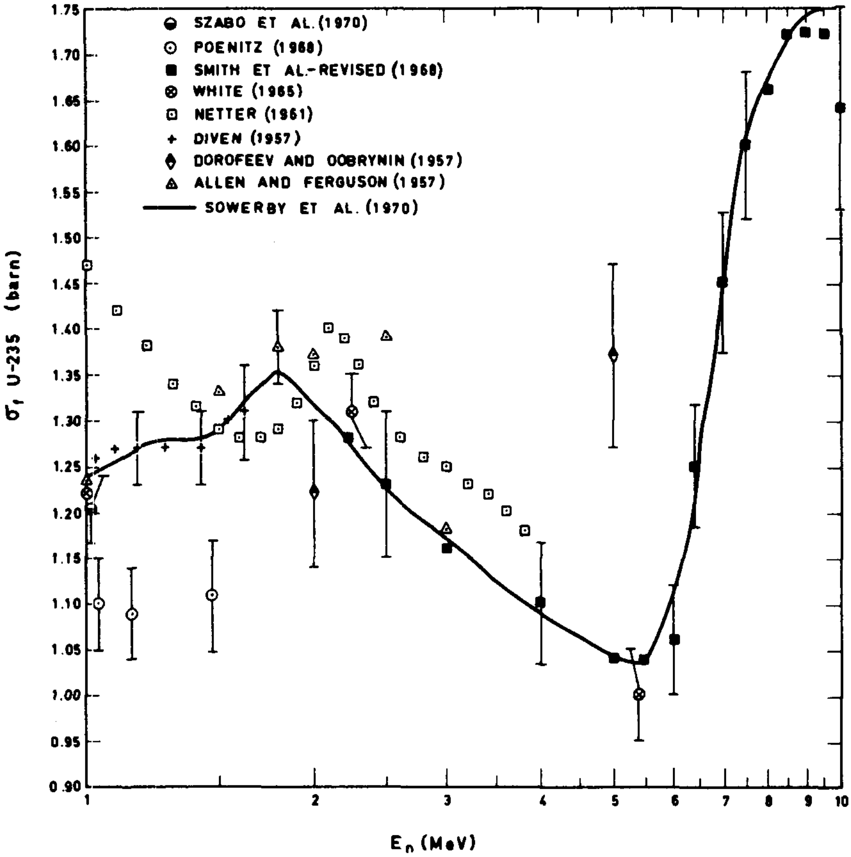

TECHNICAL REPORTS SERIES No.146

Neutron Nuclear Data Evaluation

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1973

NEUTRON NUCLEAR DATA EVALUATION

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NEUTRON NUCLEAR DATA EVALUATION

SUMMARY OF A PANEL ON
NEUTRON NUCLEAR DATA EVALUATION
HELD BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 30 AUGUST TO 3 SEPTEMBER 1971

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1973

NEUTRON NUCLEAR DATA EVALUATION
IAEA, VIENNA, 1973
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FOREWORD

Modern nuclear technologies such as that of fast breeder power reactors require a comprehensive and reliable nuclear data basis to guarantee safe and economic characteristics. Many experimental nuclear physics groups throughout the world respond to these requirements by making neutron nuclear measurements with steadily improving accuracy. These data are systematically compiled by the four world neutron data centres at Brookhaven National Laboratory (USA), at Saclay (France) (NEA), at Obninsk (USSR), and at Vienna (IAEA), and are made available to groups in national laboratories, who convert the data into a form usable in reactor design calculations. This conversion process is called evaluation and comprises the following individual steps:

Critical comparison, selection and averaging of the available experimental data;

Interpolation and extrapolation of experimental data and use of nuclear theory and systematics in the case of gaps and inconsistencies in the experimental information;

Establishment of a computer library of complete, self-consistent and easily interpolable evaluated data sets from which multigroup constants and related quantities can be calculated and then used as direct data input to reactor design and safety calculations.

Upon the recommendation of the International Nuclear Data Committee (INDC), which acts as an advisory body to the Director General of the International Atomic Energy Agency in all matters pertaining to nuclear data, the Agency convened a Panel on Neutron Data Compilation in February 1969. The panel was instrumental in developing the international computer-based system known under the name EXFOR for the systematic compilation and exchange of experimental neutron data and this work led to a considerable improvement in international co-operation in this field.

In response to a similar need for improved international co-ordination of the dispersed evaluation efforts and again upon a recommendation of INDC, the Agency convened a Panel on Neutron Nuclear Data Evaluation in Vienna from 30 August to 3 September 1971. The task of the Panel was to review the methods, quality and present status of neutron nuclear data evaluation and to examine the basic requirements and problems associated with establishing, maintaining, using and exchanging computer-based libraries of evaluated neutron data. Since the Panel's composition reflected the major evaluation activities in the world, a fruitful exchange of experience and ideas was to be expected.

In particular, the Panel gave a detailed review of the status and quality of the existing major libraries, bearing in mind the still unsatisfied needs for evaluated data in IAEA Member States, especially in developing countries. It compared in detail the main computer formats for evaluated data, knowledge of which is a prerequisite for an efficient international exchange of evaluated data.

The present book is a summary of the Panel's work. The individual papers are gathered together in the unpublished document IAEA-153, which is available from the Nuclear Data Section of the Agency.

CONTENTS

1. INTRODUCTION

The Panel on Neutron Nuclear Data Evaluation, whose findings are presented in this Summary, met in Vienna from 30 August to 3 September 1971 and consisted of 25 specialists representing 11 countries, the European Nuclear Energy Agency (ENEA)[1] and the International Atomic Energy Agency. (A list of Panel participants and their affiliations is given in Appendix A.)

A total of twenty-six papers were submitted to the Panel. These have been gathered together in the unpublished document IAEA-153, which is available from the Nuclear Data Section of the Agency. A list of titles and authors is given in Appendix C. The Panel opened with a plenary session which reviewed the evaluation activities, the important evaluation needs and their assessment in IAEA Member States. The members of the Panel were then separated into five parallel subgroups to consider and make recommendations on the topics under Agenda items 2, 3, 4, 5.I and 5.II (the Panel Agenda is given in Appendix B). Their reports were subsequently submitted to the plenary session for discussion and final adoption. The final plenary session of the Panel then addressed itself to discussing international co-operation in, and co-ordination of, neutron data evaluation activities.

The conclusions and recommendations of the Panel are incorporated in the reports of the five subgroups given in Sections 2.2 to 2.6 and may be categorized into two classes: firstly, those directed to the institutes which develop, maintain and/or use evaluated neutron data libraries, and secondly, those directed to the Agency. Though each of the Panel's recommendations and observations should be viewed within the context of the specific subgroup report, for ease of reference a summary of all the recommendations has been extracted from the subgroup reports, and this is given in Section 3.

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[1] Since May 1972 called the OECD Nuclear Energy Agency (NEA).

2. SUMMARY OF PRINCIPAL PAPERS AND SUBGROUP REPORTS

2.1. EVALUATION ACTIVITIES, IMPORTANT EVALUATION NEEDS AND THEIR ASSESSMENT IN IAEA MEMBER STATES

Under this agenda item (Agenda item 1) papers were presented to the Panel describing the national evaluation activities and evaluated neutron data needs in IAEA Member States. The Review Paper by Byer and Schmidt (see Appendix C), which was based on material supplied to the Agency by experts from 23 Member States, introduced the concept of the "neutron nuclear data cycle" in direct analogy to the commercial nuclear fuel cycle. For those states with fast reactor programs and a fully integrated "neutron nuclear data cycle" a major effort is being expended in evaluating the most important cross-sections for the most important materials in the most important energy regions.

The Review Paper pointed out that, in addition to the four major evaluated nuclear data libraries, UKNDL[2], ENDF[3], KEDAK[4] and LLL[5], work is under way in the Soviet Union to establish their own evaluated nuclear data library on the basis of a format developed there, and to develop the necessary computer programs for servicing, handling and using such a library. Also, considerable progress has been made and is being made in overcoming the problem of converting from one of these five formats to another.

For those states not possessing a fully integrated "neutron nuclear data cycle" there is either no activity in the field of neutron data evaluation or evaluations are being performed and focussed on small specialized areas where it is felt that significant contributions may be made.

The Review Paper concluded by pointing out that for those states without a fully integrated "neutron nuclear data cycle" the immediate problem in relation to their needs for evaluated neutron data is not really that of being able to gain access to evaluated nuclear data libraries, since this is possible either bilaterally or through the Agency, which already has one complete library available. The real problems in dealing with evaluated data are related to their inability to handle automatically large evaluated data files because not only are their computer facilities insufficiently large but also the computational effort required to generate their own multigroup cross-section sets is considerable.

This point was further emphasized in the papers by Vertes, and Rastogi and Huria (see Appendix C), which summarized the activities in Hungary and India, respectively, towards using evaluated nuclear data libraries to generate their own multigroup constants for reactor and

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[2] The United Kingdom Nuclear Data Library (United Kingdom).

[3] The Evaluated Nuclear Data File (United States of America).

[4] The Karlsruhe Evaluated Data File (Federal Republic of Germany).

[5] The Lawrence Livermore Laboratory Evaluated Nuclear Data Library (United States of America).

shielding calculations. Rastogi and Huria emphasized that because of the difficulties they experienced in manipulating evaluated nuclear data libraries they are concentrating on the preparation of multigroup cross-section libraries and the modification of existing ones. This therefore assumes a greater importance for their predictions of the physics performance of fast and thermal reactors than the direct use of evaluated nuclear data libraries.

The paper by Story (see Appendix C) described the evaluation work and evaluated data needs in the United Kingdom during the second half of 1971. One of the most interesting features of this paper was the review on the UKNDL and the degree to which it meets the needs of the United Kingdom reactor program from the standpoint of the quality of the evaluated data contained in this library. The Igarasi and Rapeanu papers (see Appendix C) briefly outlined the evaluation activities in Japan and Romania, respectively.

Some preliminary results of a comparative analysis of the evaluated nuclear data libraries UKNDL, KEDAK, ENDF/B-I and ENDF/B-II were presented in the papers by Yiftah, and Ilberg and Yiftah (see Appendix C). This study is essentially aimed at providing at least partial answers to the question: Are the physics parameters of the fast reactors being calculated and designed a function of the specific evaluated data file that is used in the calculations? In other words, would these physics parameters (critical masses, reaction rates, breeding ratios, Doppler coefficients, sodium void coefficients, etc.) remain the same if the calculations used as input a different evaluated data library? Such an analysis is being performed at three different levels, the first being a simple graphical comparison of the same cross-sections for the same nuclides from the four different libraries. The second level represents a comparison of the multigroup cross-section sets obtained from the four libraries using the same techniques and the same averaging fluxes. The final level involves a comparison of the physics parameters of three different systems, critical-assembly-size systems, prototype-size systems and 1000-MW(e)-size systems. These three types of systems will be calculated using the same multigroup codes with the different multigroup sets generated from the four data files as input. So far, this comparative analysis has not yet been extended beyond the first level; however, the results of this study, when completed, should throw some light on the effects of calculating the same reactor system with the four different evaluated nuclear data libraries.

2.2. STATUS AND QUALITY CONTROL OF EVALUATIONS

The report of the subgroup on "Status and Quality Control of Evaluations" consists of three parts, which are given below. The Panel endorsed the findings of this subgroup and considered that the information presented would provide users of evaluated nuclear data libraries with a concise, well-documented, up-to-date and readily accessible summary of the essential features and contents of the main libraries.

Subgroup Report

(Chairman: S. Yiftah)

Part I

Status of evaluated nuclear data libraries (1971)

A. United Kingdom Nuclear Data Library (UKNDL)

Country of origin: United Kingdom of Great Britain and Northern Ireland.

Date of release of latest version: The last general edition was released to the Centre for Neutron Data Compilation (CCDN), Saclay, France, early in 1970. Several additional files have been released in August 1971. The UKNDL is conceived as in continuous evolution.

Number of materials or nuclides: There are complete files for 51 materials and 78 fission product capture cross-section files. A number of files, mainly activation detectors, with one or two cross-sections over a limited energy range also exist. Archival tapes contain more than 80 superseded files.

List of materials or nuclides:

H-1, D-2, T-3, He-3, He-4, Li-6, Li-7, Be-9, B, B-10, B-11, C, N, O, F-19, Na-23, Al-27, Si, Cl, K, Ca, Ti, V, Cr, Mn-55, Fe, Ni, Cu, Ga, Zr, Nb, Mo, Cd, Cd-113, Xe-135, Ta-181, W, Au-197, Pb, Th-232, Pa-233, U-233, U-234, U-235, U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241 and bulk fission products.

For a recent commentary on the status of these files, see the paper by Story (Appendix C).

In addition to the above files there are 78 files of fission product capture cross-sections:

Br-81, Se-82, Kr-(83,84,85,86), Rb-(85,87), Sr-(88,89,90), Y-(89,90,91), Zr-(91,92,93,94,96), Mo-(95,97,98,100), Tc-99, Ru-(101,102,104), Rh-(103,105), Pd-(105,106,107,108), Ag-109, Cd-113, In-115, Sb-125, Te-(128,130), I-(127,129,131,135), Xe-(131,132,133,134,135,136), Cs-(133,134,135,137), Ba-138, La-139, Ce-(140,142), Pr-141, Nd-(143,144,145,146,148,150), Pm-(147,148,148m), Sm-(149,150,151, 152,154), Eu-(153,154,155), Gd-(155,156,157)

A number of files of mainly activation detector cross-sections are also contained in the UKNDL and are listed in Report AEEW-M-824:

Na-23, Mg-24, Al-27, Si-28, P-31, S-32, S-34, Cl-35, Sc-45, Ti-46, Mn-55, Fe-54, Fe-56, Co-59, Ni-58, Cu-63, Cu-65, Zn-64, Y-89, Rh-103, In-115, I-127, Tm-169, Au-197, Th-232, Np-237, Pu-242.

Availability: The library is distributed to OECD Member States through the Centre for Neutron Data Compilation, Saclay, France, and to Australia. A number of files have been made available more widely through the IAEA on request.

Special features: All of the data appear in the UKNDL format as documented in Report AWRE-O-70/63. For some materials more than one file exists but each is distinguished by a different data file number (DFN). The most recent evaluations for U-235, Pu-239 and U-238 were performed simultaneously to take full account of information on the cross-section ratios. Resonance parameter data are not incorporated in the files but are being compiled separately (see Report AEEW-R-622).

Comments: Some of the files in the UKNDL are of French origin. The fission product files were created by merging compilations of Australian and Italian origin. Some files have been translated from the ENDF/B, and in others partial use has been made of data from the SPENG, KEDAK and ENDF/B data files and from other compilations.

In addition to the UKNDL, the GENEX data library contains resonance cross-section data in great detail (120 000 energy points) from 0.4 eV to 25 keV for a few fertile and fissile materials, U-235, U-238, Pu-239, Pu-240 and Pu-241. The data are in binary format (see Report AEEW-R-622).

Main references:
Format: PARKER, K., Rep. AWRE-O-70/63 (1963).
Contents: NORTON, D.S., Rep. AEEW-M-824 (1968).
STORY, J.S. (see Appendix C).
For a general account of the UKNDL see:
STORY, J.S. et al., Third Int. Conf. peaceful Uses atom. Energy (Proc. Conf. Geneva, 1964) 2, UN, New York (1964) 168.

B. Evaluated Nuclear Data File[6] (ENDF)

Country of origin: United States of America. A co-operative effort administered by the National Neutron Cross Section Center (NNCSC) at Brookhaven National Laboratory and approximately 20 laboratories in the United States and Canada.

Date of release of latest version: January 1972.

Number of materials or nuclides: 134.

List of materials or nuclides in the ENDF/B-III:

H-1, D-2, He, He-3, Li-6, Li-7, Be-9, B-10, B-11, C-12, N-14, O-16, Na-23, Mg, Al-27, Si, Cl, K, Ca, V, Cr, Mn-55, Fe, Co-59, Ni, Cu, Cu-63, Cu-65, Kr-83, Zr-95, Nb-93, Nb-95, Mo, Mo-95, Mo-97, Mo-98, Mo-99, Mo-100, Ru-101, Ru-102, Ru-103, Ru-104, Ru-105, Ru-106, Rh-105, Pd-105, Pd-106, Pd-107, Pd-109, Ag-107, Ag-109, Cd-113, I-131, I-135, Xe-131, Xe-133, Xe-135, Cs-133, Cs-135, Cs-137, La-139, Ce-141, Pr-141, Pr-143, Nd-143, Nd-145, Nd-147, Pm-147, Pm-148g, Pm-148m, Pm-149, Pm-151, Sm-147, Sm-148, Sm-149, Sm-150, Sm-151, Sm-152, Sm-153, Eu-151, Eu-153, Eu-154, Eu-155, Eu-156, Eu-157, Gd, Gd-155, Gd-157, Dy-164, Lu-175, Lu-176, Ta-181, Ta-182, W-182, W-183, W-184, W-186, Re-185,

[6] Two libraries are maintained, ENDF/A and ENDF/B. The information given pertains to the ENDF/B library. The ENDF/A library so far merely contains evaluations from other libraries converted to the ENDF format and therefore its status is covered by the reports of other libraries.

Re-187, Au-197, Pb, Th-232, Pa-233, U-233 (4 files), U-234, U-235 (4 files), U-238, Pu-238, Pu-239 (4 files), Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-244

Availability: Distributed by the NNCSC to those countries with which the United States of America has bilateral agreements for the exchange of information concerning evaluated data. At the present time, these agreements include Member States of the European Nuclear Energy Agency, Australia, India, and Israel.

Special features: All data appear in the ENDF format as documented in ENDF-102 (Vols I and II), also catalogued as BNL-50274. Only one evaluation is presented for each material. Data may be presented in a variety of ways, but no redundancy of data exists in the files. The evaluations for U-235, U-238 and Pu-239 were performed simultaneously to attempt compatibility with both microscopic cross-section measurements and cross-section ratios between these nuclides.

Comments: ENDF/B-II generally calculated increasingly low criticality values for Pu-239-fuelled fast assemblies containing increasing concentrations of U-238 and was not considered adequate for a wide range of fast reactor designs.

Main references: No single reference is available but individual reports are listed in Newsletter 72-1, National Neutron Cross Section Center, Brookhaven National Lab., Upton, N.Y.

C. Lawrence Livermore Laboratory Evaluated Nuclear Data Library (LLL)

Country of origin: United States of America.

Date of release of latest version: Numbered editions are released periodically, usually about every six months.

Number of materials or nuclides: 65 isotopes or elements.

List of materials or nuclides:

H-1, D-2, T-3, He-3, He-4, Li, Li-6, Li-7, Be, Be-9, B, B-10, C-12, N-14, O-16, F-19, Na-23, Mg, Al-27, Si, P-31, S-32, Cl, Ar, K, Ca, Sc-45, Ti, Mu, Fe, Fe-54, Fe-56, Fe-58, Ni, Ni-58, Cu, Ga, Nb, Mo, Cd, Sn, Ba, Eu, Gd, Ho-165, Ta-181, W, Au-197, Pb, Th-232, U-233, U-234, U-235, U-236, U-237, U-238, U-239, U-240, Np-237, Pu-238, Pu-239, Pu-240, Pu-241, Am-242, and fission products

Availability: Available through bilateral negotiations.

Special features: Special features are all discussed in Report UCRL-50400, Vol. 4.

Main references: The format of the LLL library and all relevant details are described in Reports UCRL-50400, Vols 1-10.

D. Swedish Evaluated Neutron Data Library (SPENG)

Country of origin: Sweden.

Date of release of latest version: The last complete library was released in September 1970. The SPENG library is, however, in continuous evolution.

Number of materials or nuclides: 32. The number of files, however, is 38 since some nuclides have different data versions.

List of materials or nuclides:

H-1, D-2, He-4, Li-6, Li-7, Be-9, B, B-10, B-11, C, O, F, Na, Al, Si, Cr, Mn, Fe, Ni, Cu, Zr, Mo, Er, Ta, W-186, Au-197, U-235, U-238, Pu-239, Pu-240, Pu-241 and the Pu-239 fission products

Availability: The data are available on request.
Address: Dr. H. Häggblom, AB Atomenergi, Studsvik, Fack, S-611 01, Nykøping, Sweden.

Special features: The format of the library is documented in Report AE-RFN-279 (1967). Part of this report is reproduced in Annex II of the review paper by Byer and Schmidt (see Appendix C). For those nuclides with many resonances, effective group cross-sections are given with the background cross-section and the temperature as parameters.

Comments: Some files or parts of files are from the UKNDL or ENDF/B libraries. The Karlsruhe (Federal Republic of Germany) evaluations have been used to a large extent.

Main references:
Format: NYMAN, K., Rep. AE-RFN-279 (1967).
More information about the data will be published by H. Häggblom.

E. Karlsruhe Evaluated Nuclear Data File (KEDAK)

Country of origin: Federal Republic of Germany.

Date of release of latest version: July 1970.

Number of materials or nuclides: 20.

List of materials or nuclides:

H-1, D-2, He-3, He-4, C-12, N, O-16, Na-23, Al-27, Cr, Fe, Ni, Mo, Cd, U-235, U-238, Pu-239, Pu-240, Pu-241, Pu-242

Availability: The library is distributed to the OECD Centre for Neutron Data Compilation (CCDN), Saclay, France, the National Neutron Cross Section Center, Brookhaven, USA, and the Nuclear Data Section of the IAEA.

