data which includes linearization, reconstruction of resonance data from the resonance parameters and summation with the background cross-sections. This processing was done using the preprocessing codes LINEAR and RECENT by D.E. Cullen [7]. The resulting pointwise cross section data file is too big (approx. 86 MB) for many applications. We would suggest to use it mainly for creating libraries in special format for input to Monte Carlo codes, using big mainframe computers.

Full contents of File 1 is given in Table 1.

3. File 2. 640 Group Cross-Section Data

For users with mini and/or personal computers File 1 is not convenient because of its size. To satisfy their needs we have further processed all the data in File 1 to group structured format using the code GROUPIE by D.E. Cullen [7]. A so-called extended SANDII 640 group

format of data presentation was used. The structure of the energy groups (lower and upper boundaries) is given in Table 2. This has lead to a drastic reduction of the size of the file and File 2 is only about 3 MB and could easily be handled on personal computers.

4. File 3. Neutron Source Spectrum Data

This is a separate file which contains data on neutron emission spectra from Cf-252, Pu-Be and Am-Be neutron sources. Data for the Cf-252 spectrum were supplied to the IAEA by C. Eisenhauer [8] for the energy region from 1.0E-4 to 1.8E+7 eV and were extrapolated here at NDS to 2.E+7 eV using the ideas of W. Mannhart [9]. The data for Pu-Be and Am-Be were evaluated and converted to a ENDF-like format here at NDS. All the spectra are given in the same extended SANDII groups structure (640 groups). In order to put the data into ENDF format the following notations (not standard for ENDF) were chosen

	MAT	MF	MT
Cf-252	1	3	1
Pu-Be	11	3	1
Am-Be	12	3	1

5. File 4. Data Processing Codes

The user may need to process Files 2 and 3 to some other group structure depending on the input requirements of the application codes. For this purpose we include two codes LINEAR and GROUPIE by D.E. Cullen. To reformat the data to any desired group structure Files 2 and/or 3 should be processed first with the LINEAR code to put the data into linear interpolation form and then with the GROUPIE code. The detailed guidelines how to use the codes are given in the fortran modules of the codes.

File 1 of the data base is distributed on a magnetic tape, Files 2, 3 and 4 can be distributed either on a magnetic tape or on PC diskettes.

We would appreciate receiving your comments on further improvement of this computer file or on errors detected in the process of using it. Our address is:

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A-1400 Vienna, Austria

TABLE I. Contents of the International Nuclear Geophysics Data Base

Element	Data Library	Material No.	Reaction cross-section and uncertainties*	Author + Laboratory	Date of Evaluation
1-H-1	ENDF/B-6	125	1,2,102	G.M. Hale, D.C. Dodder E. Sicilliano, W.B. Wilson, LANL, USA	1990
5-B-10	ENDF/B-6	525	1,2,107,800,801	G.M. Hale, P.G. Young LANL, USA	1990
6-C-nat	ENDF/B-6	600	1,2,51,52,53,102	C.Y. Fu et al., ORNL, USA	1989
8-O-16	JENDL-3	3081	22,51,52,53,54,55, 102,103,104,107	Y. Kanda et al., JNDC Japan	1989
11-Na-23	ENDF/B-6	1125	1,2,16,102,103, 107,33/1,33/2, 33/16,33/102, 33/103,33/107	D.C. Larson, ORNL, USA	1990
12-Mg-nat	ENDF/B-6	1200	1,2,102,103	D.C. Larson, ORNL, USA	1990
12-Mg-24	JENDL-3	3121	103	M. Hatchya et al., DEC, Japan	1989
12-Mg-26	JENDL-3	3123	102	M. Hatchya et al., DEC, Japan	1989
13-Al-27	ENDF/B-6	1325	1,2,102,103,107, 33/1,33/2,33/102, 33/103,33/107	P.G. Young et al., LANL, USA	1990
14-Si-nat	ENDF/B-6	1400	1,2,51,102,103, 33/1.33/2,33/103	D.C. Larson et al., ORNL, USA	1990
14-Si-28	JENDL-3	3141	103	H. Kitazawa et al., JAERI, Japan	1989
14-Si-29	JENDL-3	3142	103	H. Kitazawa et al., JAERI, Japan	1989
14-Si-30	JENDL-3	3143	107	H. Kitazawa et al., JAERI, Japan	1989
15-P-31	JENDL-3	3151	1,2,102,107	H. Nakamura et al., JAERI, Japan	1989
16-S-nat	ENDF/B-6	1600	1,2,102,103	Divadeenam, BNL, USA	1990

Element	Data Library	Material No.	Reaction cross-section and uncertainties*	Author + Laboratory	Date of Evaluation
16-S-34	JENDL-3	3163	1,2,103	H. Nakamura, Fuji E.C., Japan	1989
16-S-36	JENDL-3	3164	102	H. Nakamura, Fuji E.C., Japan	1989
17-Cl-nat	ENDF/B-6	1700	1,2,16,102,103, 107	M.S. Allen et al., GGA, USA	1989
17-Cl-35	ADL-90	1735	16	Grudzevich et al., FEI, USSR	1989
17-Cl-37	ADL-90	1721	102,103,107	Grudzevich et al., FEI, USSR	1989
19-K-41	JENDL-3	3193	1,2,107	H. Nakamura et al., Fuji E.C., Japan	1989
20-Ca-nat	JENDL-3	2000	1,2,102,103	C.Y. Fu et al., ORNL, USA	1971
20-Ca-44	JENDL-3	3204	1,2,103	M. Hatchya, DEC, Japan	1989
20-Ca-48	JENDL-3	3206	1,2,102	M. Hatchya, DEC, Japan	1989
22-Ti-nat	JENDL-3	3220	1,2,102	K. Kobayashi et al., JAERI, Japan	1989
22-Ti-50	JENDL-3	3225	1,2,102	K. Kobayashi et al., JAERI, Japan	1989
23-V-51	JENDL-3	3231	1,2,102,103	T.Watanabe, KHI, Japan	1989
24-Cr-nat	JENDL-3	3240	1,2,102	T. Asami, DEC, Japan	1989
24-Cr-50	JENDL-3	3241	1,2,102	T. Asami, DEC, Japan	1989
24-Cr-52	JENDL-3	3242	1,2,51,52,53,103	T. Asami, DEC, Japan	1987
24-Cr-54	JENDL-3	3244	1,2,107	T. Asami, JAERI, Japan	1987
25-Mn-55	ENDF/B-6	2525	1,2,16,102,107	K. Shibata, JAERI, ORNL, USA/Japan	1990
26-Fe-nat	JENDL-3	3260	1,2,102	S. Iijima et al., JNDC, Japan	1989
26-Fe-54	JENDL-3	3261	1,2,102,107	S. Iijima et al, JNDC, Japan	1989

Element	Data Library	Material No.	Reaction cross-section and uncertainties*	Author + Laboratory	Date of Evaluation
26-Fe-56	JENDL-3	3262	1,2,103	S. Iijima et al., JNDC, Japan	1987
28-Ni-nat	JENDL-3	3280	1,2,102	S. Iijima et al., JNDC, Japan	1989
28-Ni-58	ENDF/B-6	2825	1,2,51,52,53,54, 55,33/1,33/2, 33/51,33/52 33/54,33/55	D.C. Larson et al., USA	1990
29-Cu-nat	JENDL-3	3290	1,2,102	N. Yamamuro et al., NAIG, MAPI, Japan	1989
29-Cu-63	JENDL-3	3291	1,2,51,52,53,54, 55	N. Yamamuro et al., NAIG, MAPI, Japan	1989
29-Cu-65	JENDL-3	3292	1,2,102	N. Yamamuro et al., NAIG, MAPI, Japan	1989
74-W-nat	JENDL-3	3740	1,2,102	T. Watanabe et al., KHI, Japan	1989
74-W-186	JENDL-3	3744	1,2,102	T. Watanabe et al., KHI, Japan	1989
79-Au-197	ENDF/B-6	7925	1,2,51,52,53, 54,55,102	P.G. Young, LANL, USA	1984

* See notes on the following page

* Note: In column 3 the reaction cross-sections are given in the following ENDF/B notations:

1 - total
2 - elastic
16 - n,2n
22 - n,n'a
28 - n,n'p
51 - inelastic scattering to the 1st excited level
52 - " " " " 2nd " "
53 - " " " " 3rd " "
54 - " " " " 4th " "
55 - " " " " 5th " "
102 - n,gamma
103 - n,p
104 - n,d
107 - n,a
800 - n,a (ground state)
801 - n,aγ (first excited state)
33 placed before any of the above figures means covariance information file for reaction cross-section