Special features: Resonance information is stored as resonance parameters as well as pointwise tabulated cross-sections.

Comments: The next version of the library should be released in mid-1973.

Main references:
Format: WOLL, D. et al., Rep. KFK-880.
Status: HINKELMANN, B. et al., Rep. KFK-1340.

F. Italian Fission Product Library

Country of origin: Italy.

Date of release of latest version: December 1969.

Number of materials or nuclides: All stable nuclei with $32 \leq Z \leq 66$.

List of materials or nuclides:

Ge-(70, 72, 73, 74, 76), As-75, Se-(74, 76, 77, 78, 80, 82), Br-(79, 81), Kr-(78, 80, 82, 83, 84, 86), Rb-(85, 87), Sr-(84, 86, 87, 88), Y-89, Zr-(90, 91, 92, 94, 96), Nb-93, Mo-(92, 94, 95, 96, 97, 98, 100), Ru-(96, 98, 99, 100, 101, 102, 104), Rh-103, Pd-(102, 104, 105, 106, 108, 110), Ag-(107, 109), Cd-(106, 108, 110, 111, 112, 113, 114, 116), In-(113, 115), Sn-(112, 114, 115, 116, 117, 118, 119, 120, 122, 124), Sb-(121, 123), Te-(120, 122, 123, 124, 125, 126, 128, 130), I-127, Xe-(124, 126, 128, 129, 130, 131, 132, 134, 136), Cs-133, Ba-(130, 132, 134, 135, 136, 137, 138), La-(138, 139), Ce-(136, 138, 140, 142), Pr-141, Nd-(142, 143, 144, 145, 146, 148, 150), Sm-(144, 147, 148, 149, 150, 152, 154), Eu-(151, 153), Gd-(152, 154, 155, 156, 157, 158, 160), Tb-159, Dy-(156, 158, 160, 161, 162, 163, 164)

Availability: The library is distributed to OECD Member States through the Centre for Neutron Data Compilation (CCDN), Saclay, France, and has also been made available to the IAEA Nuclear Data Section.

Special features: Only the radiative capture cross-sections over the energy range from 1 keV to 10 MeV have been evaluated and the format of the library is that of the UKNDL.

Main references:
The Newsletter, CCDN-NW/10, Centre for Neutron Data Compilation, Saclay (1969).

G. USSR Evaluated Nuclear Data Library

Country of origin: Union of Soviet Socialist Republics.

Date of release of latest version: The input of data into the library, mainly from foreign libraries, began in 1971.

Number of materials or nuclides: At present the library contains:
(a) 8 files from the UKNDL
(b) One complete file compiled at the Institute of Physics and Power Engineering, Obninsk
(c) 41 files with data on the anisotropy of elastic neutron scattering (7 angular momenta) for energies below 15 MeV.

List of materials or nuclides:
(a) Pu-239, Th-232, Zr, Li-6, Li-7, U-238, Be-9, B-11
(b) U-238

(c) D-2, T-3, He-3, He-4, Li-6, Li-7, Be, B, C, N, O, F, Na, Mg, Al, Si, P, S, K, Ca, Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Y, Zr, Nb, Mo, Ta, W, Pb, Bi, Th-232, U-235, U-238, Pu-239

Availability: The UKNDL files may be obtained through the IAEA (in the UKNDL format). There is at present no possibility of transmitting data on magnetic tape in the format of the USSR Evaluated Nuclear Data Library.

Special features: The results published up to April 1971 (including results presented at the Third Conference on Neutron Cross Sections and Technology at Knoxville (USA)) were taken into account in compiling the file for U-238. The resonance structure of the cross-sections is described by the parameters of the resolved resonances, by the mean resonance parameters, in the subgroup representation (above 416 eV), and as a detailed energy dependence (below 416 eV).

Main references:
Format: KOLESOV, V.E., NIKOLAEV, M.N., "Format of the recommended nuclear data library for reactor calculations", in Russian (see Appendix C). English Transl. INDC (CCP)-13/L, International Nuclear Data Committee, IAEA, Vienna (1970).
Data on anisotropy: BAZAZYANTS, N.O., ZABRODSKAYA, A.S., NIKOLAEV, M.N., Recommended values for the energy dependence of the coefficients of expansion of the elastic scattering cross section in Legendre polynomials, Bull. Nuclear Data Centre, Obninsk, Bull. Inform. Tsentra Jadern. Dannym 6, Suppl. 1 (1971) 67.

H. USSR Catalogues of 26-Group Constants

Country of origin: Union of Soviet Socialist Republics.

Date of release of latest version: A complete set of group constants was prepared in 1964 (see Ref. [1] below). In 1969 the data on resonance self-shielding were converted to the subgroup representation (see Ref. [2] below). At present, the catalogues contain revised data for U-235, Pu-239 and U-238 from 1970 and for a number of other nuclides (O, Fe, Ni, U-233) from before 1970. In addition, data have been introduced for some new materials and the list of data has been expanded. It is expected that the main materials of interest from the point of view of reactor calculations will be reviewed in 1972.

Number of materials or nuclides: 51.

List of materials or nuclides: See Part III, Tables XI - XIV.

Availability: All the data are given in the reports listed in the Reference list to Tables XI - XIV.

Special features: The sets of group constants contain information on: the total cross-section; fission cross-section and $\bar{\nu}$; absorption cross-section without fission (the sum of the cross-sections for (n,γ), (n,α) and (n,p) reactions, etc.); the inelastic scattering cross-section (added to the cross-sections for the reactions $(n,2n)$, $(n,\alpha n)$, etc.); the cross-sections for inelastic transitions between groups ($\sigma^{i}_{nn'} + 2\sigma^{ij}_{n,2n} + \ldots\ldots$). The resonance structure of the cross-sections is described either by means of the self-shielding factor (see Ref. [1]) or in the subgroup representation (see Ref. [2]).

Comments: Some remarks are contained in Part III, Tables XI - XIV.

Main references:
[1] ABAGYAN, L.P., BAZAZYANTS, N.O., BONDARENKO, I.D., NIKOLAEV, M.N., Group Constants for Nuclear Reactor Calculations, Atomizdat (1964); English Transl. Consultants Bureau, New York (1964).
[2] NIKOLAEV, M.N., KHOKHLOV, V.F., System of subgroup constants, Bull. Nuclear Data Centre, Obninsk, Bull. Inform. Tsentra Jadern. Dannym 4 (1967) 420.

I. Neutron data evaluations of French origin

Country of origin: France.

Date of release of latest version[7]

Number of materials or nuclides: 6.

List of materials or nuclides: Ni, Cr, U-238, Pu-239, Pu-240 and Pu-241.

Availability: The evaluations are available from the Centre for Neutron Data Compilation (CCDN), Saclay. Some of them are included in the UKNDL and therefore have the same distribution as the UKNDL (see Part I, A).

Special features: The evaluated data for the above six elements and isotopes are in the format of the UKNDL. However, several sets of resonance parameters are maintained at the Centre d'études nucléaires, Saclay, in a special format.

Main references: The files for Ni, Cr, U-238, Pu-239, Pu-240 and Pu-241 are in the UKNDL format and therefore the references given in Part I, A are relevant. The references for the above complete evaluations are:
Ni: RAVIER, J., VASTEL, Cadarache Rep. PNR/SE PR 65.010 (1965).
Cr: RAVIER, J., VASTEL, Cadarache Rep. PNR/SE PR 65.041 (1965).
U-238: RAVIER, J., VASTEL, Cadarache Rep. PNR/SE PR R025 (1968).
Pu-239: RIBON, P., LE COQ, G., Saclay Rep. CEA-N-1484 (EANDC(E)-138"AL") (1971).
Pu-240: L'HERITEAU, J.P., RIBON, P., Saclay Rep. CEA-N-1273 (EANDC(E)-126"AL") (1970).
Pu-241: L'HERITEAU, J.P., Scalay Internal Rep. SMPNF 806-70 (1970).

J. Australian Fission Product Library

Country of origin: Australia.

Date of release of latest version: November 1971.

Number of materials or nuclides: 192 fission products (184 ground states and 8 isomeric states).

[7] France contributes complete evaluated neutron data files to the UKNDL and does not maintain its own library.

List of materials or nuclides:

Zn-72, Ga-72, Ge-(72, 73, 74, 76, 77), As-(75, 76, 77), Se-(76, 77, 78, 79, 80), Br-(81, 82), Kr-(82, 83, 84, 85, 86), Rb-(85, 86, 87), Sr-(86, 88, 89, 90, 91), Y-(89, 90, 91, 93), Zr-(90, 91, 92, 93, 94, 95, 96, 97), Nb-95, Mo-(95, 96, 97, 98, 99, 100), Tc-99, Ru-(100, 101, 102, 103, 104, 105, 106), Rh-(103, 105), Pd-(104, 105, 106, 107, 108, 109, 110, 112), Ag-(109, 111), Cd-(110, 111, 112, 113, 114, 115, 116), In-115, Sn-(115, 116, 117, 118, 119, 120, 121, 122, 123, 124, 125, 126), Sb-(121, 122, 123, 124, 125, 126, 127, 128), Te-(122, 123, 124, 125, 126, 127, 128, 129, 130, 131, 132), I-(127, 129, 130, 131, 133, 135), Xe-(128, 130, 131, 132, 133, 134, 135, 136), Cs-(133, 134, 135, 136, 137), Ba-(134, 136, 137, 138, 140), La-(139, 140), Ce-(140, 141, 142, 143, 144), Pr-(141, 142, 143, 145), Nd-(142, 143, 144, 145, 146, 147, 148, 150), Pm-(147, 148, 149, 151), Sm-(147, 148, 149, 150, 151, 152, 153, 154, 156), Eu-(153, 154, 155, 156, 157), Gd-(155, 156, 157, 158, 159, 160), Tb-(159, 160, 161), Dy-(160, 161, 162, 163, 164), Ho-165

In addition to the above 184 ground states the following eight isomeric states are in the library. The mass numbers of these eight isomeric states have been assigned by adding 700:

Tc-799, Cd-815, Te-(823, 825, 827, 831), Pm-848.

Availability: The library has been distributed to the Centre for Neutron Data Compilation, Saclay, France, the National Neutron Cross Section Center, Brookhaven, USA, and the IAEA Nuclear Data Section.

Special features: The evaluated data in the library are in the format of the UKNDL and evaluations of the total, elastic, inelastic, capture, non-elastic, and transport cross-sections for all the fission product nuclei have been performed.

Comments: The library consists of two parts, one giving the microscopic evaluated data and the other containing evaluated group cross-sections.

Main references: Since all the data are in the UKNDL format, the references given in Part I, A are relevant. The references relating to the evaluations and the programs used in performing them are:
COOK, J.L., Rep. AAEC-TM-549 (1970).
BERTRAM, W.K. et al., Rep. AAEC-E-214 (1971).
ROSE, E.K., Rep. AAEC-TM-587 (1971).

Part II

Summaries of the current philosophy concerning the influence of macroscopic experiments and adjustments on the different evaluated neutron data files

A. Brookhaven National Laboratory (USA)

The ENDF/B library is intended to be a documented library of evaluated data that can serve as a reference file of microscopic data. It is not immediately revised upon the appearance of new data but only periodically so as to slowly improve the state of the reference file. In this way it is hoped that the ENDF/B iterations will converge toward a cross-section standard for a given material and in the interim provide a reasonable reference file.

For the development of the ENDF/B library the importance of the work of measurers in providing new data, evaluators in selecting from among data, and applied physicists in the testing of data is recognized and it is believed that all three of these talents represent important sources of judgement that will aid in the improvement of the evaluated data files. It is further believed that all evaluated files containing microscopic data must not contain blind adjustments to fit integral experiments but must contain information completely consistent with a reasonable interpolation of microscopic measurements. At the present time, the recommendations for ENDF/B-III have received the endorsement of a large group of measurers, evaluators, and reactor physicists as a reasonable reference evaluated data file to undergo further testing.

B. Lawrence Livermore Laboratory (USA)

For the Lawrence Livermore Laboratory Evaluated Nuclear Data Library (LLL) partial or complete re-evaluations are made as the need arises and as time permits. When calculations of parameters of integral neutron experiments show that important isotopes need to be reviewed, pertinent experimental data are examined and changes to the evaluations are made. Under no circumstances are the microscopic data adjusted to force agreement of calculated and experimental parameters of integral neutron experiments. An integral part of the LLL system for producing calculational constants for neutronics codes is the checking of evaluated data against integral neutron experiments. Since the ENDF/B-II and UKNDL libraries are also maintained at Livermore, those libraries are likewise checked against the same integral neutron experiments whenever those data files contain evaluations for all the materials required for the calculations. At Livermore use has been made of the TART Monte Carlo code in the 176-group modes with the equivalent of a P_{16}, S_∞ calculation. For checking of the TART code use has been made of ANISN in a 66-group, P_3, S_8 mode for a representative number of integral neutron experiments. The two codes yield essentially the same answers. Their results reveal that, for 114 integral neutron experiments, the LLL library yields 75% of the k_{eff} values between 0.99 and 1.01 while the equivalent number for 55 integral neutron experiments using the ENDF/B-II library is 36% and for 113 integral neutron experiments using the UKNDL the number of k_{eff} values in that range is 10%. The integral neutron experiments used for the ENDF/B-II and UKNDL calculations are subsets of those used for the LLL library. All the integral neutron experiments used for these calculations are simple spherical assemblies with cores of U-235, Pu-239 or U-233. The reflectors for some cases form a series with varying thicknesses of the same reflector material.

C. United Kingdom Atomic Energy Establishment, Winfrith (United Kingdom)

Since some of the most important cross-sections cannot be measured to sufficient accuracy by differential methods, it has proved necessary to institute in the United Kingdom a program for "data adjustment", which takes account of both differential and integral data, to improve the accuracy of fast reactor performance predictions. This development was described in detail by Campbell and Rowlands[8]. This technique is used to adjust the performance predictions and may be used for adjustments of the multigroup cross-sections. However, this method is not used to adjust the basic evaluated data files but is a stimulus to: (i) review the evaluated neutron cross-sections, (ii) maintain a special interest in new measurements of particular neutron nuclear data, and (iii) influence the cross-section measurement program.

D. Kernforschungszentrum Karlsruhe (Federal Republic of Germany)

With regard to the KEDAK library, the results of integral measurements are looked upon as an additional source of relevant information in relation to neutron nuclear data evaluation. However, these results are not used to adjust microscopic data, but rather act as a stimulus to: (i) review evaluated neutron cross-sections, (ii) observe with special interest the development in the nuclear field of certain quantities, and (iii) influence the experimental cross-section measurement program either at Karlsruhe or elsewhere.

E. Centro di Calcolo, Bologna (Italy)

At the Centro di Calcolo of the Comitato Nazionale per L'Energia Nucleare, the results of integral measurements are considered as a source of relevant information for evaluating neutron data, provided that the integral measurements were explicitly designed for this purpose (e.g. activation measurements in well-known spectra). In addition, "clean" integral data, such as the critical mass of small homogeneous fast assemblies of simple geometry, are used in order to make a choice between strongly discrepant quantities.

F. Institute of Physics and Power Engineering, Obninsk (Union of Soviet Socialist Republics)

Within the scope of the present evaluation activities in the Soviet Union there are no provisions for introducing corrections based on integral experiments. In future, however, it is expected that use will be made of such experiments for checking the effectiveness of the evaluated data files. The results of macroscopic integral experiments are not used for adjusting the microscopic evaluated neutron data. However, the results of such integral experiments have important repercussions on the experimental microscopic data measurement program.

G. Centre d'études nucléaires, Saclay (France)

In France, the philosophy concerning adjustments of microscopic evaluated neutron data using the results of integral measurements is similar to that in the United Kingdom and the Federal Republic of Germany. That

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[8] CAMPBELL, C.G., ROWLANDS, J.L., "The relationship of microscopic and integrated data", Nuclear Data for Reactors (Proc. Conf. Helsinki, 1970) 2, IAEA,. Vienna (1970) 391.

is, the results of integral measurements are used to adjust the multigroup cross-section sets but not the microscopic evaluated data sets. Secondly, discrepancies between the results of integral experiments and calculations based on the microscopic evaluated data are used as a "trigger" to initiate investigative action by a program of both re-evaluation and further microscopic data measurements.

Part III

Detailed summary, nuclide per nuclide, of the energy range, reaction types, references and evaluation dates appearing in the different libraries

Detailed summaries of the contents of the different evaluated neutron data libraries are given in Tables I - XIV. For each nuclide or material within each library the energy range, reaction types, references, evaluation dates and comments (when available) on the quality of a specific file are listed in tabular form. In the tables, the following points should be noted:

Evaluation date: The date entered in this column does not necessarily imply that a "total re-evaluation" was done on that date, but rather that some energy range of the incident neutron was investigated and changes were made on that date. In the case of the Lawrence Livermore Laboratory Library (Table V) the evaluation date for each reaction type for each nuclide is given; however, in all other cases, only the evaluation date for each nuclide is given and this does not necessarily imply that all reaction types for that nuclide were "totally or partially re-evaluated".

Energy range: The energy range is given in units of electron volts. For example, the energies 1.6×10^{-4} eV and 1.8×10^{7} eV are denoted by 1.6 - 4 and 1.8 + 7, respectively.

Quantity, data type:

The quantities or reaction types given in the tables are all self-explanatory except for the following:

KSI	- Average logarithmic energy decrement per elastic scattering event
GAMMA	- Parameter derived in the slowing-down theory of Goertzel-Greuling
$\bar{\mu}$	- Average laboratory angle cosine in elastic scattering
$\bar{\nu}$ (or Nu)	- Average number of neutrons emitted per fission
TR	- Transport cross-section
α	- Capture/fission cross-section ratio
η	- $\bar{\nu}/(1 + \alpha)$
FS	- Fission spectrum
RRP	- Resolved resonance parameters
SRP	- Statistical resonance parameters
C	- Continuum part of the inelastic cross-section

The data type is a symbol giving a general quantitative classification of the evaluation. The six categories of data types appearing in the tables are:

(G) - General information
(R) - Resonance parameters
(I) - Smooth or integrated cross-sections
(A) - Secondary neutron angular distributions
(E) - Secondary neutron energy distributions
(T) - Thermal neutron scattering laws

The data type is denoted in parentheses after the reaction type; for example, Elastic (I, A) means that both the integrated elastic cross-section and the elastic angular distribution have been evaluated.

Some data libraries do not include angular distribution data for (all) non-elastic secondary neutrons. Then the users have to make the implicit assumption that these secondary neutrons are distributed isotropically. The inelastic scattering cross-section is denoted as Inelastic, except in the case of the Lawrence Livermore Laboratory's Library (LLL) where it is denoted as N, N' Gamma. In general, inelastic scattering to individual levels and to a "continuum" are evaluated simultaneously, and the data may be presented in one of two different ways. In the more flexible method[9] the cross-sections to the discrete levels are presented explicitly, with or without an accompanying cross-section to a "continuum". Alternatively, the total inelastic cross-section may be given, and the branching to discrete levels and "continuum" may be represented in the secondary neutron energy distribution. The inelastic scattering cross-section in the tables appears in one of the following forms: N, N' Gamma (I, E); or Inelastic (I, E); or Inelastic (I, 1-10, C). In the first two examples, the individual levels are not listed, whilst in the third example they are. In this last example, the numbers 1-10 mean the first to the tenth excited states, and C refers to the "continuum". The use of the term "continuum" here is somewhat loose since it caters for that part of the (n, n') or (n, n' γ) reaction which is not covered by the (n, n') or (n, n' γ) reaction from the first to the tenth excited states.

Similarly, the individual excitation functions for the n, f; n, n'f; n, 2nf and n, 3nf reactions or a single excitation function for the fission process may be presented, the first, second, third and fourth chance of fission being dealt with by means of an energy-dependent fission spectrum. The latter mode of representation has been chosen throughout.

[9] That this mode of representation is more flexible can be appreciated if one considers the problem of representing different secondary angular distributions for the different discrete level excitation functions.