Laboratory Codes

ORNL	Oak Ridge National Laboratory
JNDC	Japanese Nuclear Data Centre
LANL	Los Alamos National Laboratory
JAERI	Japanese Atomic Energy Research Institute
BNL	Brookhaven National Laboratory
GGA	Gulf General Atomic
NAIG	Nippon Atomic Industry Group
DEC	Data Engineering Co., Japan
KHI	Kawasaki Heavy Industries, Japan
FEI	Obninsk, USSR

TABLE 2. SAND II EXTENDED ENERGY GRID STRUCTURE

UPPER AND LOWER BOUNDARIES OF ENERGY INTERVALS IN EV

1.0000E-04, 1.0500E-04, 1.1000E-04, 1.1500E-04, 1.2000E-04,
1.2750E-04, 1.3500E-04, 1.4250E-04, 1.5000E-04, 1.6000E-04,
1.7000E-04, 1.8000E-04, 1.9000E-04, 2.0000E-04, 2.1000E-04,
2.2000E-04, 2.3000E-04, 2.4000E-04, 2.5500E-04, 2.7000E-04,
2.8000E-04, 3.0000E-04, 3.2000E-04, 3.4000E-04, 3.6000E-04,
3.8000E-04, 4.0000E-04, 4.2500E-04, 4.5000E-04, 4.7500E-04,
5.0000E-04, 5.2500E-04, 5.5000E-04, 5.7500E-04, 6.0000E-04,
6.3000E-04, 6.6000E-04, 6.9000E-04, 7.2000E-04, 7.6000E-04,
8.0000E-04, 8.4000E-04, 8.8000E-04, 9.2000E-04, 9.6000E-04,
1.0000E-03, 1.0500E-03, 1.1000E-03, 1.1500E-03, 1.2000E-03,
1.2750E-03, 1.3500E-03, 1.4250E-03, 1.5000E-03, 1.6000E-03,
1.7000E-03, 1.8000E-03, 1.9000E-03, 2.0000E-03, 2.1000E-03,
2.2000E-03, 2.3000E-03, 2.4000E-03, 2.5500E-03, 2.7000E-03,
2.8000E-03, 3.0000E-03, 3.2000E-03, 3.4000E-03, 3.6000E-03,
3.8000E-03, 4.0000E-03, 4.2500E-03, 4.5000E-03, 4.7500E-03,
5.0000E-03, 5.2500E-03, 5.5000E-03, 5.7500E-03, 6.0000E-03,
6.3000E-03, 6.6000E-03, 6.9000E-03, 7.2000E-03, 7.6000E-03,
8.0000E-03, 8.4000E-03, 8.8000E-03, 9.2000E-03, 9.6000E-03,
1.0000E-02, 1.0500E-02, 1.1000E-02, 1.1500E-02, 1.2000E-02,
1.2750E-02, 1.3500E-02, 1.4250E-02, 1.5000E-02, 1.6000E-02,
1.7000E-02, 1.8000E-02, 1.9000E-02, 2.0000E-02, 2.1000E-02,
2.2000E-02, 2.3000E-02, 2.4000E-02, 2.5500E-02, 2.7000E-02,
2.8000E-02, 3.0000E-02, 3.2000E-02, 3.4000E-02, 3.6000E-02,
3.8000E-02, 4.0000E-02, 4.2500E-02, 4.5000E-02, 4.7500E-02,
5.0000E-02, 5.2500E-02, 5.5000E-02, 5.7500E-02, 6.0000E-02,
6.3000E-02, 6.6000E-02, 6.9000E-02, 7.2000E-02, 7.6000E-02,
8.0000E-02, 8.4000E-02, 8.8000E-02, 9.2000E-02, 9.6000E-02,
1.0000E-01, 1.0500E-01, 1.1000E-01, 1.1500E-01, 1.2000E-01,
1.2750E-01, 1.3500E-01, 1.4250E-01, 1.5000E-01, 1.6000E-01,
1.7000E-01, 1.8000E-01, 1.9000E-01, 2.0000E-01, 2.1000E-01,
2.2000E-01, 2.3000E-01, 2.4000E-01, 2.5500E-01, 2.7000E-01,
2.8000E-01, 3.0000E-01, 3.2000E-01, 3.4000E-01, 3.6000E-01,
3.8000E-01, 4.0000E-01, 4.2500E-01, 4.5000E-01, 4.7500E-01,
5.0000E-01, 5.2500E-01, 5.5000E-01, 5.7500E-01, 6.0000E-01,
6.3000E-01, 6.6000E-01, 6.9000E-01, 7.2000E-01, 7.6000E-01,
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1.0000E+00, 1.0500E+00, 1.1000E+00, 1.1500E+00, 1.2000E+00,
1.2750E+00, 1.3500E+00, 1.4250E+00, 1.5000E+00, 1.6000E+00,
1.7000E+00, 1.8000E+00, 1.9000E+00, 2.0000E+00, 2.1000E+00,
2.2000E+00, 2.3000E+00, 2.4000E+00, 2.5500E+00, 2.7000E+00,
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1.2750E+01, 1.3500E+01, 1.4250E+01, 1.5000E+01, 1.6000E+01,
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2.8000E+01, 3.0000E+01, 3.2000E+01, 3.4000E+01, 3.6000E+01,

3.8000E+01, 4.0000E+01, 4.2500E+01, 4.5000E+01, 4.7500E+01,
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6.3000E+01, 6.6000E+01, 6.9000E+01, 7.2000E+01, 7.6000E+01,
8.0000E+01, 8.4000E+01, 8.8000E+01, 9.2000E+01, 9.6000E+01,
1.0000E+02, 1.0500E+02, 1.1000E+02, 1.1500E+02, 1.2000E+02,
1.2750E+02, 1.3500E+02, 1.4250E+02, 1.5000E+02, 1.6000E+02,
1.7000E+02, 1.8000E+02, 1.9000E+02, 2.0000E+02, 2.1000E+02,
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9.0000E+06, 9.1000E+06, 9.2000E+06, 9.3000E+06, 9.4000E+06,
9.5000E+06, 9.6000E+06, 9.7000E+06, 9.8000E+06, 9.9000E+06,
1.0000E+07, 1.0100E+07, 1.0200E+07, 1.0300E+07, 1.0400E+07,
1.0500E+07, 1.0600E+07, 1.0700E+07, 1.0800E+07, 1.0900E+07,
1.1000E+07, 1.1100E+07, 1.1200E+07, 1.1300E+07, 1.1400E+07,
1.1500E+07, 1.1600E+07, 1.1700E+07, 1.1800E+07, 1.1900E+07,
1.2000E+07, 1.2100E+07, 1.2200E+07, 1.2300E+07, 1.2400E+07,
1.2500E+07, 1.2600E+07, 1.2700E+07, 1.2800E+07, 1.2900E+07,
1.3000E+07, 1.3100E+07, 1.3200E+07, 1.3300E+07, 1.3400E+07,
1.3500E+07, 1.3600E+07, 1.3700E+07, 1.3800E+07, 1.3900E+07,
1.4000E+07, 1.4100E+07, 1.4200E+07, 1.4300E+07, 1.4400E+07,
1.4500E+07, 1.4600E+07, 1.4700E+07, 1.4800E+07, 1.4900E+07,
1.5000E+07, 1.5100E+07, 1.5200E+07, 1.5300E+07, 1.5400E+07,
1.5500E+07, 1.5600E+07, 1.5700E+07, 1.5800E+07, 1.5900E+07,
1.6000E+07, 1.6100E+07, 1.6200E+07, 1.6300E+07, 1.6400E+07,
1.6500E+07, 1.6600E+07, 1.6700E+07, 1.6800E+07, 1.6900E+07,
1.7000E+07, 1.7100E+07, 1.7200E+07, 1.7300E+07, 1.7400E+07,
1.7500E+07, 1.7600E+07, 1.7700E+07, 1.7800E+07, 1.7900E+07,
1.8000E+07, 1.8100E+07, 1.8200E+07, 1.8300E+07, 1.8400E+07,
1.8500E+07, 1.8600E+07, 1.8700E+07, 1.8800E+07, 1.8900E+07,
1.9000E+07, 1.9100E+07, 1.9200E+07, 1.9300E+07, 1.9400E+07,
1.9500E+07, 1.9600E+07, 1.9700E+07, 1.9800E+07, 1.9900E+07,
2.0000E+07.

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