Laboratory codes: The abbreviations given in the tables have the following meaning:

AE	Aktiebolaget Atomenergi, Studsvik, Sweden
AI	Atomics International, Canoga Park, Calif., USA
ALD	UKAEA Atomic Weapons Research Establishment, Aldermaston, United Kingdom
ANC	Aerojet Nuclear Corp., Idaho Falls, Idaho, USA
ANL	Argonne National Lab., Idaho Falls, Idaho, USA
APD	Atomic Power Development Associates, Inc., Detroit, Mich., USA
BET	Bettis Atomic Power Lab., Pittsburgh, Pa., USA
BNL	Brookhaven National Lab., Upton, N.Y., USA
BNW	Battelle Pacific Northwest Lab., Richland, Wash., USA
B+W	Babcock and Wilcox Co., Lynchburg, Va., USA
CAD	CEA Centre d'études nucléaires de Cadarache, Saint-Paul-lez-Durance, France
GEA	General Electric Co., Nuclear Materials and Propulsion, Cincinnati, Ohio, USA
GEB	General Electric Co., Sunnyvale, Calif., USA
GEL	Central Bureau for Nuclear Measurements, Commission of the European Communities, Geel, Belgium
GGA	Gulf General Atomic, Inc., San Diego, Calif., USA
HAR	UKAEA Research Group, Atomic Energy Research Establishment, Harwell, United Kingdom
HED	Hanford Engineering Development Lab., Hanford, Wash., USA
IIT	Illinois Inst. of Tech., Chicago, Ill., USA
JAE	Japan Atomic Energy Research Inst., Tokyo, Japan
KAP	Knolls Atomic Power Lab., Schenectady, N.Y., USA
KFK	Kernforschungszentrum Karlsruhe, Federal Republic of Germany
LAS	Los Alamos Scientific Lab., N.Mex., USA
LLL	Lawrence Livermore Laboratory, Livermore, Calif., USA
ORL	Oak Ridge National Lab., Tenn., USA
RLY	UKAEA Reactor Group, Risley, United Kingdom
SAC	CEA Centre d'études nucléaires de Saclay, Gif-sur-Yvette, France
SOR	Soreq Research Centre, Yavne, Israel
TOK	Tokyo University, Tokyo, Japan
WAP	Westinghouse Electric Corp., Atomic Power Div., Pittsburgh, Pa., USA
WEW	Westinghouse Electric Corp., Advanced Reactor Div., Pittsburgh, Pa., USA
WIN	UKAEA Reactor Group, Atomic Energy Establishment, Winfrith, Dorset, United Kingdom

TABLE I. UNITED KINGDOM NUCLEAR DATA LIBRARY (UKNDL): COMPLETE DATA FILES

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
H-1 (901)	1.0 - 4	2.0 + 7	Horsley (1966), Nucl. Data A2, 243 above 1 keV. Butland (1970) WNDG/76, unpublished below 1 keV	ALD WIN	For H in H_2O; adequate, but could be improved marginally. Total (I); Elastic (I, A); N, Gamma (I)	9/66 2/70
D-2 (905)	1.0 - 4	2.0 + 7	Butland (1971), unpublished, to 10 eV. Story[a], above 10 eV	WIN	For D in D_2O. Total (I); Elastic (I, A); N, 2N (I, A, E); N, Gamma (I)	4/71
T-3 (252)	1.0 - 4	2.0 + 7	From a compilation by Stewart (1965) LA-3270, Horsley, unpublished	LAS ALD	Total (I); Elastic (I, A); Nonel (I); N, 2N (I, A, E)	10/67
He-3 (220 D)	1.0 - 4	1.5 + 7	From a compilation by Stewart (1965) unpublished	LAS (WIN)	See comment for He-4. Total (I); Elastic (I, A); Nonel (I); N, P (I); N, D (I)	8/65 (8/67)
He-4 (221 D)	1.0 - 4	1.5 + 7	From a compilation by Stewart (1965) unpublished	LAS (WIN)	Believed adequate, ENDF/B file MAT-1088 may be marginally better for natural helium. Total (I); Elastic (I, A)	8/65 (8/67)
Li-6 (214 D)	1.0 - 4	1.5 + 7	Pendlebury (1964) AWRE-O-60/64	ALD (WIN)	Total (I); Elastic (I, A); Nonel (I); Inelastic (I 1, A); N, N' Alpha (I, A, E); N, 2N Alpha (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, Alpha (I);	4/65 (8/67)
(914)			Sowerby, Uttley (1972)	HAR	revised file (DFN 914) now available	2/72

Li-7 (215 D)	1.0 + 4	1.5 + 7	Pendlebury (1964) AWRE-O-61/64	ALD (WIN)	Old file, probably adequate for the present. (Same reaction types as for Li-6 above, but add N, 2N (I, A, E) and omit N, Alpha)	4/65 (8/67)
Be-9 (50 A)	1.0 + 4	1.5 + 7	Doherty (1965) AEEW-M513	WIN	Reasonably satisfactory, but virtually no requirement in the UK reactor program. Total (I); Elastic (I, A); Nonel (I); N, 2N Alpha (I, A, E); N, Gamma (I); N, T (I); N, Alpha (I)	1/65 (2/68)
B (57)	1.0 - 4	1.5 + 7	Parker (1957) unpublished, Norton (1968) AEEW-M824, Appendix A	ALD WIN	Interim file; use of more recent files for B-10 and B-11 is recommended. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); Parasitic absorption (I)	4/57 4/67
B-10 (90 A)	1.0 + 4	1.5 + 7	Iijuna (1970) unpublished	WIN (JAE)	Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-6, C; A, E); Parasitic absorption (I); N, P (I); N, D (I); N, T (I); N, Alpha (I)	5/70
B-11 (49 A)	1.0 + 4	1.5 + 7	Norton et al. (1966) unpublished	WIN (ALD)	Probably adequate, but could be improved. Total (I); Elastic (I, A); Inelastic (I, 1-3, A); Inelastic (I, C, A, E); N, Gamma (I); N, P (I); N, T (I); N, Alpha (I)	8/66 (2/68)
C (902 A)	1.0 + 4	1.5 + 7	Douglas, Porter, Wyld (1971) to be published	ALD	Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1, A); N, N' 3 Alpha (I, A, E); N, Gamma (I); N, Alpha (I)	10/71

[a] Nuclear Data for Reactors - 1970 (Proc. Conf. Helsinki, 1970) 1, IAEA, Vienna (1970) 721.

TABLE I. (cont.)

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
N (259)	1.0 - 4	1.5 + 7	Craven (1967) above 10 keV, Pope (1967) NDFWP/P69, unpublished below 10 keV	ORL WIN	Probably adequate, but could be improved. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1, 2; A); Inelastic (I, C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); N, 2 Alpha (I)	10/67 7/67
O (33 E)	1.0 - 4	1.5 + 7	Slaggie, Reynolds (1965) KAPL-M6452, above 15 keV, Butland, Pope, Story (1967, 1971), below 15 keV	KAP WIN	Believed adequate – mainly ENDF/B file MAT-1013; LASL revision to be converted to UK format in due course. Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-10; A); Inelastic (C, A, E); N, Gamma (I); N, P (I); N, D (I); N, Alpha (I)	4/65 10/67 (12/71)
F-19 (23 D)	1.0 - 4	1.5 + 7	Parker, Pendlebury (1958) unpublished. See also Barrington, Pope, Story (1964) AEEW-R351	ALD (WIN)	Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I)	11/57 (8/67)
Na-23 (93)	1.0 - 5	1.5 + 7	Pitterle (1968) APDA-217	APD (WIN)	ENDF/B file MAT-1059, converted to UKNDL format; probably adequate at present. Some format corrections are still required. Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-7, A); Inelastic (C, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	6/68 (6/70)

Al-27 (35 E)	1.0 - 4	1.5 + 7	King (1964) AEEW-M445	WIN	Probably adequate; revision desirable, especially of (N, Gamma) in resonance range. Total (I); Elastic (I, A); Inelastic (I, 1-9, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	1/64 (8/67)
Si (25 D)	1.0 - 4	1.5 + 7	Buckingham, Huives, Pendlebury (1959) unpublished. See also Barrington, Pope, Story (1964) AEEW-R 351	ALD (WIN)	Very old file; (n, gamma) needs improvement in the resonance range. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	10/58 (8/67)
Cl (141 D)	1.0 - 4	1.5 + 7	Buckingham, Pendlebury (1959) unpublished	ALD (WIN)	Very old file. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	2/59 (1/67)
K (84 A)	1.0 - 4	2.0 + 7	Mainly from the compilation by Drake (1967) GA-7829, Part V. See also Norton (1969) NDFWP/P93, unpublished	GGA (WIN)	Probably adequate for the present. Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-6, A); Inelastic (C, A, E); N, 2N (I, A, E); N, N' Alpha (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	(4/70)
Ca (138 D)	1.0 - 4	1.5 + 7	Buckingham, Huives, Pendlebury (1959) unpublished	ALD (WIN)	Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	10/58 (1/67)

TABLE I. (cont.)

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
Ti (190 A)	1.0 - 4	1.8 + 7	Miller, Parker (1964) AWRE-O-77/64. Mainly from Tralli et al. (1962) UNC-5002	ALD (WIN)	May be adequate. The ENDF/B file MAT-1016 should be better, but further revision of resonance range may be needed, especially for (n, gamma). Total (I); Elastic (I, A); Nonel (I); Inelastic (I,1-4, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	6/64 (1/67)
V (952)	1.0 - 5	1.5 + 7	Pennington, Gajniak (1968) ANL-7387, Cameron, Douglas (1972) unpublished	ANL (ALD)	ENDF/B file MAT-1017 converted to UKNDL format. May be adequate, but further revision in the resonance range desirable, especially for (n, gamma). Total (I); Elastic (I, A); Total inelastic (I); Inelastic (I,1-4, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	6/70 (1/72)
Cr (45 D)	1.0 - 4	1.5 + 7	Ravier, Vastel (1965) PNR/SEPR-65.041	CAD (WIN)	Revision is needed, especially for (n, gamma) in the resonance region. Total (I); Elastic (I, A); Nonel (I); Inelastic (I,1-8, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I)	9/65 (10/67)

Mn-55 (88)	1.0 - 5	1.5 + 7	Stephenson, Prince, Pearlstein (1967) BNL-50060, Macdougall (1970) unpublished	BNL (WIN)	ENDF/B file MAT-1019, converted to UKNDL format and believed adequate. Some format correction still needed. Total (I): Elastic (I, A); Total inelastic (I); Inelastic (I,1-6, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	6/67 (6/70)
Fe (906 A)	1.0 - 4	3.3 + 5	Pope, Story (1971, 1972) unpublished	WIN	Continuing revision in resolved resonance range. DFN 950 (see below) to be used above 330 keV. Total (I); Elastic (I, A); N, Gamma (I)	10/71 (3/72)
Fe (950)	1.0 - 5	1.5 + 7	Penny, Kinney (1970) ORNL-4617, Cameron, Dean (1972)	ORL (ALD) (WIN)	ENDF/B file MAT-1124, converted to UKNDL format, for use above 330 keV. Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I,1-20, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	/70 (1/72)
Ni (907)	1.0 - 4	2.4 + 5	Moxon (1970) unpublished	HAR	Revision in resolved resonance range. DFN 951 (see below) to be used above 240 keV. Total (I); Elastic (I, A); N, Gamma (I)	10/71
Ni (951)	1.0 - 5	1.5 + 7	Azziz, Cornyn (1969) WCAP-7281 Cameron, Dean (1972)	WAP KAP (ALD) (WIN)	ENDF/B file MAT-1123 converted to UKNDL format, for use above 240 keV. Total (I); Elastic (I, A); Total inelastic (I); Inelastic (I,1-12, A); Inelastic (C, A, E); N, 2N (I, A, E); N, N' Alpha (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	7/70 (1/72)

TABLE I. (cont.)

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, date type	Evaluation date
	Minimum	Maximum				
Cu (73)	1.0 - 4	1.5 + 7	Häggblom, Pope (1969) NDFWP/P96, unpublished	AE (WIN) ALD	Probably adequate; revises DFN 249 in resonance range. Total (I); Elastic (I, A); Total inelastic (I); Inelastic (I, 1-9, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	4/70
Ga (105 A)	1.0 - 4	1.5 + 7	Parker, Pendlebury (1958) unpublished	ALD (WIN)	May be adequate, but revision in the resonance range is desirable, especially for (n, gamma). Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I)	11/57 (1/67)
Zr (82 A)	1.0 - 4	1.5 + 7	Pope, Story (1969) AEEW-M 921	WIN	Probably adequate; could be further improved in the resonance region. Total (I); Elastic (I, A); Nonel (I); Inelastic (I,1-8, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I)	4/70
Nb (79 A)	1.0 - 4	1.5 + 7	Blow, Lipscombe (1969) AERE-M2230	HAR (GGA)	Mainly ENDF/B file MAT-1024; probably adequate, but MAT-1164 may be better. Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-10, A); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I)	/68 (4/70)

Mo (81 A)	1.0 - 4	1.5 + 7	Blow, Lipscombe (1969) AERE-M2230	HAR (ANL) (KFK)	Mainly ENDF/B file MAT-1025. Revision needed, especially (n, gamma). Total (I); Elastic (I, A); Nonel (I); Total inelastic (I); Inelastic (I, 1-4, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, N' Alpha (I, A, E); N, Gamma (I); N, P (I)	/68 (4/70)
Cd (70)	1.0 - 4	1.5 + 7	James (1970) unpublished	WIN	Resonance range reviewed. Evaluation of Drake (1966) GA-6997 used at higher energies. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-4, A); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I); N, Gamma (I); N, P (I); N, Alpha (I)	4/70
Cd-113 (71 A)	1.0 - 4	1.5 + 7	James (1970) unpublished	WIN	Resonance range reviewed. Evaluation of Drake (1966) GA-6997 used at higher energies. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-3, A); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I); N, Gamma (I); N, P (I); N, Alpha (I)	4/70
Xe-135 (4E)	1.0 - 4	1.0 + 3	Summer (1962) AEEW-R116	WIN	Believed adequate, extends only to 1 keV. Total (I); Elastic (I, A); N, Gamma (I)	3/62 (8/67)
Ta-181 (328 A)	1.0 - 4	1.5 + 7	Hart (1966) AHSB(S)R-141	RLY (WIN)	Believed adequate. Total (I); Elastic (I, A); Total inelastic (I); Inelastic (I, 1-8, A); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I)	7/66 (4/70)

TABLE I. (cont.)

Material (Data file No.)	Energy range (eV) Minimum	Energy range (eV) Maximum	Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
W (213 A)	1.0 - 4	1.5 + 7	Gately, Parker (1965) unpublished	LLL (ALD)	Mainly from a compilation by Howerton, LLL. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I)	3/65 (1/67)
Au-197 (222 D)	1.0 - 4	1.5 + 7	Offord et al. (1966) unpublished	(ALD) LLL	Based on a compilation by Howerton, LLL, with improvement of (n, n'), (n, 2n) and (n, gamma) data. Complete revision of (n, gamma) desirable, especially in resonance range, for use as reference standard. No reactor requirement. ENDF/B file MAT-1166 may be better basis for revision. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-9, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 2N (isomer activation); N, Gamma (I); N, P (I)	10/65 (4/70)
Pb (26 B)	1.0 - 4	1.5 + 7	Buckingham, Pendlebury (1960) unpublished. See also Barrington et al. (1964) AEEW-R351	ALD (WIN)	Probably adequate. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, Gamma (I)	 8/59 (3/63) (4/70)
Th-232 (22 A)	1.0 - 4	1.5 + 7	Hinves et al. (1959) unpublished. See also Barrington et al. (1964) AEEW-R351	ALD (WIN)	For (n, fission) only, DFN 332 is preferred. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	3/59 (3/63) (4/70)

Pa-233 (86)	1.0 - 4	1.0 + 7	Evaluation of Drake, Nichols (1967) GA-7462. Compiled by Barchay (1970)	GGA (WIN)	Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	9/67 (4/70)
U-233 (87 A)	1.0 - 4	1.5 + 7	Ainger (1969) NDFWP/P97, unpublished, Hart (1969) AHSB(S)R-169, fission cross-section	WIN RLY (WIN)	Total (I); Elastic (I, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	8/69 (4/70)
U-234 (74 A)	1.0 + 3	1.5 + 7	Parker (1964) AWRE-O-37/64, Hart (1967) AHSB(S)R-124, fission cross-section	ALD RLY (WIN)	Covers the range 1 keV to 15 MeV only. Conversion of ENDF/B file MAT-1043 to UKNDL format in progress. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-6, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	8/62 2/67 (4/70)
U-235 (271 D)	1.0 - 4	1.5 + 7	Douglas (1972) AWRE Nuclear Research Note NRN-4/72, Sowerby et al. (1972) AERE-M2497, Mather, Bampton (1971) AWRE-O-55/71	ALD HAR ALD	Recently revised above 25.5 keV, using older file DFN 66A at lower energies. Nu-bar revised over whole energy range. A GENEX file spanning the range 0.4 eV to 25 keV is also available, which is more up to date than DFN 66A. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-6, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	2/72
U-236 (75 A)	1.0 + 3	1.5 + 7	Parker (1964) AWRE-O-30/64, Hart (1967) AHSB(S)R-124, fission cross-section	ALD RLY (WIN)	Covers the range 1 keV to 15 MeV only. Conversion of ENDF/B file MAT-1046 to UKNDL format in progress. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-6, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	8/62 2/67 (4/70)

TABLE I. (cont.)

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
U-238 (272 A)	1.0 - 4	1.5 + 7	Douglas (1972) AWRE Nuclear Research Note NRN-4/72, Sowerby et al. (1972) AERE-M2497, Mather, Bampton (1971) AWRE-O-44/71	ALD HAR ALD	Recently revised above 25 keV, using at lower energies the French file DFN 401A. A new GENEX file is available also, from 0.4 eV to 25 keV. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-10, A); Inelastic (C, A, E); N, Gamma (I)	2/72
Pu-238 (216 D)	1.0 - 4	1.5 + 7	Adams, Parker (1965) unpublished	(ALD) LLL (WIN)	Mainly from a compilation by Howerton, LLL, which may be adequate for the present. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	6/65 (8/67)
Pu-239 (269 D)	1.0 - 4	1.5 + 7	Douglas (1972) AWRE Nuclear Research Note NRN-4/72, Sowerby et al. (1972) AERE-M2497, Mather et al. (1970) AWRE-O-86/70	ALD HAR ALD	Recently revised above 8.5 keV, using older file DFN 65A at lower energies. Nu-bar revised over whole energy range. A new GENEX file is available also, from 0.4 eV to 25 keV. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-9, A); Inelastic (C, A, E); N, 2N (I); N, 3N (I); N, Fission (I, A, E, Nu); N, Gamma (I)	2/72
Pu-240 (77 A) (402 A)	1.0 - 4	1.5 + 7	Douglas (1965) AWRE-O-91/64, Hart (1967) AHSB(S)R-124, fission cross-section above 1 keV	ALD RLY (WIN)	A new French file is available, DFN 402 A, and is currently preferred. Total (I); Elastic (I, A); Nonel (I); Inelastic (I, 1-3, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	5/64 3/67 (4/70)

Pu-241 (60)	1.0 - 4	1.5 + 7	Pope (1968) AEEW-M824, Appendix B, Doherty (1966) AEEW-M714	WIN WIN	A new French file is available, DFN 403A, and is currently preferred. Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	8/67
Pu-242 (955)	1.0 - 5	1.5 + 7	Alter, Dunford (1967) NAA-SR-12271 and supplement	AI (ALD)	Conversion of ENDF/B file MAT-1055 to UKNDL format. Total (I); Elastic (I, A); Total inelastic (I); Inelastic (I, 1-10, A); Inelastic (C, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E, Nu); N, Gamma (I)	5/67 (3/72)
Am-241	1.0 - 5	1.5 + 7	Smith, Grimesay (1966) unpublished	ANC (ALD)	Conversion of ENDF/B file MAT-1056 to UKNDL format in progress	11/66 (6/70) (3/72)
Am-243	1.0 - 5	1.5 + 7	Smith, Grimesay (1966) unpublished	ANC (ALD)	Conversion of ENDF/B file MAT-1057 to UKNDL format in progress	11/66 (6/70) (3/72)
Cm-244	1.0 - 5	1.5 + 7	Dunford, Alter (1967) NAA-SR-12271 and supplement	AI (ALD)	Conversion of ENDF/B file MAT-1058 to UKNDL format in progress	5/67 (7/67) (3/72)
FISPROD (106)	2.5 - 2	1.5 + 7	Buckingham, Pendlebury (1959) unpublished	ALD	Total (I); Elastic (I, A); Nonel (I); Inelastic (C, A, E); N, 2N (I, A, E); Parasitic absorption (I)	6/59

TABLE II. UNITED KINGDOM NUCLEAR DATA LIBRARY (UKNDL): ACTIVATION DETECTORS

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
Na-23 (224 A)	1.0 - 4	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (WIN)	N, Gamma (I)	6/65 (8/67)
Mg-24 (225)	5.0 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I)	6/65
Al-27 (226)	2.6 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I) (N, Alpha (I))	6/65
(95)	5.0 + 6	2.0 + 7	Kanda, Nakasima (1968) Washington Conf. [a]	TOK (WIN)	N, Alpha (I)	3/68
(96)	5.5 + 6	2.0 + 7	Spaepen (1967) INDC-107, 258	GEL (WIN)	N, Alpha (I)	5/67
Si-28 (227)	4.6 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I)	6/65
P-31 (228)	2.02 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I)	6/65
S-32 (229)	1.76 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I)	6/65

S-32 (97)	1.6 + 6	2.0 + 7	Spaepen (1967) INDC-107, 258	GEL (WIN)	N, P (I)	5/67
S-34 (230)	4.8 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, Alpha (I)	6/65
Cl-35 (231)	2.71 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	ALD	N, Alpha (I)	6/65
Sc-45 (207)	1.19 + 7	1.5 + 7	Parker, Pendlebury (1964) unpublished	ALD	N, 2N activation of Sc-44, 3.92 h and 2.44 d	8/64
Ti-46 (912)	2.45 + 6	1.5 + 7	Kamphouse et al. (1971) J. nucl. Materials 39, 1	GEA (WIN)	N, P (I)	/71
Mn-55 (232 A)	1.0 - 4	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (WIN)	N, Gamma (I)	6/65 (8/67)
Fe-54 (63)	8.0 + 5	2.0 + 7	Story (1967) unpublished	WIN	N, P (I)	9/67
(911)	1.2 + 6	1.5 + 7	Kamphouse et al. (1971) J. nucl. Materials 39, 1	GEA (WIN)	N, P (I)	/71

[a] KANDA, Y., NAKASIMA, R., "Review of some fast neutron cross-section data", Neutron Cross Sections and Technology (GOLDMAN, D.T., Ed.), National Bureau of Standards, Washington (1968) 193.

TABLE II. (cont.)

Material (Data file No.)	Energy range (eV) Minimum	Energy range (eV) Maximum	Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
Fe-56 (62)	3.8 + 6	2.0 + 7	Story (1967) unpublished	WIN	N, P (I)	9/67
(234)	3.5 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, P (I)	6/65
(98)	5.0 + 6	2.0 + 7	Kanda, Nakasima (1968) Washington Conf. [a]	TOK (WIN)	N, P (I)	3/68
Co-59 (235 A)	1.0 - 4	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (WIN)	N, Gamma (I)	6/65 (8/67)
Ni-58 (236)	1.0 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, 2N (I) ; N, P (I)	6/65
(909)	7.45 + 5	1.5 + 7	Meyer (1971) unpublished	KFK (WIN)	N, P (I)	8/71
(910)	1.0 + 6	1.5 + 7	Kamphouse et al. (1971) J. nucl. Materials 39, 1	GEA (WIN)	N, P (I)	/71
Cu-63 (237 A)	1.0 - 4	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (WIN)	N, 2N (I); N, Gamma (I)	6/65
(99)	1.2 + 7	2.0 + 7	Kanda, Nakasima (1968) Washington Conf. [a]	TOK (WIN)	N, 2N (I)	3/68

Cu-65 (100)	1.05 + 7	2.0 + 7	Kanda, Nakasima (1968) Washington Conf. [a]	TOK (WIN)	N, 2N (I)	3/68
Y-89 (208)	1.2 + 7	1.5 + 7	Parker, Pendlebury (1964) unpublished	ALD	N, 2N (I)	8/64
Zr-90 (238)	1.22 + 7	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, 2N (I)	6/65
Rh-103 (204)	9.5 + 6	1.5 + 7	Parker (1964) unpublished	ALD	N, 2N (I)	6/64
(94 A)	4.5 + 4	1.5 + 7	Story (1968) unpublished	WIN	Inelastic, activation of 57-min isomer. Based on data of Butler, Santry (1968) AECL-3043	/68 6/70
In-115 (239)	3.4 + 5	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	Inelastic, activation of 4.5-h isomer	6/65
I-127 (240)	9.5 + 6	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, 2N (I)	6/65
Tm-169 (209)	8.1 + 6	1.5 + 7	Pendlebury (1964) unpublished	ALD	N, 2N (I)	8/64
Lu-175 (210)	7.9 + 6	1.5 + 7	Pendlebury (1964)	ALD	N, 2N (I)	8/64

[a] KANDA, Y., NAKASIMA, R., "Review of some fast neutron cross-section data", Neutron Cross Sections and Technology (GOLDMAN, D.T., Ed.), National Bureau of Standards, Washington (1968) 193.

TABLE II (cont.)

Material (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Comments, quantity, data type	Evaluation date
	Minimum	Maximum				
Au-197 (241)	1.0 - 2	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (WIN)	N, Gamma (I)	6/65 (8/67)
Th-232 (242)	1.0 - 2	1.8 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II	IIT (ALD)	N, Fission (I); N, Gamma (I)	6/65
Np-237 (61)	1.0 - 4	1.4 + 7	Barrall, McElroy (1965) AFWL-TR 65-34, Vol. II, below 1 keV	IIT	N, Fission (I)	6/65 3/67
			Hart (1967) AHSB(S)R-124, above 1 keV	RLY (WIN)		(9/67)

TABLE III. UNITED KINGDOM NUCLEAR DATA LIBRARY (UKNDL): FISSION PRODUCTS [a]

Isotope	Data file No.	Isotope	Data file No.	Isotope	Data file No.
Se-82	702	Ru-104	728	Cs-137	753
Br-81	701	Rh-103	727	Ba-138	754
Kr-83	904 [b]	Rh-105	729	La-139	755
Kr-84	704	Pd-105	730	Ce-140	756
Kr-85	705	Pd-106	731	Ce-142	758
Kr-86	707	Pd-107	732	Pr-141	757
Rb-85	706	Pd-108	733	Nd-143	759
Rb-87	708	Ag-109	734	Nd-144	760
Sr-88	709	Cd-113	735	Nd-145	761
Sr-89	710	In-115	736	Nd-146	762
Sr-90	712	Sb-125	737	Nd-148	764
Y-89	711	Te-128	739	Nd-150	768
Y-90	713	Te-130	741	Pm-147	903 [c]
Y-91	714	I-127	738	Pm-148	765
Zr-91	715	I-129	740	Pm-148m	766
Zr-92	716	I-131	742	Sm-149	767
Zr-93	717	I-135	749	Sm-150	769
Zr-94	718	Xe-131	743	Sm-151	770
Zr-96	720	Xe-132	744	Sm-152	771
Mo-95	719	Xe-133	745	Sm-154	773
Mo-97	721	Xe-134	747	Eu-153	772
Mo-98	722	Xe-135	750	Eu-154	774
Mo-100	724	Xe-136	752	Eu-155	775
Te-99	723	Cs-133	746	Gd-155	776
Ru-101	725	Cs-134	748	Gd-156	777
Ru-102	726	Cs-135	751	Gd-157	778

[a] Unless otherwise indicated, the files span the energy range 0.0001 eV to 10 MeV and were compiled in August 1967 using unpublished compilations by Cook (Lucas Heights) up to 1 keV or above, and BENZI, V., BORTOLANI, M.V., "Fission-product neutron-capture cross-sections in the energy range 1 keV - 10 MeV", Nuclear Data for Reactors (Proc. Conf. Paris, 1966) 1, IAEA, Vienna (1967) 537.

[b] Hammond (WIN, 1970) WNDG/101, unpublished, below 1 keV; Benzi, Reffo (1969) CCDN-NW/10, from 1 keV to 10 MeV.

[c] Hammond (WIN, 1970) WNDG/101, unpublished, below 246 eV; Benzi, Reffo, CNEN, Bologna (1970) private communication, from 246 eV to 15 MeV.

TABLE IV. EVALUATED NUCLEAR DATA FILE (ENDF/B-III)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
H-1 (1148)	1.0 - 5	2.0 + 7	LA-4574 (1971)	LAS	Total (I); Elastic (I, A); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Inelastic (T); Photon production (I, A, E)	10/70
H-2 (1120)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW BNL	Total (I); Elastic (I, A); N, 2N (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Inelastic (T); Decay chain data (G)	6/67
He (1088)	1.0 - 5	1.5 + 7	ANL-7462 (1968)	ANL	Total (I); Elastic (I, A); N, P (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Inelastic (T)	6/68
He-3 (1146)	1.0 - 5	2.0 + 7	Private communication (1971)	LAS	Total (I); Elastic (I, A); N, P (I); N, D (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	1968
Li-6 (1115)	1.0 - 5	2.0 + 7	AWRE-O-60/64 (in part)	LAS	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N Alpha (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	8/71
Li-7 (1116)	1.0 - 5	1.5 + 7	AWRE-O-61/64 (in part)	LAS	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 2N Alpha (I, A, E); N, Gamma (I); N, D (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	8/71
Be-9 (1154)	1.0 - 5	2.0 + 7	Private communication (1971)	LLL	Total (I); Elastic (I, A); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	12/71

B-10 (1155)	1.0 - 5	1.5 + 7	ORNL-TM-1872 (1967)	LAS ORL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, D (I); N, T (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	1/72
B-11 (1160)	1.0 - 5	1.5 + 7	DFN 49A (1971)	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); N, P (I); N, T (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	9/71
C-12 (1165)	1.0 - 5	1.5 + 7	KAPL-3099 (1966)	KAP	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	1/72
N-14 (1133)	1.0 - 5	2.0 + 7	To be published (LASL) Young, Foster, Jr.	LAS	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, T); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); N, N Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A)	1/71
O-16 (1134)	1.0 - 5	2.0 + 7	To be published (LASL) Young, Foster, Jr.	LAS	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A); N, Gamma (I); N, P (I); N, D (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A)	8/71
Na-23 (1156)	1.0 - 5	1.5 + 7	To be published (WARD) Paik, Pitterle To be published (ORNL) Perey	WEW ORL	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, E, T); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G); RRP (R); Photon production (I, A, E)	1971
Mg (1014)	1.0 - 5	1.8 + 7	ANL-7387 (1968)	ANL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	9/66

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
V (1017)	1.0 - 5	1.5 + 7	ANL-7387 (1968)	ANL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I)	9/66
Cr (1121)	1.0 - 5	1.5 + 7	WCAP-7281 (1969)	WAP BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R)	7/70
Mn-55 (1019)	1.0 - 5	2.0 + 7	BNL-50060 (1967)	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); Decay chain data (G); RRP (R)	6/67
Fe (1180)	1.0 - 5	1.5 + 7	ORNL-4617 (1970)	ORL	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); Photon production (I, A, E)	1/72
Co-59 (1118)	1.0 - 5	2.0 + 7	To be published (BNL) Stephenson, Prince	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); Decay chain data (G); RRP (R)	1971
Al-27 (1135)	1.0 - 5	2.0 + 7	To be published (LASL) Foster, Jr., Young	LAS	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); Photon production (I, A, E)	4/71

Si (1151)	1.0 - 5	2.0 + 7	GA-8628 (1968)	BNL GGA	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, E); N, 2N (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	8/71
Cl (1149)	1.0 - 5	2.0 + 7	GA-7829, Vol. 4 (1967)	GGA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, N' Alpha (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	2/67
K (1150)	1.0 - 5	2.0 + 7	GA-7829, Vol. 5 (1967)	GGA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, N'Alpha (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	2/67
Ca (1152)	1.0 - 5	2.0 + 7	GA-7829, Vol. 6 (1967)	ORL GGA	Total (I); Elastic (I, A); Non-elastic (I), Inelastic (I, A, E); N, 2N (I, A, E); N, N'Alpha (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	10/71
Ni (1123)	1.0 - 5	1.5 + 7	WCAP-7387 (1969)	WAP BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, N'P (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	6/71
Cu (1087)	1.0 - 5	1.5 + 7	AI-AEC-12741 (1968)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	9/68

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Cu-63 (1085)	1.0 - 5	1.5 + 7	AI-AEC-12741 (1968)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	9/68
Cu-65 (1086)	1.0 - 5	1.5 + 7	AI-AEC-12741 (1968)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	9/68
Kr-83 (1201)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Zr-95 (1202)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Nb-93 (1164)	1.0 - 5	1.5 + 7	GA-8133 + ADD (1967)	GGA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R)	1/67
Nb-95 (1203)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

Mo (1111)	1.0 - 5	1.5 + 7	ANL-7387 (1968)	ANL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R)	10/66
Mo-95 (1204)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E), N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Mo-97 (1205)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Mo-98 (1206)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	7/71
Mo-99 (1207)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	7/71
Mo-100 (1208)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	7/71
Ru-101 (1210)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Ru-102 (1211)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Ru-103 (1212)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Ru-104 (1213)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Ru-105 (1214)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Ru-106 (1215)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Rh-105 (1217)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Pd-105 (1218)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Pd-106 (1219)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71

Pd-107 (1220)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Pd-109 (1221)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I)	7/71
Ag-107 (1138)	1.0 - 5	1.5 + 7	To be published (BNL) Bhat, Prince	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R)	10/71
Ag-109 (1139)	1.0 - 5	1.5 + 7	To be published (BNL) Bhat, Prince	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R)	10/71
Cd-113 (1223)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
I-131 (1224)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
I-135 (1225)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Xe-131 (1226)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV) Minimum	Energy range (eV) Maximum	Reference, source	Laboratory code	Quantity, data type	Evaluation date
Xe-133 (1227)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Xe-135 (1026)	1.0 - 5	1.0 + 3	Private communication (1967)	BNW	Total (I); Elastic (I); N, Gamma (I); Decay chain data (G)	6/67
Cs-133 (1141)	1.0 - 5	1.5 + 7	To be published (BNL) Bhat, Prince	BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	10/71
Cs-135 (1229)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Cs-137 (1230)	1.0 - 5	1.5 + 7	HEDL-TME 71-106 (1971)	B+W HED	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I)	7/71
La-139 (1231)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Ce-141 (1232)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

Pr-141 (1233)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Pr-143 (1234)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Nd-143 (1235)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Nd-145 (1236)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Nd-147 (1237)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Pm-147 (1238)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Pm-148g (1239)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Pm-148m (1254)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Pm-149 (1240)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Pm-151 (1241)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Sm-147 (1242)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Sm-148 (1243)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Sm-149 (1027)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/67
Sm-150 (1244)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

Sm-151 (1245)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Sm-152 (1246)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); μ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Sm-153 (1247)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); μ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Eu-151 (1028)	1.0 - 5	2.0 + 7	Private communication (1970)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); μ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/70
Eu-153 (1029)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); μ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/67
Eu-154 (1248)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); μ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Eu-155 (1249)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); μ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Eu-156 (1250)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Eu-157 (1251)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Decay chain data (G)	7/71
Gd (1030)	1.0 - 5	1.5 + 7	ANL-7387 (1968)	ANL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R)	10/66
Gd-155 (1252)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Gd-157 (1253)	1.0 - 5	1.5 + 7	BAW-409 (1971)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); Decay chain data (G)	7/71
Dy-164 (1031)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/67

Lu-175 (1032)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/67
Lu-176 (1033)	1.0 - 5	2.0 + 7	Private communication (1967)	BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	6/67
Ta-181 (1126)	1.0 - 5	1.7 + 7	AI-AEC-12990 (1971)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R)	4/71
Ta-182 (1127)	1.0 - 5	1.7 + 7	AI-AEC-12990 (1971)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R)	4/71
W-182 (1060)	1.0 - 5	1.5 + 7	GEMP-448 (1966)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	11/66
W-183 (1061)	1.0 - 5	1.5 + 7	GEMP-448 (1966)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	11/66

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
W-184 (1062)	1.0 - 5	1.5 + 7	GEMP-448 (1966)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	11/66
W-186 (1063)	1.0 - 5	1.5 + 7	GEMP-448 (1966)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, Gamma (I); N, P (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	11/66
Re-185 (1083)	1.0 - 5	1.5 + 7	GEMP-587 (1968)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R): SRP (R); Decay chain data (G)	1/68
Re-187 (1084)	1.0 - 5	1.5 + 7	GEMP-587 (1968)	GEA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	1/68
Au-197 (1166)	1.0 - 5	2.0 + 7	Private communication (1967) Private communication (1972)	BNW BNL	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); N, P (I); N, Alpha (I); β (I); KSI (I); GAMMA (I); RRP (R); SRP (R); Decay chain data (G)	1/72

Pb (1136)	1.0 - 5	2.0 + 7	To be published (ORNL) Fu, Perey	ORL	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); Photon production (I, A, E)	7/71
Th-232 (1117)	1.0 - 5	1.5 + 7	BAW-317 (1970)	B+W	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G)	11/66
Pa-233 (1119)	1.0 - 5	1.5 + 7	Private communication (1970)	BET	Total (I); Elastic (I, A); Non-Elastic (I); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	1/70
U-233 (1110)	1.0 - 5	1.5 + 7	WAPD-TM-691 (1969)	BET	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); $\bar{\nu}$ (G)	3/71
U-233 (RSFP) (1042)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for rapid saturation fission products	12/66
U-233 (SSFP) (1066)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for slow saturation fission products	12/66
U-233 (NSFP) (1067)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for non-saturation fission products	12/66

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV) Minimum	Energy range (eV) Maximum	Reference, source	Laboratory code	Quantity, data type	Evaluation date
U-234 (1043)	1.0 - 5	1.5 + 7	GA-8135 (1967)	GGA	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	1/67
U-235 (1157)	1.0 - 5	1.5 + 7	BNWL-1586 (1971) ANCR-1044 (1971)	AI BNW ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G); Decay chain data (G)	8/71
U-235 (RSFP) (1045)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for rapid saturation fission products	12/66
U-235 (SSFP) (1068)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for slow saturation fission products	12/66

U-235 (NSFP) (1069)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for non-saturation fission products	12/66
U-238 (1158)	1.0 - 5	1.5 + 7	To be published (WARD) Pitterle, Durston	WEW	Total (I); Elastic (I, A); Non-elastic (I); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G)	8/71
Pu-238 (1050)	1.0 - 5	1.5 + 7	NAA-SR-12271 (1967)	AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	5/67
Pu-239 (1159)	1.0 - 5	2.0 + 7	ANCR-1045 (1971) BNWL-1586 (1971)	GEB BNW ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); N, P (I); N, D (I); N, T (I); N, Alpha (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G); Decay chain data (G)	8/71
Pu-239 (RSFP) (1052)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for rapid saturation fission products	12/66

TABLE IV. (cont.)

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Pu-239 (SSFP) (1070)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for slow saturation fission products	12/66
Pu-239 (NSFP) (1071)	1.0 - 5	1.5 + 7	BAW-320 (1966)	B+W	Absorption (I) for non-saturation fission products	12/66
Pu-240 (1105)	1.0 - 5	1.5 + 7	Private communication CSEWG (1969)	GGA BNW	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	9/69
Pu-241 (1106)	1.0 - 5	1.5 + 7	Private communication (1969)	BNL AI	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G); Decay chain data (G)	11/69

Pu-242 (1161)	1.0 - 5	1.5 + 7	NAA-SR-12271 (1967)	AI ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	8/71
Am-241 (1056)	1.0 - 5	1.5 + 7	Private communication (1966)	ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G); Decay chain data (G)	11/66
Am-243 (1057)	1.0 - 5	1.5 + 7	Private communication (1966)	ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Yield (G); Decay chain data (G)	11/66
Cm-244 (1162)	1.0 - 5	1.5 + 7	NAA-SR-12271 (1967)	AI ANC	Total (I); Elastic (I, A); Inelastic (I, A, E); N, 2N (I, A, E); N, 3N (I, A, E); N, Fission (I, A, E); N, Gamma (I); $\bar{\mu}$ (I); KSI (I); GAMMA (I); RRP (R); SRP (R); $\bar{\nu}$ (G); Decay chain data (G)	8/71

TABLE V. LAWRENCE LIVERMORE LABORATORY EVALUATED NUCLEAR DATA LIBRARY (LLL)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
H-1	1.0 - 4	2.0 + 7	Evaluation by Stewart	LAS	Elastic (I, A) N, Gamma (I)	24/ 2/71 24/ 2/71
D-2	2.5 - 2	1.5 + 7	Evaluation primarily by Stewart, partly by Howerton	LAS + LLL	Elastic (I) Elastic (A); N, Gamma (I) N, 2N (I) N, 2N (E)	27/ 6/65 16/ 6/64 18/12/65 8/ 6/64
T-3	2.5 - 2	1.5 + 7	Evaluation primarily by Stewart, partly by Howerton	LAS + LLL	Elastic (I, A) N, 2N (I) N, 2N (E)	18/12/65 18/12/64 8/ 6/64
He-3	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, P (I) N, D (I)	15/ 5/67 4/ 8/64 18/12/70
He-4	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I) Elastic (A)	18/12/65 1/ 8/69
Li	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	N, T (I) N, Gamma (I)	8/ 9/70 8/ 9/70

Li-6	1.0 - 2	2.0 + 7	Evaluation primarily by Howerton, partly by Hansen	LLL	Elastic (I)	1/ 4/71
					Elastic (A)	15/10/70
					N, 2N (I, E)	15/10/70
					N, N' D (I)	1/ 4/71
					N, N' D (A, E)	15/10/70
					N, N' D (2.18 MeV level) (I)	1/ 4/71
					N, N' D (2.18 MeV level) (A)	19/ 4/71
					N, P (I); N, T (I)	1/ 4/71
					N, Gamma (I)	8/ 9/70
Li-7	2.5 - 2	1.5 + 7	Evaluation primarily by Parker	ALD	Elastic (I)	3/11/64
					Elastic (A)	8/ 6/64
					N, N' Gamma (I)	18/12/65
					N, N' Gamma (E)	8/ 6/64
					N, 2N (I); N, Gamma (I)	18/12/65
					N, 2N (E); N, N' T (A, E)	8/ 6/64
					N, N' T (I)	18/12/65
Be-7	1.0 - 4	2.0 + 7	Evaluation primarily by Parker, partly by Barr	ALD + LAS	N, P (I)	16/ 2/70
Be-9	2.5 - 2	1.5 + 7	Evaluation primarily by Perkins, Howerton	LLL	Elastic (I, A)	1/ 8/71
					N, 2N (I, E)	1/ 8/71
					N, 2N (A)	8/ 6/64
					N, P (I); N, T (I); N, D (I)	1/ 8/71
					N, Alpha (I)	28/ 7/71
					N, Gamma (I)	1/ 8/71
B	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I)	23/ 8/66
					Elastic (A)	8/ 6/64
					N, N' Gamma (I); N, 2N (I)	18/12/65
					N, N' Gamma (E); N, 2N (E)	8/ 6/64
					N, T 2 Alpha (I)	15/10/70
					N, Alpha (I)	8/ 9/70

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
B-10	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A)	8/ 9/70
					N, N' Gamma (I, A, E)	8/ 9/70
					N, 2N (I, E); N, Alpha (I)	8/ 9/70
					N, N' D 2 Alpha (I, E)	8/ 9/70
					N, T 2 Alpha (I)	8/ 9/70
					N, Gamma (I)	8/ 9/70
C-12	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A)	15/ 6/70
					N, N' Gamma (I, A)	15/ 6/70
					N, N' 3 Alpha (I, E); N, P (I)	15/ 6/70
					N, D (I); N, Alpha (I)	15/ 6/70
N-14	1.0 - 4	2.0 + 7	Evaluation of the cross-sections by Young, of the angular distributions by Howerton	LAS + LLL	Elastic (I, A)	29/ 6/70
					N, N' Gamma (I, A)	29/ 6/70
					N, 2N (I, E); N, N' P (I, E)	29/ 6/70
					N, P (I); N, D (I); N, T (I)	29/ 6/70
					N, Alpha (I); N, Gamma (I)	29/ 6/70
O-16	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I)	22/ 4/70
					Elastic (A)	27/ 5/70
					N, N' Gamma (I); N, N' Alpha (I)	21/ 8/70
					N, N' Gamma (A); N, 2N (I)	22/ 1/70
					N, N' P (I, E)	22/ 4/70
					N, 2N (E)	17/ 3/70
					N, N' Alpha (E)	21/ 8/70
					N, Alpha (I)	22/ 1/70

F-19	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I) Elastic (A) N, N' Gamma (I); N, 2N (I) N, P (I); N, D (I); N, T (I) N, Alpha (I); N, Gamma (I) N, N' Gamma (F) N, 2N (E)	13/ 6/65 18/12/65 18/12/65 18/12/65 18/12/65 13/ 3/70 18/12/65
Na-23	1.0 - 2	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I); N, 2N (I) Elastic (A) N, N' Gamma (I) N, N' Gamma (A, E); N, 2N (E) N, N' P (I, E); N, N' Alpha (I, E) N, P (I); N, Alpha (I); N, Gamma (I)	19/ 8/70 15/10/70 13/10/70 19/ 8/70 19/ 8/70 19/ 8/70
Mg	1.0 - 2	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I); N, 2N (I) N, N' Gamma (I) Elastic (A); N, N' Gamma (A) N, N' Gamma (E) N, 2N (E) N, P (I) N, Gamma (I)	5/ 1/71 5/ 1/71 8/12/70 14/12/70 2/12/70 8/12/70 8/ 1/71
Al-27	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, 2N (I); N, N' Gamma (I) N, P (I); N, Alpha (I) N, 2N (E); N, N' Gamma (E) N, Gamma (I)	8/ 6/64 18/12/65 18/12/65 2/12/70 9/ 6/70

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Si	1.0 - 5	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I); N, N' Gamma (I, A) Elastic (A) N, N' Gamma (E) N, 2N (I, E); N, N' P (I, E) N, P (I); N, D (I); N, Alpha (I) N, Gamma (I)	19/ 8/70 25/ 5/70 15/10/70 19/ 8/70 19/ 8/70 19/ 8/70
P-31	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, 2N (I); N, N' Gamma (I) N, 2N (E); N, N' Gamma (E) N, P (I); N, Gamma (I)	8/ 6/64 18/12/65 8/ 6/64 18/12/65
S-32	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, P (I); N, T (I); N, Alpha (I) N, Gamma (I); N, N' Gamma (I, E)	1/ 1/66 1/ 1/66 1/ 1/66
Cl	1.0 - 2	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I, A); N, N' Gamma (I, A) N, N' Gamma (E) N, 2N (I, E); N, N' P (I, E) N, N'Alpha (I, E); N, P (I) N, Alpha (I); N, Gamma (I)	19/ 8/70 15/10/70 19/ 8/70 19/ 8/70 19/ 8/70
Ar	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (I, E) N, 2N (I, E); N, Gamma (I)	8/ 6/64 8/ 6/64

K	1.0 - 2	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I) Elastic (A) N, N' Gamma (I, A, E); N, 2N (I, E) N, N' P (I, E); N, N' Alpha (I, E) N, P (I); N, Alpha (I); N, Gamma (I)	25/ 5/71 10/ 8/70 19/ 8/70 19/ 8/70 19/ 8/70
Ca	1.0 - 2	2.0 + 7	Evaluation primarily by Drake, modified by Howerton	GGA + LLL	Elastic (I, A); N, N' Gamma (I, A, E) N, 2N (I, E); N, N' P (I, E) N, N' Alpha (I, E); N, P (I) N, Alpha (I); N, Gamma (I)	19/ 8/70 19/ 8/70 19/ 8/70 19/ 8/70
Sc-45	2.5 - 2	2.0 + 7	Evaluation by Howerton	LLL	N, Gamma (I)	8/ 3/71
Ti	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, N' Gamma (I); N, 2N (I) N, N' Gamma (E); N, 2N (E) N, Gamma (I)	8/ 6/64 18/12/65 1/12/70 18/12/65
Mn-55	2.5 - 2	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, 2N (I, E); N, Gamma (I) N, Alpha (I); N, N' Gamma (E) N, N' Gamma (I)	8/ 6/64 9/ 6/70 8/ 6/64 18/12/65
Fe	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, P (I) N, N' Gamma (I, E); N, 2N (I, E) N, Gamma (I)	13/11/70 20/10/70 10/ 9/70

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Fe-54	1.0 + 3	2.0 + 7	Evaluation by Howerton	LLL	N, Gamma (I)	27/ 5/70
Fe-56	1.1 + 7	2.0 + 7	Evaluation by Howerton	LLL	N, 2N (I)	27/ 5/70
Fe-58	1.0 + 3	2.0 + 7	Evaluation by Howerton	LLL	N, Gamma (I)	27/ 5/70
Ni-58	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (I, E) N, 2N (I, E); N, P (I); N, D (I) N, Alpha (I) N, Gamma (I)	1/ 8/69 1/ 8/69 1/ 8/69 15/ 5/70
Cu	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, 2N (E) N, N' Gamma (E) N, N' Gamma (I); N, 2N (I) N, P (I); N, Gamma (I)	8/ 6/64 8/ 6/64 18/12/65 18/12/65
Ga	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	8/ 6/64 8/ 8/64 18/12/65 18/12/65
Nb-93	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	8/ 6/64 8/ 6/64 18/12/65 18/12/65

Mo	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	8/ 6/64 8/ 6/64 18/12/65 18/12/65
Cd	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	1/ 8/69 1/ 8/69 26/ 4/67 26/ 4/67
Sn	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A) N, N' Gamma (E); N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	1/ 8/69 20/ 5/65 18/12/65 18/12/65
Ba-138	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, P (I) N, Alpha (I); N, Gamma (I) N, N' Gamma (I, E); N, 2N (I, E)	1/ 8/69 1/ 8/69 10/ 4/69
Eu	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I) N, N' Gamma (I); N, 2N (I) Elastic (A); N, N' Gamma (E) N, 2N (E); N, 3N (E) N, 3N (I) N, Gamma (I)	18/10/65 18/12/65 8/ 6/64 8/ 6/64 18/12/65 22/ 4/70
Gd	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E); N, 3N (E) N, N' Gamma (I); N, 2N (I) N, 3N (I); N, Gamma (I)	8/ 6/64 8/ 6/64 18/12/65 18/12/65

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Ho-165	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I) Elastic (A); N, N' Gamma (E) N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I)	11/10/66 28/ 9/66 28/ 9/66 26/ 9/66 22/ 4/70
Ta-181	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, Gamma (I) N, N' Gamma (I); N, 2N (I) Elastic (A); N, N' Gamma (E) N, 2N (E)	3/ 4/70 18/12/65 8/ 6/64 8/ 6/64
W	2.5 - 2	2.5 + 7	Evaluation by Howerton	LLL	Elastic (I); N, N' Gamma (I) N, 2N (I, E); N, 3N (I, E) N, N' Gamma (E); N, Gamma (E) Elastic (A)	13/ 3/70 13/ 3/70 13/ 3/70 8/ 6/64
Au-197	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, N' Gamma (I); N, 2N (I) N, P (I); N, Gamma (I) N, N' Gamma (E); N, 2N (E)	8/ 6/64 18/12/65 18/12/65 8/ 6/64
Ph	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I) N, 2N (E)	8/ 6/64 18/12/65 18/12/65 8/ 6/64

Th-232	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, N' Gamma (I)	7/11/67
					Elastic (A); N, N' Gamma (E)	11/ 7/67
					N, 2N (I); N, 3N (I); N, Gamma (I)	7/11/67
					N, 2N (E); N, 3N (E)	11/ 7/67
					N, Fission (I)	1/ 9/70
					N, Fission (E)	8/ 9/71
					$\bar{\nu}$ (G)	7/11/67
U-233	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, N' Gamma (I)	13/ 3/70
					N, 2N (I); N, 3N (I); N, Fission (I)	13/ 3/70
					Elastic (A); N, N' Gamma (E)	8/ 6/64
					N, 2N (E); N, 3N (E)	8/ 6/64
					N, Fission (E)	8/ 9/71
					N, Gamma (I)	22/ 1/70
					$\bar{\nu}$ (G)	7/12/70
U-234	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E)	8/ 6/64
					N, 2N (E); N, 3N (E)	8/ 6/64
					N, N' Gamma (I); N, 2N (I)	18/12/65
					N, 3N (I); N, Fission (I)	18/12/65
					N, Gamma (I)	22/ 1/70
					N, Fission (E)	8/ 9/71
					$\bar{\nu}$ (G)	18/12/65
U-235	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, 4N (E)	4/ 5/71
					Elastic (A)	8/ 6/64
					N, N' Gamma (I, E); N, 2N (I, E)	22/ 9/71
					N, 3N (I, E); $\bar{\nu}$ (G)	22/ 9/71
					N, 4N (I); N, Fission (E)	4/ 5/71
					N, Fission (I)	8/ 1/71
					N, Gamma (I)	22/ 1/70

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
U-236	1.0 - 4	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E)	8/ 6/64
					N, 2N (E); N, 3N (E); $\bar{\nu}$ (G)	8/ 6/64
					N, N' Gamma (I); N, 2N (I)	18/12/65
					N, 3N (I); N, Fission (I)	18/12/65
					N, Fission (E)	8/ 9/71
					N, Gamma (I)	22/ 1/70
U-237	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E)	8/ 6/64
					N, 2N (E); N, 3N (E)	8/ 6/64
					N, N' Gamma (I); N, 2N (I)	18/12/65
					N, 3N (I); N, Fission (I)	18/12/65
					N, Fission (E)	8/ 9/71
					N, Gamma (I)	22/ 1/70
					$\bar{\nu}$ (G)	6/ 8/64
U-238	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, 2N (I, E)	12/ 3/71
					N, N' Gamma (I, A, E); N, 3N (I, E)	12/ 3/71
					N, 4N (I, E); N, Fission (I, E)	12/ 3/71
					N, Gamma (I)	22/ 1/70
					Elastic (A)	1/ 8/69
					$\bar{\nu}$ (G)	17/ 1/70
U-239	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, 2N (I); N, 3N (I)	16/11/66
					N, N' Gamma (I); N, Fission (I)	16/11/66
					Elastic (A); N, 2N (E); N, 3N (E)	8/ 6/64
					N, N' Gamma (E); $\bar{\nu}$ (G)	8/ 6/64
					N, Fission (E)	8/ 9/71
					N, Gamma (I)	23/ 9/70

U-240	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (I, E) N, 2N (I, E); N, 3N (I, E) N, Fission (I); $\bar{\nu}$ (G) N, Fission (E) N, Gamma (I)	8/ 6/64 8/ 6/64 8/ 6/64 8/ 9/71 22/ 1/70
Np-237	6.6 - 0	2.0 + 7	Evaluation by Howerton, Lindner	LLL	N, 2N (I)	13/ 7/70
Pu-238	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I); N, Fission (I) N, Gamma (I) Elastic (A); N, N' Gamma (E) N, 2N (E); N, 3N (E); $\bar{\nu}$ (G) N, Fission (E) N, N' Gamma (I); N, 2N (I); N, 3N (I)	22/ 1/70 22/ 1/70 8/ 6/64 8/ 6/64 8/ 9/71 22/ 9/66
Pu-239	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I) N, N' Gamma (I); N, 2N (I); N, 3N (I) N, 4N (I) N, Fission (I) N, Gamma (I) Elastic (A) N, N' Gamma (A); N, 3N (E); N, 4N (E) N, N' Gamma (E); N, 2N (E) N, Fission (E) $\bar{\nu}$ (G)	16/12/70 10/12/70 27/ 5/70 10/12/70 13/11/70 8/ 6/64 5/ 1/71 18/ 3/71 15/12/70 14/12/70

TABLE V. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Pu-240	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E); N, 3N (E); $\bar{\nu}$ (G) N, N' Gamma (I); N, 2N (I) N, 3N (I) N, Fission (I); N, Gamma (I) N, Fission (E)	8/ 6/64 8/ 6/64 18/12/65 18/12/65 22/ 1/70 8/ 9/71
Pu-241	1.0 - 4	2.0 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, N' Gamma (E) N, 2N (E); N, 3N (E); $\bar{\nu}$ (G) N, Gamma (I); N, 2N (I) N, Fission (I); N, 3N (I) N, N' Gamma (I) N, Fission (E)	8/ 6/64 8/ 6/64 18/12/65 18/12/65 18/12/65 8/ 9/71
Am-242	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, Gamma (I) N, N' Gamma (I, E); N, 2N (I, E) N, 3N (I, E); N, Fission (I) N, Fission (E) $\bar{\nu}$ (G)	1/ 8/69 1/ 8/69 1/ 8/69 8/ 9/71 8/ 6/64
FP-120 (Crude fission product)	2.5 - 2	1.5 + 7	Evaluation by Howerton	LLL	Elastic (I, A); N, 2N (E) N, N' Gamma (I); N, 2N (I) N, Gamma (I) N, N' Gamma (E)	8/ 6/64 18/12/65 18/12/65 8/ 6/64

TABLE VI. SWEDISH EVALUATED NEUTRON DATA LIBRARY (SPENG)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date	Comments
	Minimum	Maximum					
H-1	2.0 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I)	2/67	H in H_2O. Believed adequate
D-2	1.0 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I); N, 2N (I)	4/67	D in D_2O. Believed adequate
He-4	1.0 - 2	1.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, Gamma (I)	10/67	
Li-6	1.0 - 4	1.5 + 7	Unpublished	AE	Total (I); Elastic (I); Inelastic (I); N, Gamma (I)	3/70	
Li-7	1.0 - 4	1.5 + 7	Unpublished	AE	Total (I); Elastic (I); Inelastic (I); N, Gamma (I)	5/70	
Be-9	2.5 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I)	11/69	
B	3.7 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I); Inelastic (I, 1-4)	9/69	
B-10	2.0 - 3	1.5 + 7	Unpublished	AE	Total (I); Elastic (I); Inelastic (I); N, Gamma (I)	2/70	
B-11	1.0 - 4	1.5 + 7	Unpublished	AE	Total (I); Elastic (I); Inelastic (I, 1-3); N, Gamma (I)	3/70	

TABLE VI. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date	Comments
	Minimum	Maximum					
C	1.0 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I, A); Inelastic (I); N, Gamma (I)	12/67	
O	1.0 - 3	1.6 + 7	KAPL-M-6452	KAP	Total (I); Elastic (I, A); N, Gamma (I); Inelastic (I, 1)	/67	ENDF/B-1 file
F	1.0 - 2	1.0 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I); Inelastic (I, 1-3)	/62	Probably not adequate
Na	1.0 - 2	2.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, Gamma (I); Inelastic (I, 1-7)	2/67	
Al	1.0 - 2	1.8 + 7	AE-RFA-409	AE	Total (I); Elastic (I, A); N, Gamma (I); Inelastic (I, 1-8)	3/63	Old file
Si	3.7 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I); Inelastic (I, 1-3)	5/65	
Cr	1.0 - 2	1.6 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, Gamma (I); Inelastic (I, 1-10)	2/65	
Mn	3.8 - 2	1.8 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-7)	9/68	Partly from ENDF/B-I
Fe	1.0 - 3	1.8 + 7	KFK-750	KFK	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-10)	/68	Studsvik evaluation above 10 MeV

Ni	1.0 - 2	1.6 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, P (I); N, Gamma (I); Inelastic (I, 1-10)	2/66	Poor accuracy of the (n, p) data
Cu	1.0 - 2	2.0 + 7	AE-RFR-597	AE	Total (I); Elastic (I); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-10)	3/67	
Pu-239	1.0 - 6	1.6 + 7	AE-RFR-695	AE	Total (I); Elastic (I, A); N, Gamma (I); N, Fission (I); Inelastic (I, 1-10, C); N, 2N (I, E)	2/68	
Pu-240	2.5 - 4	1.6 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Fission (I); N, Gamma (I); Inelastic (I, 1-10, C)	4/68	
Pu-241	9.1 - 3	1.6 + 7	AWRE-O-101/64	ALD	Total (I); Elastic (I, A); N, Fission (I); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-10, C)	/64	
Fission products	1.0 - 3	1.0 + 7	Unpublished	AE	Total (I); Elastic (I); N, Gamma (I); Inelastic (I)	8/68	

TABLE VII. KARLSRUHE EVALUATED DATA FILE (KEDAK)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date	Comments
	Minimum	Maximum					
H-1 [a]	1.0 - 3	1.0 + 7	KFK-120/I (1966)	KFK	Total (I); Elastic (I); $\bar{\mu}$ (I); TR (I); N, Gamma (I); Non-elastic (I); Absorption (I)	/66	Data given for H bound in H_2 and in H_2O. Next release will contain revised file
D-2	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); TR (I); Non-elastic (I); N, 2N (I); Absorption (I); N, Gamma (I)	/66	
He-3	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I); TR (I); Non-elastic (I); Absorption (I); N, P (I)	/66	
He-4	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); TR (I); $\bar{\mu}$ (I)	/66	
C-12 [a]	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Absorption (I); N, Gamma (I); N, Alpha (I); TR (I); RRP (R); Inelastic (I, 1, 2)	/66	Next release will contain extended file up to 15 MeV
N	1.0 + 5	1.6 + 7	KFK-1340 (1971)	KFK	Elastic (A)	/64	
Zr	1.0 - 2	1.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, E)	10/68	
Mo	1.0 - 2	1.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, Gamma (I); Inelastic (I, E)	12/68	

Er	1.0 - 1	1.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, E)	4/70	
Ta	1.0 - 4	1.5 + 7	Unpublished	WIN	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-8)	7/66	This file is part of the UKNDL with Data File Number 328
W-186	1.0 - 2	2.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-6)	6/67	
Au-197	1.0 - 2	2.0 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Gamma (I); Inelastic (I, 1-8)	8/67	
U-235	1.0 - 6	1.6 + 7	Unpublished	AE	Total (I); Elastic (I, A); N, Gamma (I); N, Fission (I); Inelastic (I, 1-10, C); N, 2N (I, E)	10/68	
U-238	1.0 - 3	1.8 + 7	AE-RFR-695	AE	Total (I); Elastic (I, A); N, 2N (I, E); N, Fission (I); N, Gamma (I); Inelastic (I, 1-10, C)	2/68	
O-16	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); TR (I); RRP (R); N, Gamma (I); N, Alpha (I); Absorption (I); Non-elastic (I); Inelastic (I, 1-4)	/66	
Na-23 [a]	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); TR (I); RRP (R); SRP (R); N, Gamma (I); N, P (I); N, Alpha (I); Absorption (I); N, 2N (I); Non-elastic (I); Inelastic (I, 1-7)	/66	Next release will contain revised data above 1 MeV and in the resonance region

[a] Materials for which the next release of the KEDAK file will contain revised or extended data.

TABLE VII. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date	Comments
	Minimum	Maximum					
Al-27	6.0 - 4	1.0 + 7	AEEW-M-445 KFK-1340 (1971)	KFK + WIN	Total (I); Elastic (I, A); $\bar{\mu}$ (I); TR (I); RRP (R); SRP (R); Non-elastic (I); N, 2N (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); Inelastic (I, 1-9)	/67	
Cr[a]	1.0 - 3	1.5 + 7	KFK-120/I (1966) KFK-750 (1968) Unpublished internal report	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); N, 2N (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); RRP (R); SRP (R); Inelastic (I, 1-8)	/70	Revised data above 1 MeV and isotopic data for threshold reactions will be included in next release
Fe[a]	1.0 - 3	1.5 + 7	KFK-120/I (1966) KFK-750 (1968) Unpublished internal report	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); N, 2N (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); RRP (R); SRP (R); Inelastic (I, 1-10); Non-elastic (I)	/70	Revised data above 1 MeV and isotopic data for threshold reactions will be included in next release
Ni[a]	1.0 - 3	1.5 + 7	KFK-120/I (1966) KFK-750 (1968) Unpublished internal report	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); N, 2N (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); RRP (R); SRP (R); Inelastic (I, 1-12)	/70	Revised data above 1 MeV and isotopic data for threshold reactions will be included in next release

Mo [a]	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Absorption (I); N, 2N (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); RRP (R); SRP (R); Inelastic (I, 1-8)	/66	Revised data above 1 MeV and isotopic data for threshold reactions will be included in next release
Cd	1.0 - 3	1.5 + 7	KFK-1080 (1969)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); N, 2N (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); RRP (R); Inelastic (I, 1-4)	/69	
U-235 [a]	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Inelastic (I, 1-10); N, 2N (I); N, Fission (I); $\bar{\nu}$ (G); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); α (G); η (G); FS (G); RRP (R); SRP (R)	/66	Data will be revised before next release
U-238	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Inelastic (I, 1-24); N, 2N (I); N, Fission (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); TR (I); $\bar{\nu}$ (G); RRP (R); SRP (R); FS (G)	/66	

[a] Materials for which the next release of the KEDAK file will contain revised or extended data.

TABLE VII. (cont.)

Nuclide	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date	Comments
	Minimum	Maximum					
Pu-239	1.0 - 3	1.0 + 7	KFK-120/I (1966) KFK-750 (1968) KFK-1340 (1971)	KFK	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Inelastic (I, 1-7); N, 2N (I); N, Fission (I); TR (I); Absorption (I); N, Gamma (I); N, P (I); N, Alpha (I); $\bar{\nu}$ (G); η (G); α (G); FS (G); RRP (R); SRP (R)	/70	
Pu-240 [a]	1.0 - 3	1.0 + 7	Symp. Karlsruhe [b] KFK-1340 (1971)	KFK + SOR	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); Inelastic (I, 1-8); N, 2N (I); N, Fission (I); Absorption (I); N, Gamma (I); $\bar{\nu}$ (G); α (G); η (G); TR (I); RRP (R); SRP (R); FS (G)	/69	Complete re-evaluation will be contained in next release
Pu-241	1.0 - 3	1.0 + 7	Symp. Karlsruhe [b]	KFK + SOR	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); N, 2N (I); N, Fission (I); N, Gamma (I); Inelastic (I, 1-8); Absorption (I); α (G); $\bar{\nu}$ (G); TR (I); FS (G); η (G); RRP (R); SRP (R)	/67	
Pu-242	1.0 - 3	1.0 + 7	Symp. Karlsruhe [b]	KFK + SOR	Total (I); Elastic (I, A); $\bar{\mu}$ (I); Non-elastic (I); N, 2N (I); N, Gamma (I); N, Fission (I); Absorption (I); α (G); TR (I); FS (G); η (G); $\bar{\nu}$ (G); RRP (R); SRP (R)	/67	

[a] Materials for which the next release of the KEDAK file will contain revised or extended data.
[b] YIFTAH, S. et al., "Basic nuclear data for the higher plutonium isotopes", *Fast Reactor Physics* (Proc. Symp. Karlsruhe, 1967) 1, IAEA, Vienna (1968) 123.

TABLE VIII. ITALIAN FISSION PRODUCT LIBRARY

For each of the fission product nuclei listed the capture cross-section has been evaluated from 1 keV to 10 MeV

A. The capture cross-sections of the following nuclei have been evaluated on a purely theoretical basis and the cross-sections for these nuclei are believed to have an uncertainty of about 50% on the average

Ge-70	Kr-84	Pd-106	Xe-126	Ba-137	Sm-147
Ge-72	Kr-86	Cd-106	Xe-128	La-138	Sm-148
Ge-73	Sr-84	Cd-108	Xe-129	La-139	Sm-149
Ge-76	Sr-88	Cd-110	Xe-130	Ce-136	Sm-150
Se-74	Mo-92	Cd-111	Xe-131	Ce-138	Gd-152
Se-76	Mo-94	Cd-112	Xe-132	Ce-140	Gd-154
Se-77	Ru-98	Cd-113	Xe-134	Ce-142	Gd-155
Se-78	Ru-99	Sn-112	Xe-136	Nd-142	Gd-156
Se-82	Ru-100	Sn-114	Ba-130	Nd-143	Gd-157
Kr-78	Ru-101	Sn-115	Ba-132	Nd-144	Dy-156
Kr-80	Pd-102	Sn-124	Ba-134	Nd-145	Dy-158
Kr-82	Pd-104	Te-120	Ba-135	Nd-146	Dy-160
Kr-83	Pd-105	Xe-124	Ba-136	Sm-144	Dy-164

B. The capture cross-sections of the following nuclei have been estimated very roughly

Zr-92	Cd-114	Sn-122	Sb-123	Sm-154	Eu-153
Ru-104	Cd-116	Sb-121	Ba-138	Eu-151	Gd-160

C. The capture cross-sections of the following nuclei are believed adequate, for fission product calculations only, in the ~10 keV - 1 MeV energy region

Ge-74	Sr-86	Pd-108	Sn-117	Te-123	Gd-158
Se-80	Sr-87	Pd-110	Sn-118	Nd-148	Dy-161
Br-79	Y-89	In-113	Sn-119	Nd-150	Dy-162
Br-81	Ru-96	In-115	Sn-120	Sm-152	Dy-163
Rb-87	Ru-102	Sn-116	Te-120		

D. The capture cross-sections of the following nuclei are believed adequate, for fission product calculations only

As-75	Zr-94	Mo-96	Rh-103	Te-125	I-127
Rb-85	Zr-96	Mo-97	Ag-107	Te-126	Cs-133
Zr-90	Nb-93	Mo-98	Ag-109	Te-128	Pr-141
Zr-91	Mo-95	Mo-100	Te-124	Te-130	Tb-159

TABLE IX. FRENCH EVALUATIONS INCORPORATED INTO THE UKNDL

Nuclide (Data file No.)	Energy range (eV)		Reference, source	Laboratory code	Quantity, data type	Evaluation date
	Minimum	Maximum				
Cr (45D)	1.0 - 4	1.5 + 7	PNR/SEPR 65.041	CAD	Total (I); Elastic (I, A); Non-elastic (I); N, 2N (I, E); N, Gamma (I); N, P (I); Inelastic (I, 1-8, C)	11/65
Ni (46)	1.0 - 4	1.5 + 7	PNR/SEPR 65.010	CAD	Total (I); Elastic (I, A); Non-elastic (I); N, 2N (I, E); N, Gamma (I); N, P (I); N, Alpha (I); Inelastic (I, 1-8, C)	3/65
U-238 (401D)	1.0 - 4	1.5 + 7	PNR (SEPR) R 025	CAD	Total (I); Elastic (I, A); $\bar{\nu}$ (G); Non-elastic (I); N, 2N (I, E); N, 3N (I, E); N, Fission (I, E); N, Gamma (I); Inelastic (I, 1-10, C)	2/68
Pu-239 (404)	2.5 - 8	1.5 + 7	CEA-N-1484	SAC	Total (I); Elastic (I, A); $\bar{\nu}$ (G); Non-elastic (I); Inelastic (I, 1-10, C); N, 2N (I, E); N, 3N (I, E); N, Fission (I, E); FS (G); N, Gamma (I)	/71
Pu-240 (402)	2.5 - 8	1.5 + 7	CEA-N-1273	SAC	Total (I); Elastic (I, A); N, 2N (I, E); N, 3N (I, E); N, Fission (I, E); N, Gamma (I); FS (G); $\bar{\nu}$ (G); Inelastic (I, 1-5, C)	/70
Pu-241 (403)	2.5 - 8	1.5 + 7	SMPNF-806-70	SAC	Total (I); Elastic (I, A); N, 2N (I, E); N, 3N (I, E); N, Fission (I, E); N, Gamma (I); FS (G); $\bar{\nu}$ (G); Inelastic (I, 1-4, C)	/70

TABLE X. AUSTRALIAN FISSION PRODUCT LIBRARY

For each of the 192 fission product nuclei listed (184 ground states and 8 isomeric states) the total, elastic non-elastic, capture, inelastic and transport cross-sections have been evaluated from 10^{-3} eV to 15 MeV

Zn-72	Y-91	Pd-109	Sb-128	Ba-136	Sm-152
Ga-72	Y-93	Pd-110	Te-122	Ba-137	Sm-153
Ge-72	Zr-90	Pd-112	Te-123	Ba-138	Sm-154
Ge-73	Zr-91	Ag-109	Te-124	Ba-140	Sm-156
Ge-74	Zr-92	Ag-111	Te-125	La-139	Eu-153
Ge-76	Zr-93	Cd-110	Te-126	La-140	Eu-154
Ge-77	Zr-94	Cd-111	Te-127	Ce-140	Eu-155
As-75	Zr-95	Cd-112	Te-128	Ce-141	Eu-156
As-76	Zr-96	Cd-113	Te-129	Ce-142	Eu-157
As-77	Zr-97	Cd-114	Te-130	Ce-143	Gd-155
Se-76	Nb-95	Cd-115	Te-131	Ce-144	Gd-156
Se-77	Mo-95	Cd-116	Te-132	Pr-141	Gd-157
Se-78	Mo-96	In-115	I-127	Pr-142	Gd-158
Se-79	Mo-97	Sn-115	I-129	Pr-143	Gd-159
Se-80	Mo-98	Sn-116	I-130	Pr-145	Gd-160
Br-81	Mo-99	Sn-117	I-131	Nd-142	Tb-159
Br-82	Mo-100	Sn-118	I-133	Nd-143	Tb-160
Kr-82	Tc-99	Sn-119	I-135	Nd-144	Tb-161
Kr-83	Ru-100	Sn-120	Xe-128	Nd-145	Dy-160
Kr-84	Ru-101	Sn-121	Xe-130	Nd-146	Dy-161
Kr-85	Ru-102	Sn-122	Xe-131	Nd-147	Dy-162
Kr-86	Ru-103	Sn-123	Xe-132	Nd-148	Dy-163
Rb-85	Ru-104	Sn-124	Xe-133	Nd-150	Dy-164
Rb-86	Ru-105	Sn-125	Xe-134	Pm-147	Ho-165
Rb-87	Ru-106	Sn-126	Xe-135	Pm-148	Tc-799 [a]
Sr-86	Rh-103	Sb-121	Xe-136	Pm-149	Cd-815 [a]
Sr-88	Rh-105	Sb-122	Cs-133	Pm-151	Te-823 [a]
Sr-89	Pd-104	Sb-123	Cs-134	Sm-147	Te-825 [a]
Sr-90	Pd-105	Sb-124	Cs-135	Sm-148	Te-827 [a]
Sr-91	Pd-106	Sb-125	Cs-136	Sm-149	Te-829 [a]
Y-89	Pd-107	Sb-126	Cs-137	Sm-150	Te-831 [a]
Y-90	Pd-108	Sb-127	Ba-134	Sm-151	Pm-848 [a]

[a] For these eight isomeric states the mass number has been defined by adding 700 to the mass number.

TABLE XI. USSR CATALOGUE OF 26-GROUP CONSTANTS: GROUP CONSTANTS FOR ALL REACTIONS FOR EACH NUCLIDE LISTED

Nuclide	Reference/year [a]		Date of inclusion in the Catalogue: M-26 [1] [a] Magnetic tape	Date of inclusion in the Catalogue: M-26 [1] [a] Punched cards	Date of inclusion in the Catalogue: ARAMAKO [2] [a] Magnetic tape
H	[3]	1964	1966		1970
	[4]	1968			
D	[3]	1964			1970
	[4]	1968			
He	[4]	1968			
Li-6	[3]	1964	1966		1970
Li-7	[3]	1964	1966		1970
Be	[3]	1964	1966		1970
	[4]	1968			
B-10	[3]	1964	1966		1970
B-11	[3]	1964	1966		1970
C	[3]	1964	1966		1970
	[4]	1968			

N	[3] [4]	1964 1968	1966		1970
O	[3] [5] [4]	1964 1964 1968	1966 16 May 1968		1970
F	[6]	1966		1967	1970
Na	[3] [4]	1964 1968	1966		1970
Mg	[3]	1964	1966		1970
Al	[3] [4]	1964 1968	1966		1970
Si	[3]	1964	1966		1970
Cl	[6] [4]	1966 1968		1967	1970
K	[3]	1964	1966		1970
Ca	[3]	1964	1966		1970
Ti	[3]	1964	1966		1970
V	[3]	1964	1966		1970

[a] A list of References is given after Table XIV.

TABLE XI. (cont.)

Nuclide	Reference/year [a]	Date of inclusion in the Catalogue		
		M-26 [1] [a]		ARAMAKO [2] [a]
		Magnetic tape	Punched cards	Magnetic tape
Cr	[3] 1964 [4] 1968	1966		1970
Fe	[3] 1964 [7] 1964 [4] 1968	1966 14 May 1969		1970
Steel (1Kh18NaT)	[7] 1964		1967	
Steel (E1-847)	[7] 1964		1967	
Ni	[3] 1964 [7] 1964 [4] 1968	1966 16 May 1968		1970
Cu	[3] 1964	1966		1970
Y	[6] 1966		1967	1970
Zr	[3] 1964	1966		1970
Nb	[3] 1964	1966		1970

Mo	[3]	1964	1966		1970
Ce	[7]	1964		1967	
Sm	[8, 9]	1965			
Eu	[8, 9] [4]	1965 1968		 1970	
Gd	[8, 9] [4]	1965 1968		1970 1970	
Hf	[8, 9] [4]	1965 1968			
Ta	[3]	1964	1966		1970
W	[3]	1964	1966		1970
Re	[3]	1964	1966		1970
Pb	[3]	1964	1966		1970
Bi	[3]	1964	1966		1970
Th-232	[3]	1964	1966		1970
U-233	[3] [7]	1964 1964	1966 26 Feb. 1967		 1970
U-234	[3]	1964	1966		1970

[a] A list of References is given after Table XIV.

TABLE XI. (cont.)

Nuclide	Reference/year [a]	Date of inclusion in the Catalogue		
		M-26 [1] [a]		ARAMAKO [2] [a]
		Magnetic tape	Punched cards	Magnetic tape
U-235	[3] 1964 [4] 1968	1966 23 Feb. 1970		1970
U-236	[3] 1964	1966		1970
U-238	[3] 1964 [4] 1968	1966 23 Feb. 1970		1970
Pu-239	[3] 1964 [7] 1964 [4] 1968	1966 1968 23 Feb. 1970		1970
Pu-240	[3] 1964	1966		1970
Pu-241	[3] 1964	1966		1970
Pu-242	[3] 1964	1966		1970

U-233 fission products	[3] 1964	1966		1970
U-235 fission products	[3] 1964	1966		1970
Pu-239 fission products	[3] 1964	1966		1970

[a] A list of References is given after Table XIV.

TABLE XII. USSR CATALOGUE OF 26-GROUP CONSTANTS: GROUP CONSTANTS FOR SINGLE REACTIONS FOR EACH NUCLIDE LISTED [10, 11][a]

Nuclide	Type of reaction	Evaluation date
He	(n, p)	Sep. 1967
Al-27	(n, α)	May 1966
Si-28	(n, p)	1964
P-31	(n, p)	1964
S-32	(n, p)	1964
Cl-37	(n, γ)	Aug. 1967
Cr-50	(n, γ)	Jul. 1968
Mn-55	(n, γ)	1964
	(n, γ)	Feb. 1969
Fe-54	(n, p)	Oct. 1968
Fe-56	(n, p)	May 1966
Fe-58	(n, γ)	Oct. 1968
Co-59	(n, γ)	Feb. 1969
Ni-58	(n, p)	1964
	(n, p)	Jul. 1967
Cu-63	(n, γ)	1964
Cu-65	(n, γ)	1964
Ag-109	(n, γ)	Feb. 1969
In-115	(n, γ)	1964
Sb-121	(n, γ)	May 1967
Sb-123	(n, γ)	May 1967
I-127	(n, γ)	1964
Eu-153	(n, γ)	Jan. 1969
W-186	(n, γ)	Jul. 1967
Au	(n, γ)	1964
	(n, γ), (n, n) and self-shielding coefficients for T = 300°K	1970
Pa-231	(n, γ)	1968
Pa-233	(n, γ)	1968
Np-237	(n, f)	1964

[a] A list of References is given after Table XIV.

TABLE XIII. USSR CATALOGUE OF 26-GROUP CONSTANTS: FISSION PRODUCT CAPTURE GROUP CONSTANTS [12][a]

Fission fragments
Ge-(70, 72, 73, 74, 76), natural Ge
As-75
Se-(74, 76, 77, 78, 80), natural Se
Br-(79, 81), natural Br
Kr-(78, 80, 82, 83, 84, 86), natural Kr
Rb-(85, 87), natural Rb
Sr-(84, 86, 87, 88), natural Sr
Y-89
Zr-(90, 91, 92, 94, 96), natural Zr
Nb-93
Mo-(92, 94, 95, 96, 97, 98, 100), natural Mo
Ru-(96, 98, 99, 100, 101, 102, 104), natural Ru
Rh-103
Pd-(102, 104, 105, 106, 108, 110), natural Pd
Ag-(107, 109), natural Ag
Cd-(106, 108, 110, 111, 112, 113, 114, 116), natural Cd
In-(113, 115), natural In
Sn-(112, 114, 115, 116, 117, 118, 119, 120, 122, 124), natural Sn
Sb-(121, 123), natural Sb
Te-(120, 122, 123, 124, 125, 126, 128, 130), natural Te
I-127
Xe-(124, 126, 128, 129, 130, 131, 132, 134, 136), natural Xe
Cs-133
Ba-(130, 132, 134, 135, 136, 137, 138), natural Ba
La-(138, 139), natural La
Ce-(136, 138, 140, 142), natural Ce
Pr-141
Nd-(142, 143, 144, 145, 146, 148, 150), natural Nd
Sm-(144, 147, 148, 149, 150, 152, 154), natural Sm
Eu-(151, 153), natural Eu
Gd-(152, 154, 155, 156, 157, 158, 160), natural Gd
Tb-159
Dy-(156, 158, 160, 161, 162, 163, 164), natural Dy

[a] A list of References is given after Table XIV.

Note: For groups 1-14 (i.e. above 1 keV) there are capture cross-section group constants [12] for *all* stable fission fragments and for natural nuclide mixtures of which some nuclides occur among the fission fragments. The group constants of the stable fission fragments were obtained on the basis of Benzi's evaluated data [13].

TABLE XIV. USSR CATALOGUE OF 26-GROUP CONSTANTS: GROUP ELASTIC SCATTERING ANISOTROPY PARAMETERS [14]

Nuclides
D-2, T-3, He-3, He-4, Li-6, Li-7, Be, B, C, N, O, F, Na, Mg,
Al, Si, P, S, K, Ca, Ti, V, Cr, Mn, Fe, Co, Ni, Cu, Zn, Y,
Zr, Nb, Mo, Ta, W, Pb, Bi, Th, U-235, U-238, Pu

Note: For the nuclides listed there are group elastic scattering anisotropy parameters $B^{i\to j}_{e,\ell}/B^{i}_{e,0}$ for $0 \le \ell \le 5$ [14]. Values of $B^{i\to j}_{e,\ell}$ have been determined by Bazazyants et al. [14]; values of $B^{i}_{e,\ell}$ have been calculated by means of the following relation:

$$B^{i}_{e,\ell} = 2\pi\,(2\ell+1) \int_{E_i}^{E_{i-1}} dE' \int_{-1}^{+1} \sigma^{i}_{e}(E', \mu_L)\, P_\ell(\mu_L)\, d\mu_L$$

The initial information used for obtaining the values of $B^{i\to j}_{e,\ell}$ and $B^{i}_{e,0}$ is contained in Ref. [15].

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2.3. BASIC RULES OF NEUTRON NUCLEAR DATA EVALUATION

In endorsing the report of the subgroup on "Basic Rules of Neutron Nuclear Data Evaluation", the Panel remarked that because of the limitations of time and manpower many of the existing evaluations fall some way short of the ideal, both in the analysis and selection of data and as regards documentation. For the most important materials, the most important reactions and the most important parts of the energy range the evaluations are being steadily improved, depending on the differing needs of the various requesting agencies. This degree of progress has been much expedited through collaboration and exchange between existing evaluation groups. For the less important materials, reactions and energy ranges there is still much room for improvements.

Subgroup Report

(Chairman: J.S. Story)

Story's dicta:

(1) If there are only two measurements and they agree, both are wrong
(2) Any new measurement will be more discrepant than those already available

Evaluation is a complex and difficult task, and none of us can expect to have complete knowledge of theory and experimental technique for all parts of the energy range which is covered in typical evaluated data files (this is usually about 0 to 20 MeV). What one can reasonable ask of an evaluator is that he should have enough knowledge and common sense to go and discuss the data with the appropriate experts whenever it seems worth while and possible to do so. The typical sequence of operations in evaluating neutron nuclear data is as follows:

(a) Prepare a reference list, using CINDA and other reference sources
(b) Collect and compare experimental data utilizing the international data centres and other sources
(c) Fill gaps by theory or art
(d) Select and tabulate a complete consistent set of microscopic data over the full energy range (EVALUATED DATA)
(e) Transfer these data to magnetic tape, with a simple check operation
(f) Check file completely
(g) Check further, using automatic plot routines, wherever possible.

Ideally, an evaluator should: consider all relevant experiments; renormalize the results as necessary, using up-to-date and consistent reference cross-sections; review the uncertainties, taking account of both statistical and systematic sources of error; resolve as many discrepancies as possible; compare with all relevant parts of nuclear theory and with nuclear systematics where appropriate; and, finally, evolve a self-consistent set of data which is incorporated into an evaluated data file in a well-documented standard format.

However, in the necessary compromise between the ideal and economic reality, the first duty of an evaluator is to satisfy his own users' most urgent needs as expeditiously as possible. Therefore, in reality an evaluation effort must often fall short of the ideal. The time and manpower available are usually insufficient for such detail at the first and even the second round. Another reason for making only a restricted evaluation effort arises when the evaluator knows that important new experimental data will shortly be available which could justify further changes.

Economy of effort can often be effected if the evaluator is able to find an existing evaluation which is sufficiently accurate for his needs over part of the energy range. The limited manpower available can then be used to do a better job on those cross-sections and those parts of the energy range which are felt to be most important and to supplement, where necessary, missing parts of the file.

An evaluator often spends a great deal of time and effort trying to resolve discrepancies, but often enough the discrepancies are still there at the end of all his efforts. There are then several courses open to him:

(a) He can choose the most recent set of data, or base his choice on some other extreme argument;
(b) He can choose some middle course, using weighting arguments of greater or less sophistication;
(c) He can produce two data files giving high and low values.

Whichever course is adopted the evaluator should give his reason for the choice, even if it was only that the choice was made arbitrarily. Otherwise it will acquire canonical weight — someone will think there was some magic about it, or that the evaluator had some private information not yet published.

An evaluator should reflect that, if the resolution of a discrepancy is a matter of great importance, it is possible that the real need is for new and improved measurements.

As far as the use of nuclear theory is concerned it is usually better to follow the experimental data rather than the theory when there is a conflict between them. There are obvious exceptions to this recommendation, for example, Wick's inequality for the forward scattering cross-section which is based on a principle of great generality.

In general, attention should be paid to the possible structure of the reference cross-section when normalizing results which were obtained with much higher resolution than that of the reference set. The Panel felt that at the present time the U-235 fission cross-section below 100 keV, the B-10(n,α) or B-10(n,$\alpha\gamma$) cross-sections above 100 keV and the gold capture cross-section in the keV energy range are not satisfactory as reference standards.

The use of consistent sets of data, such as the periodically updated IAEA set of thermal values for the main fissile nuclides, is generally preferred.

Adequate documentation, such as a formal report, should be produced for each evaluated data file[10] placed in the library. However, it must be recognized that the availability of the numerical data file may precede the formal documentation by several months. Therefore, a one- to two-page summary describing each evaluation should accompany the data file and this should be updated to reflect all changes. It is important in the documentation to state precisely which reference standards and normalization procedures were adopted and whether the measured values were changed by the evaluator. The evaluation could then be more readily updated when the values of the reference standard have been improved.

There has been increasing demand from reactor physicists for reliable assessments of the uncertainties of evaluated data. This might prove a tremendous task unless the evaluator confines himself to simple ad hoc estimates over broad energy regions and attempts to obtain only approximately the correlation of uncertainties between these regions. Despite the difficulties, any ad hoc uncertainty estimates would be most desirable and it is recommended that these should be attempted whenever time allows. The participation of experimentalists in these assessments of errors is also desirable, and extensive new measurements should not be undertaken before the assessments have been made.

2.4. ESTABLISHMENT OF COMPUTER LIBRARIES OF EVALUATED DATA AND ASSOCIATED COMPUTER PROGRAMS

The paper presented to the Panel on the "Establishment of Computer Libraries of Evaluated Data and Associated Computer Programs" by S. Pearlstein (see Appendix C) was generally endorsed by the subgroup. The subgroup also summarized information on the following topics:

A. Formats for evaluated nuclear data libraries
B. Format problems and recommendations
C. Checking procedures for evaluated data files
D. Extensibility of existing formats
E. Subgroup representation in the USSR Evaluated Nuclear Data Library

Subgroup Report

(Chairman: S. Pearlstein)

A. Formats for evaluated nuclear data libraries

File name	ENDF (Evaluated Nuclear Data File)
Documentation	Data Formats and Procedures for the ENDF Neutron Cross Sections Library, ENDF 102 (BNL-50274) Vol. 1 (M. Drake, Ed.). ENDF Formats and Procedures for Photon Production and Interaction Data, ENDF 102 (LA-4549) Vol. 2 (D. Dudziak, Comp.).

[10] There are differences in the meaning attached to the term "data file" in different countries. In this section the term has been used to imply the whole set of data, uniquely identified by a particular Data File Number (DFN), e.g. in the UKNDL, or Material (MAT) Number, e.g. in the ENDF/B, for a single material.

Description Card image formats are specified for neutron and photon data. Information can be specified for

- Radioactive decay
- Smooth cross-section
- Angular distribution
- Energy distribution
- Resonance parameters
- Fission product yield
- Delayed neutron fractions
- Thermal neutron scattering law
- Atomic form factors
- Gamma-ray multiplicities
- Nuclear parameters

Data can often be specified in either parametric or tabular form. Although some data may be specified redundantly for checking purposes or convenience, such as the specification of elastic scattering, which can also be derived from the total and non-elastic cross-sections, no data for a given reaction type may be specified in two equivalent ways.

File name KEDAK (Karlsruhe Evaluated Data File)

Documentation Card image format of the Karlsruhe Evaluated Nuclear Data File, D. Woll, KFK-880.
Status report, B. Hinkelmann et al., KFK-1340.

Description Card image formats are specified for neutron data. Information can be specified for:

- Characteristics of the nuclei (mass number, isotope abundance, etc.)
- Resonance information in resolved and unresolved regions
- Neutron cross-section data in pointwise representation
- Angular distributions
- Energy distributions of prompt and delayed fission neutrons, inelastically scattered neutrons, (n, 2n) neutrons.

Data can be specified in tabular form with a unique, presently linear-linear interpolation scheme. Angular distribution can be given in centre-of-mass and laboratory systems, either by tabulation or Legendre coefficients. Some quantities can be given in parametrized form.

Redundant information is given, some for the convenience of users, some for checking purposes, for:

- Partial and total cross-sections
- Partial and total width of resonance in both resolved and unresolved resonance regions
- Strength functions and averaged resonance parameters
- Pointwise cross-sections and resonance parameters (in resonance region)
- Cross-section ratios and cross-sections themselves
- Averaged cosine of elastic scattering cross-section and angular distribution
- Transport cross-section.

File name	LLL (Lawrence Livermore Library)
Documentation	Description of Evaluated Nuclear Data System used at Lawrence Livermore Laboratory, R. J. Howerton, UCRL-50400, Vol. 4.
Description	Formats are specified for neutron, photon and charged particles data. Information can be specified similar to ENDF/B, UKNDL, KEDAK and the USSR library. Part of the Library content is in close relation to that of ENDF/B, but the format is different.

File name	UKNDL (United Kingdom Nuclear Data Library)
Documentation	PARKER, K., AWRE-O-70/63 (1963) (Format).[11] NORTON, D.S., AEEW M-824 (1968). STORY, J.S., unpublished report (WNDG/91); (see also Appendix C).
Description	Card image formats are specified for neutron and photon data. Information can be specified for: Cross-sections in pointwise representation; Secondary distributions; Secondary energy distributions; Resonance information in resolved and unresolved regions (not utilized)[12]; Thermal scattering law data (not utilized).

Cross-section data are interpolated linearly in a log-log scale. Formats for polynomial representation of angular distributions are available but are not utilized in the United Kingdom.

File name	USSR Evaluated Nuclear Data Library
Documentation	KOLESOV, V.E., NIKOLAEV, M.N., "Format of the recommended nuclear data library for reactor calculations", in Russian (see Appendix C). English Transl. INDC (CCP)-13/L, International Nuclear Data Committee, IAEA, Vienna (1970).
Description	Card image formats are specified for neutron data. Information can be specified for: Neutron cross-section data in pointwise and subgroup representation; Resolved and unresolved resonance information; Angular distribution; Energy distribution; Thermal neutron scattering law.

[11] It should be noted that a few extensions to the format have been developed; unfortunately, the documentation of these developments is still only in draft.

[12] A separate library of resonance parameter data is being created, following the formats of JAMES, M.F., AEEW-R-621 (1968).

Data can be specified in tabular form with different interpolation schemes. Angular distribution can be given in centre-of-mass and laboratory systems, either by tabulation or Legendre coefficients. Many quantities can be specified in parametrized form.

B. Format problems and recommendations

B.1. Difficulties encountered with existing formats and format translations

Several of the Panel members were familiar with existing formats for evaluated data and the translation from one format to another. Some of the operational difficulties in these tasks are enumerated here. All formats presented difficulties. It is not intended to include all difficulties experienced in the handling of formats, but to indicate the types of problems that arise. This should give guidance to what the user may expect and provide information for the future development of evaluated data formats.

(1) If the library is not ordered sequentially by Z, A or material identification, searching procedures are more cumbersome.

(2) Angular distributions specified in both tabular form and as Legendre coefficients within a material (as allowed in ENDF materials but not within a reaction type) may require processing codes not only for handling both forms but also for interpolating between two angular distributions of different functional form.

(3) For some applications Doppler broadening effects are important and also resonance parameters are useful in both the resolved and unresolved energy range, although alternate representations may be appropriate for the latter. In the UKNDL these parameters are not included except as a separate auxiliary library.

(4) Where data for a reaction type are divided into ranges, processing is made more difficult unless the complete length of the data section can be learned at the beginning of the section. This problem occurs in the UKNDL[13].

(5) Processing of the data for a particular reaction type is impeded if the different subsections (for inelastic scattering to different levels, for example) do not appear in consecutive order[14].

(6) Certain energy distribution laws appearing in the ENDF, KEDAK, and UKNDL do not have a counterpart in the other formats and cannot be translated exactly.

(7) Fields containing floating point numbers sometimes reflect the loss of significant figures, i.e. a 6-digit number written in an E-11.4 format will only contain 5 significant figures.

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[13] In the UKNDL, section 0 at the beginning of the file does indeed give the number of card images in each section. This is found quite adequate for practical purposes in the United Kingdom, though it is true that the last card of each energy range may have from 1 to 5 of the 6 data fields blank.

[14] In the United Kingdom, this problem, though acknowledged, is not considered to be a very serious difficulty. For the UKNDL the Reaction Type Numbers (RTN) 5-14 and 31-50 have been allocated for inelastic scattering to discrete levels. On those rare occasions when the second group of numbers is used it may be possible to arrange the data so that the two groups of RTNs appear consecutively in the data files (and are then followed by RTNs 15-30 and from 51 onwards). However, this possibility has not been tested through the editing, checking and processing programs associated with the UKNDL and it may take some time before this can be done.

(8) The documentation for the data within a file is often not available and this can hamper both the processing of the format and the understanding of the data[15].

(9) General-purpose handling codes for the libraries are often not released or are software and machine dependent[16].

B. 2. Recommendations on formats or procedures

(1) An index of the contents of each library tape should appear at the beginning of each tape and should contain the name of the materials and the number of card images in each material, similar to the procedure followed in the UKNDL.

(2) Materials should be transmitted with the Z, A or material identification number in ascending order.

(3) Within a material, all numbers denoting reaction types should appear in ascending order.

(4) It should be possible to identify a material as an isotope, element or mixture.

(5) Standardization of units among formats is recommended.

(6) It was agreed that specification of angular distributions in only one representation within a material offered programming advantages, but the Panel members had differing views on whether this procedure should be recommended to the evaluator.

(7) The beginning of any section of a data file describing a reaction type should contain book-keeping information which establishes the number of points or the length of the file.

(8) All components of a reaction type such as the total inelastic scattering cross-section and partial cross-section for the excitation of levels should be given sequentially within the file.

(9) If energy-dependent average resonance parameters are specified, the energy range over which they are applicable should also be specified.

(10) If the tabulated values are derived from resonance parameters, the resonance parameters should also be specified because of their importance.

(11) The physics information entered into a data file should not be restricted by format. The number of points used to describe a cross-section should not exceed the number of points necessary to convey the physics information. Because of the problems of small computers mentioned elsewhere, there were different opinions on how or whether the format should allow section subdivisions or limited array sizes.

(12) Care should be taken to avoid the loss of significant figures in the transmission of data, such as the transmission of a 6-significant digit number in an E-11.4 format.

(13) Each distinct data set in the library should have a unique identification number to facilitate the retrieval of data, and this identification number should appear on each record.

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[15] In the United Kingdom, methods are being explored for introducing documentary information into the UKNDL.

[16] A systematic program of work for converting the UKNDL editing, checking and user codes to FORTRAN-4 is in progress; however, a large amount of effort cannot be diverted to this task.

TABLE XV. COMPARISON AMONG THE SECONDARY ENERGY DISTRIBUTION LAWS IN ENDF/B, KEDAK, LLL, UK AND USSR LIBRARIES

Definition	Law number assignment					
For a given ingoing energy E_0 the probability distribution of the outgoing energy E' is:	ENDF/B-II	KEDAK	LLL Trans-mission format	LLL Basic format	UK	USSR
1. Known discrete energy		*			1	1
2. $E' = k(E_0 - E_d)$					2	2
3. Continuous spectra independent of E_0			3		3	
4. Probability function, $p(E') = f(E_0, E')$					4	5
5. As above, but $p(E'/E_0^{\frac{1}{2}}) = f(E_0, E'/E_0^{\frac{1}{2}})$					5	6
6. As above, but $p(E'/E_0) = f(E_0, E'/E_0)$					6	7
7. General fission spectrum					7	4
8. Tabulated function dependent on E_0	1		8		8	8
9. Energy-angle correlated $f(E_0, E', \mu)$			9	3	9	
10. Simple evaporation spectrum, $\theta = \theta(E_0)$	9		10	5	10	
11. Discrete level excitation	3					
12. General evaporation spectrum, $\theta = \theta(E_0)$	5					
13. Maxwellian function	7					
14. Watt spectrum	10	*				3
15. Energy-angle correlated Legendre coefficients				4		
16. Cumulative probability distribution				6		
17. Histogram form of energy distribution				8		

Notes:

The only laws presently given by KEDAK are the ones marked by *, although provisions are made in order to insert the distribution laws for the continuum region in the file itself. No number has been assigned to the laws in the table since they do not belong to a section uniquely devoted to the secondary energy distributions.

ENDF-10 and USSR-3 are exactly corresponding, although the data given in the respective files differ slightly; the same remark applies to ENDF-9 with respect to LLL-10, LLL-5 and UK-10.

(14) Checking, updating, plotting and other handling codes should be made available to all users of the formats. They should be as software and hardware independent as feasible.

(15) The feasibility of using a common format for the transmission of evaluated data should be investigated.

(16) In the revision and development of formats for evaluated data, small computers should be considered whenever practical.

(17) When translating from one format to another a faithful translation should be attempted, but it should be realized that the pursuit of high fidelity may sometimes result in an overly lengthy file, and some relaxation of fidelity may be necessary on practical grounds. In testing the degree of fidelity attained it is useful to compare integral quantities such as Maxwellian averages, resonance integrals and fission spectrum averages between the original and the translated version.

(18) Increased simplification and greater accuracy could be achieved if common laws for secondary energies could be used in the evaluated data libraries. To initiate discussion of this point the differing energy laws now in use in the various nuclear data libraries are summarized in Table XV. Some notes on the secondary neutron energy distribution laws in the UKNDL are given below.

(19) While it is recognized that certain redundancies are desirable for checking purposes and some applications, the Panel recommended avoiding them as far as possible. An example is the pointwise presentation of angular distributions which may be in the form of either normalized probabilities or differential cross-sections. The latter can be derived from the former and the appropriate interaction cross-section.

Secondary neutron energy distribution laws in the UKNDL

One of the most distinct differences between the secondary energy distributions allowed in the UKNDL and those in the ENDF/B lies in the variety of different interpolation schemes available in the ENDF/B. In the UKNDL every "continuum" secondary energy law is invariant throughout a specified range of incident neutron energies. To some extent this limitation has been overcome in the UKNDL scheme through the availability of the three different models (Laws 4, 5 and 6) for evaporation-type spectra. In practice the restriction has not proved embarrassing, probably because of the relatively poor accuracy of the available experimental data.

Another distinction is that secondary energy distributions in the UKNDL are usually represented in tabular form, whereas the ENDF/B makes more use of parametric forms.

A much more important distinction is that with the UKNDL the processing programs make use of the reaction Q-values to determine the incident-energy-dependent upper energy limit of several of the continuum distributions. This is a serious limitation in practice because the Q-values appear only once, at the head of the associated cross-section data. In the ENDF/B, for each continuum distribution the format includes an upper cut-off parameter U.

From the compiler's point of view it would be desirable to have formats available which include both upper and lower cut-off parameters U and L, such that

$$E-L \le kE' \le E-U, \quad \text{with } k \approx (1+A)^2/(1+A^2)$$

where E is the incident neutron energy, E' the secondary neutron energy (in the laboratory frame), and A is the atomic mass in units of the neutron mass. With this development it would become possible to represent in reasonable approximation the (n, n') secondary energy distribution at energies above the (n, 2n) threshold, for example.

There is of course no serious difficulty in designing additional secondary energy distribution laws for UKNDL and ENDF/B, with different interpolation schemes if desired. They would be of no use, however, until the user programs had been modified to recognize them and interpret them.

There follow a few remarks on the individual secondary energy laws in the UKNDL:

Law 2 (UKNDL) appears to embrace and to be more general than the ENDF/B Law 3.

Law 3 (UKNDL) appears somewhat redundant since it could be replaced by Law 4. However, Law 3 is primarily intended for fission neutron spectra (of arbitrary form) with invariant upper cut-off energies and could be used with a faster processing subroutine.

Laws 4, 5, 6 (UKNDL). In general, these laws are intended for use with evaporation-type spectra and will be processed by a subroutine which will calculate the upper cut-off E'_{max} from an expression of the form

$$E'_{max} = U = [\,(1+A^2)\,E + A(1+A)Q] \,/\, (1+A)^2$$

where E is the incident neutron energy, and Q is the reaction Q-value, so Q must be carefully chosen. An evaporation spectrum with constant temperature is conveniently represented by Law 4 while, if the temperature is proportional to $\sqrt{E}$, Law 5 is used instead.

Law 7. This is one of the few parametric forms used for secondary energy spectra, but note that the relative probabilities for the two components of the spectrum are computed from the fission cross-section data and from $\bar{\nu}$.

Laws 8, 9, 10. These additional laws have been defined in a revised draft of the UKNDL, issued in June 1965 with a very limited distribution. The definitions are reproduced below. Law 8 has been used in some compilations by Drake and others at General Atomics; Law 10 has been used by Hart in DFN-328 for tantalum, and more recently by Benzi et al. in a set of 18 files of fission-product data. Law 9 has been used for (n, 2n) secondaries in two files for D and T compiled by Horsley and Stewart in October 1967. However, these three distribution laws have not yet been implemented in the United Kingdom's main user program GALAXY, and for this reason at Winfrith they were replaced by alternative representations from the original set, Laws 1 to 7.

Law 8 allows the presentation of differential cross-sections for secondary neutron production $\sigma(E,E')$ in barns/MeV. Values of $\sigma(E,E')$ are tabulated over the allowable range of E' for each of a set of values of the incident energy E. If the reaction produces more than one secondary neutron, all are included within the single tabulation, so that for the (n, 2n) reaction, for example,

$$\int \sigma(E, E')\, dE' = 2\,\sigma_{n,2n}\,(E)$$

Law 9 allows the representation of a secondary neutron energy distribution correlated with the angle between the directions of the incident and secondary neutrons.

Law 10 allows the representation of the normal evaporation spectrum in parametric form:

$$f(E, E') = (E'/T^2) \exp(-E'/T), \qquad T = \sqrt{(E/a)}$$

This has some advantage for use in Monte Carlo processors.

C. Checking procedures for evaluated data files

Automated checking procedures have been developed for use with the evaluated data libraries. A brief description of the codes used with each format is given below.

ENDF[17,18]

Two classes of checking procedures have been developed: (i) Format consistency checks, (ii) Physics consistency checks.

(i) Format consistency checks

Three programs are available: CHECKER, SUMUP, CAREN. Some of the functions performed by these codes are listed below.

CHECKER: This code, for example, checks for magnitude errors in cross-section tabulations, for negative cross-sections in angular distributions calculated from Legendre coefficients, and performs normalization checks of probability distributions.

SUMUP: Checks sums of partial quantities against the sum-quantity, e.g. the sum of the partial cross-sections against the total cross-section and the sum of partial resonance widths against the total resonance width.

CAREN: Checks the continuity of cross-sections at the interface of different regions of cross-section representation, e.g. resolved and unresolved resonance region.

(ii) Physics consistency checks

Two codes are available now, PSYCHE and PHØX. PSYCHE is intended for physics checking of neutron data, and PHØX for testing the physical validity of the photon data.

PSYCHE: A few examples will describe the basic idea of this code: It tests nuclear masses, decay constants, ground-state spins, etc., against an internally stored nuclear mass table, and tests spin and orbital angular momentum resonance information against the sum rules; it also calculates resonance integrals from resolved parameters and strength functions calculated

[17] Description of checking codes for neutron data will be published by the National Neutron Cross-Section Center, Brookhaven National Lab., Upton, N.Y., and distributed by the Argonne Code Centre.

[18] DUDZIAK, D.J., Translation of ENDF/B and Physics Checking of Cross-sections for Shielding, DASA-2379, ENDF-130.

from unresolved resonance parameters and compares resolved resonance parameters against statistical parameters. It also checks the violation of Wick's limit in elastic angular distributions.

PHØX: Performs also formal checking of the contents of the photon data part of the library. Some of the tests check the Q-value against the lowest energy point, the energy released against the available energy, and the normalization of photon fields.

KEDAK

Presently, programs for checking the format and physics consistency of nuclear data files are not in use but are in the status of problem definition. This work will make extensive use of the definitions of checking procedures published by the National Neutron Cross-Section Center, Brookhaven National Lab., Upton, N. Y. However, it has always been the KEDAK policy to set up procedures which will minimize the format and consistency errors that occur before entry into the library.

Two updating programs, KEMA and KEDABE are currently in use. Both contain input checking routines, and KEDABE does permit changing mutually dependent cross-sections consistently. Since the updating procedure is fully automated, format errors and errors in the sum of the quantities cannot occur.

The generation of input data for updating the library is computerized whenever fitting or interpolation programs are used.

Data entered into the library are checked twice, once before starting the updating programs and again after they have been stored in the files, both by plotting and printing.

Japan

Presently, checking of the nuclear data library is performed by plotting and printing the contents. It is hoped, however, to develop a checking program within the next twelve months. This program will be based on the extensive work of the National Neutron Cross-Section Center, Brookhaven National Lab., Upton, N. Y.

UKNDL

The program CHECK (AEEW-M-347) performs more than 70 format and consistency checks of various kinds. It checks, for example, that the number of card images agrees with the number quoted at the head of the section; that the neutron energies run in monotonic sequence; that the partial cross-sections agree with total cross-sections; that reaction thresholds are not inconsistent with the listed Q-values. The program GROD (AEEW-M-426 and AEEW-M-683) prepares a tape for Graphical Representation of Data by means of a cathode-ray-tube graph plotter.

The program MINIGAL calculates from the data files: cross-sections at 0.025298 eV; cross-sections averaged over a Maxwellian flux spectrum at 20.44°C; resonance integrals from 0.55 eV to 2 MeV; and fission spectrum average cross-sections using the U-235 Watt-Cranberg spectrum.

USSR[19]

Testing of the contents of the neutron nuclear data library is performed using the program POSOSHOK. This program, though already being intensively used, was especially designed for future inclusion of further checking routines.

Presently, tests of the following nature are performed: Formal ordering of files and the file contents; agreement of the file heading with the file contents; normalization of the probability distributions; magnitude errors in the tabulated cross-sections; and tests against negative energies and negative cross-sections.

The program will be extended to include also, for example, tests of sum-quantities and tests on resonance parameters.

D. Extensibility of existing formats

Existing formats were examined to determine whether they included or could be expanded to include:

(i) Data for incident particles other than neutrons or photons
(ii) Data for the angular and energy distributions of outgoing particles other than neutrons or photons

The conclusions for item (i) are summarized in Table XVI and the conclusions for item (ii) in Table XVII.

TABLE XVI. PROVISION FOR INCIDENT PARTICLES OTHER THAN NEUTRONS OR PHOTONS

Format	Provision exists	Extensible	Planned
ENDF	No	Yes	Yes
KEDAK	No	Yes	No
LLL	Yes	-	-
UKNDL	No	Yes	No
USSR	No	Yes	No

[19] KOLESOV, V.E., KRIVTSOV, A.S., SOLOV'EV, N.A., "Automation of the procedure for checking information contained in the library of evaluated nuclear data: the "Pososhok" programme", in Russian (see Appendix C): English Transl. INDC(CCP)-23/G, International Nuclear Data Committee, IAEA, Vienna (1972).

TABLE XVII. PROVISION FOR ANGULAR OR ENERGY DISTRIBUTION OF OUTGOING PARTICLES OTHER THAN NEUTRONS OR PHOTONS

Format	Provision exists	Extensible	Planned
ENDF	No	Yes	Yes
KEDAK	No	With difficulty	No
LLL	Yes	-	-
UKNDL	No	Yes [a]	No
USSR	No	Yes	No

[a] Designing the extension of the format might be tedious, but it would not be difficult. The difficulty lies in constructing user programs to utilize the data in the new format, or in modifying existing processing programs.

E. Subgroup representation[20] in the USSR Evaluated Nuclear Data Library

The subgroup representation of the neutron cross-section structure is a convenient means of data presentation in the field of non-resolved resonances. Such a presentation can also be applied in the field of resolved resonances if it is not necessary to have detailed information on the cross-sections in that region. This also makes it possible significantly to diminish the volume of information to be stored.

Subgroup representation is widely used in the Soviet Union; a similar method is used in France. The format of the evaluated data library in the Soviet Union is designed with a view to storing the neutron cross-sections in the form of the subgroup representation.

Dr. M. N. Nikolaev, of the Institute of Physics and Power Engineering, Obninsk, who is one of the originators of the subgroup representation, has submitted a report on the subgroup method, which is given below.

SUBGROUP METHOD

(Fundamental postulates and comments)

M.N. Nikolaev
Institute of Physics and Power Engineering
Obninsk, USSR

The subgroup method was developed to describe the resonance self-shielding of cross-sections in the region of unresolved resonances near interfaces, where the neutron flux structure — and hence the cross-sections averaged over it — is strongly dependent on the co-ordinates.

[20] NIKOLAEV, M.N., UK-USSR Seminar on Computations of Nuclear Data for Reactors, Dubna, June 1968. See also: KOLESOV, V.E., NIKOLAEV, M.N. (Appendix C).

The method is based on the following assumptions:

(i) The scattering integral is a smooth function of energy (the narrow-resonance approximation);

(ii) The neutron flux averaged around each energy E over an interval ΔE containing many resonances varies little with energy over intervals of the order of the averaging interval.

Let us average the exact kinetic equation around each energy E over an interval ΔE containing many resonances (smoothing interval). Further, let us determine for each smoothing interval the distribution function of the total cross-section $p(\Sigma)$ and the combined density of the probability $Q(\Sigma_s/\Sigma)$ that for a total cross-section Σ a neutron in the interval ΔE will have a scattering cross-section Σ_s. By introducing these functions, one can replace averaging over energy in the interval ΔE by averaging over the distribution functions of the total cross-section. Equating the integrands in the external integrals over Σ and replacing $\Sigma_s\varphi$ by $\int \Sigma_s(\Sigma)\varphi(\Sigma)d\Sigma$ in the scattering interval (which can be done according to assumption (ii), we obtain a kinetic equation for $\bar{\varphi}(r, E, \Omega, \Sigma)$, the neutron flux averaged over those parts of the region ΔE around energy E in which the total cross-section is equal to Σ.

On the basis of assumption (i), one may take it that the probability of a scattered neutron having a total cross-section for interaction with the medium in the range from Σ to $\Sigma+d\Sigma$ does not depend on what total interaction cross-section it had before scattering; this, together with assumption (ii), considerably simplifies the way in which the scattering integral is written. We shall call the equation for $\bar{\varphi}(\vec{r}, E, \vec{\Omega}, \Sigma)$ an equation in the Σ-representation. If assumptions (i) and (ii) hold, the exact value of the neutron flux averaged over the interval ΔE around the energy E can be obtained as follows:

$$\bar{\varphi}(\vec{r}, E, \vec{\Omega}) = \int \bar{\varphi}(\vec{r}, E, \vec{\Omega}, \Sigma)\, d\Sigma$$

Further, the exact form of $p(\Sigma)$ is approximated by a sum of weighted δ-functions:

$$p(\Sigma) = \sum_{k=1}^{M} a^k \delta\,(\Sigma - \Sigma^k); \qquad \sum_{k=1}^{M} a^k = 1$$

where

$$\bar{\Sigma}_s(\Sigma) = \int \Sigma_s Q\,(\Sigma_s/\Sigma)\, d\Sigma_s$$

in the equation for $\bar{\varphi}(\vec{r}, E, \vec{\Omega}, \Sigma)$ is replaced by $\bar{\Sigma}_s^k$, the mean cross-section for the scattering of neutrons having a total cross-section for interaction with the medium of Σ^k. Accordingly, we arrive at a system of interconnected equations for fluxes $\bar{\varphi}_k(\vec{r}, E, \vec{\Omega})$ of neutrons having a total cross-section for interaction with the medium equal to Σ^k. In this $\delta\Sigma$-approximation,

$$\bar{\varphi}(\vec{r}, E, \vec{\Omega}) = \sum_{k=1}^{M} \bar{\varphi}_k(\vec{r}, E, \vec{\Omega})$$

Since $\bar{\varphi}_k$, Σ^k and $\bar{\Sigma}_s^k$ are defined as averages over the interval ΔE around the energy E, by virtue of assumptions (i) and (ii) they are smooth functions of energy. It is therefore natural to solve the equations in the $\delta\Sigma$-representation in the multigroup approximation. Averaging them over the group intervals (within the limits of which $\bar{\varphi}$ can vary considerably) gives us a system of "subgroup" equations where the k-th subgroup of a group g means all those neutrons of group g which have a total cross-section for interaction with the medium in some interval $\Delta\Sigma^{gk}$ around Σ^{gk}. The energy position of the subgroup within group g is not identified: the neutrons of each subgroup are assumed to have the same energy distribution within the group interval. The group flux is the sum of the subgroup fluxes:

$$\varphi_g = \sum_{k=1}^{M_g} \varphi_{gk}$$

The equations for φ_{gk} contain the following macroscopic constants: Σ^{gk}, the total cross-section in the k-th subgroup of group g; $\bar{\Sigma}_s^{gk}$ and $\nu\bar{\Sigma}_f^{gk}$, the scattering cross-section, and $\nu\Sigma_f$, both averaged over the k-th subgroup of group g; $W^{gk\to g'k'}(\mu_0)$, the probability of a transition from subgroup k of group g to subgroup k' of group g' in scattering through an angle arc cos μ_0. In view of assumption (i) it may be taken that

$$W^{gk\to g'k'}(\mu_0) = W^{gk\to g'}(\mu_0) \cdot a^{g'k'}$$

Similarly, for the fission spectrum,

$$\chi^{gk} = \chi^g \cdot a^{gk}$$

If the angular distribution of the scattered neutrons is a smooth function of energy, then $W^{gk\to g'}$ will not depend on k. For elastic scattering, the dependence of $W^{g\to g'}$ on the shape of the intra-group spectrum (which is the same for all the subgroups) is estimated as with the normal multigroup approach. It should be noted that the matrix of the transitions between subgroups is not triangular: $W^{gk\to gk'} \neq 0$ for $k \neq k'$. Naturally, $W^{gk\to g'k'} = 0$ for $g' < g$.

Particularly important are an optimum selection of the number of subgroups in a group and the determination of the subgroup constants. From an analysis of the kinetic equation in the integral form it becomes clear that these parameters must closely approximate transmission functions of the form

$$T^g(t) = \frac{1}{\Delta E_g} \int_{\Delta E_g} e^{-\Sigma(E)\cdot t}\, dE \simeq \sum_{k=1}^{M_g} a^{gk}\, e^{-\Sigma^{gk}\cdot t}$$

$$T_s^g(t) = \frac{\int_{\Delta E_g} \Sigma_s(E)\, e^{-\Sigma(E)\cdot t}\, dE}{\int_{\Delta E_g} \Sigma_s(E)\, dE} \simeq \frac{\sum_{k=1}^{M_g} a^{gk} \cdot \bar{\Sigma}_s^{gk} e^{-\Sigma^{gk}\cdot t}}{\sum_{k=1}^{M_g} a^{gk}\, \bar{\Sigma}_s^{gk}}$$

The same applies to fission and capture.

It has been shown that four subgroups are enough for approximating the transmission functions with the accuracy warranted by the accuracy with which the cross-section structure is known; in the region of unresolved resonances two subgroups are generally enough. The main thing is that it should be possible to determine the subgroup parameters by an approximation to transmissions directly measured with low resolution; it is important only to measure these curves to sufficient thicknesses. In practical calculations, it is enough to calculate accurately the first moments of the transmission functions

$$\frac{1}{n!}\int^{\infty} t^n \cdot T(t)dt = \langle 1/\Sigma^{n+1}\rangle \qquad \text{etc.}$$

This is how the present system of subgroup constants was built up [1].

It is also important to establish the boundary conditions for the subgroup method. At the interface between two adjacent homogeneous regions with dimensions much greater than the free path, the boundary conditions have the matrix form

$$\left.\begin{aligned} \varphi_1^{gk}(\vec{r}_{\text{bound.}}) &= \epsilon_{12}^{k\ell}\cdot\varphi_2^{g\ell}(\vec{r}_{\text{bound.}}) \\ \varphi_2^{g\ell}(\vec{r}_{\text{bound.}}) &= \epsilon_{21}^{\ell k}\cdot\varphi_1^{gk}(\vec{r}_{\text{bound.}}) \end{aligned}\right\} \qquad \begin{aligned} k &= 1,\ 2,\ \ldots,\ M, \\ \ell &= 1,\ 2,\ \ldots,\ L. \end{aligned}$$

If the regions do not have any common isotopes with a cross-section resonance structure, then

$$\epsilon_{12}^{k\ell} = a_2^{\ell}$$

$$\epsilon_{21}^{\ell k} = a_1^{k}$$

Otherwise $\epsilon^{\ell k}$ depends on both indexes. In the case under consideration, the calculations can be performed on the basis of the microscopic subgroup constants of the homogeneous regions, determined by approximating the transmission functions (or their moments), which in turn are computed from the subgroup constants for the isotopes in each region.

When calculating heterogeneous lattices, one must take into account correlations between the cross-section structures of all the regions through which the neutron has a reasonable chance of passing without collisions. In this case, the macroscopic subgroup constants should be formed in such a way that these correlations are easily taken into account. This can be done by making the number of macroscopic subgroups the same for all zones and equal to the product of the number of subgroups for all isotopes in a subset (for isotopes encountered in only one specific combination the overall subgroup parameters can be calculated beforehand as for a particular nucleus). The total number of macroscopic subgroups M is found to be high, but on the other hand there are absolutely no correlations between different subgroups:

$$\varphi_1^{gk}(\vec{r}_{12}) = \varphi_2^{gk}(\vec{r}_{12})$$

$$\varphi_2^{gk}(\vec{r}_{23}) = \varphi_3^{gk}(\vec{r}_{23}) \qquad \text{etc.}$$

Another approach, which is particularly convenient in Monte Carlo calculations, is to consider individually the subgroup structure of the cross-section of each of the isotopes as the neutron histories unfold. As the history of a particular neutron unfolds after it has been scattered or produced in a target, one finds by random selection the subgroup vector which determines the cross-sections for the interaction of this neutron with each of the isotopes present.

The subgroup method may also be useful in calculating fluxes in the region of resolved resonances, if the purpose of the calculations is to obtain a neutron flux averaged over many resonances and if the resonances are narrow. Generally speaking, the subgroup method is not suitable for describing the resonance structure of cross-sections in the region of intermediate resonances. However, if the resonances are purely scattering resonances, the accuracy with which the mean-group flux is calculated is very high even in the unfortunate case where a group contains only one intermediate resonance [2].

A more detailed account of the subgroup method can be found in Refs [2-5].

The subgroup approach is convenient not only for calculating the distribution of neutrons near the interfaces between resonance media, but also for representing information on the average characteristics of the resonance structure of cross-sections. Cross-sections averaged over the resonance structure in the narrow-resonance approximation can easily be computed from subgroup parameters. This was the reason for introducing the subgroup representation of cross-section structure in the format of the USSR Evaluated Data Library [5].

It is hoped that the subgroup approach will make it possible to achieve an optimum breakdown of large groups into smaller ones when the multigroup library of cross-sections is established.

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2.5. ROLE AND EFFICIENCY OF NUCLEAR THEORY IN EVALUATION: RESOLVED AND UNRESOLVED RESONANCES

The subgroup on "Role and Efficiency of Nuclear Theory in Evaluation: Resolved and Unresolved Resonances" based its discussions on the papers submitted to the Panel by P. Ribon and M. Motta (see Appendix C). The findings of the subgroup, contained in its report below, were endorsed by the Panel.

Subgroup Report

(Chairman: P. Ribon)

A. Formalisms utilized

The main formalisms used are:

Breit-Wigner single-level formula (SLBW)
Breit-Wigner multilevel formula (MLBW)
Reich-Moore formalism (RM)
R-matrix one-channel approximation
Adler-Adler formalism (AA)

They are all based on the R-matrix formalism, with the exception of that of "Adler-Adler" which is related to the "Kapur-Peierls" formalism. The "Adler-Adler" formalism requires a shorter computer time than that of the "Reich-Moore", but it does not give the R-matrix parameters.

In the case of the Reich-Moore formalism it is necessary to do numerical calculations for Doppler and resolution broadening, while it is computed by the ψ, ϕ functions in all the other cases (except in the one-channel approximation, where it is usually not needed).

The formalisms used for the analysis of experimental data for fissile nuclei are either the SLBW formalism or some more rigorous formalism (RM or AA), while it is often possible to use some intermediate approximation such as the MLBW, which gives the R-matrix parameters and is easier to handle than the RM formalism. These approximate forms may be adequate also for cross-section calculations as required for reactor applications.

Users should avoid calculating cross-sections with a certain formalism using parameters which are obtained by another one; otherwise, a different average cross-section may be obtained by calculation.

It is recommended to have a list of existing codes used either to analyse experimental data or to compute cross-sections from given parameters. This list should mention the characteristics of these codes (language, etc.) and their availability. Specifications about the Doppler (and resolution) broadening procedure adopted should also be given.

B. Methods for evaluating resonance parameters

The evaluator should avoid mixing parameters obtained with different formalisms, except in some cases for the SLBW and MLBW formalisms. In addition, it should be checked that the average cross-sections calculated from the parameters agree with the experimental values.

Shape analysis is preferable to the area method for a better extraction of the information contained in the experimental data; the use of the area method is not adequate in the case of widely overlapping resonances as in the case of the fissile nuclei. Greater attention should be paid to the evaluation of the resolution function of the experimental apparatus.

Testing of the various methods used for the analysis of resonance parameters

It is well known that there are large discrepancies between the various sets of resonance parameters for one and the same isotope, mainly for fissile nuclei. To try to understand the origin of these discrepancies it has been suggested that the various laboratories concerned with resonance parameter determination should analyse the same simulated data. These simulated cross-sections are being produced by J. E. Lynn, AERE, Harwell (United Kingdom), who will co-ordinate the analyses and distribute the cross-section values and other necessary details through the Centre for Neutron Data Compilation at Saclay, France. It is desirable that the results of this work should be discussed at a future meeting.

C. Application of theories in the analysis of resonance parameters

In addition to the problem of parameter determination for experimental data, with a certain formalism one has to consider the influence of the number of degrees of freedom and of the correlations existing between the partial widths, both of which may be important for the calculation of cross-sections by statistical methods. From preliminary calculations this seems clear, at least as far as the number of degrees of freedom of the fission width distribution is concerned. In this context, attention should be paid to the fact that because of experimental errors the number of degrees of freedom ν_{true} of the χ^2-law, which characterizes the distribution of a total reaction width, is greater than the value ν_{exp} obtained from a crude analysis of experimental data. Furthermore, it should be noticed that the number of channels is greater than the ν_{true} value adopted, and this has consequences mainly in the case of the fission reaction whose width distribution is also affected by a constant component due to the presence of the $(n, \gamma f)$-reaction.

D. Dependence of average values of resonance parameters on E, J, ℓ systematics

It appears that during the last years there have not been considerable changes of our knowledge of the dependence on E, J, ℓ of average parameters. In particular, there is still no evidence for a dependence of the s-wave strength function upon the spin of the compound nucleus. Deeper studies are recommended.

Also in connection with these problems, the importance of work on systematics should be emphasized. It is recommended that the laboratories already involved in this field continue in this activity which requires a long time and wide experience.

E. Unresolved resonance energy range and intermediate structures

It is obvious that evaluators must be conscious of the existence of intermediate structures and "doorway states"; however, they should avoid interpreting any fluctuation which appears in the experimental data as such structures. More rigorous studies of the fluctuations in the unresolved resonance energy range are necessary in order to achieve a better understanding of the phenomena involved.

Importance of interferences and intermediate structures for reactors

Preliminary studies in the literature seem to indicate that interference effects and the existence of intermediate structures may have an influence on the calculation of the Doppler effect. More investigations on this question would therefore be useful.

Representation of fluctuations

Some evaluation groups customarily give energy-dependent average resonance parameters.

(i) If there is no intermediate structure the energy-dependent average parameters cannot be deduced separately from the average cross-sections for each significant (ℓ, J) state; therefore, the uncertainties in the interpretation of these parameters are much greater than might at first appear.

(ii) If there are intermediate structures, studies are necessary to establish methods to take them into account.

F. Representation of resonance data

It seems that the best representation of data in the resonance region is given:

(i) In the resolved resonance energy range, by the individual resonance parameters;

(ii) In the unresolved energy range, by the average parameters and their statistical laws (see E above). The representation by the probability distribution of the cross-sections in tabulated form for various temperatures seems to be very appropriate, but the procedure for its application in reactor calculations must be well specified.

2.6. ROLE AND EFFICIENCY OF NUCLEAR THEORY IN EVALUATION: STATISTICAL, OPTICAL AND DIRECT INTERACTION MODELS

In accordance with Agenda item 5.II (see Appendix B) the subgroup on "Role and Efficiency of Nuclear Theory in Evaluation: Statistical, Optical and Direct Interaction Models" discussed the following subjects:

A. Availability of the existing computer codes
B. Quality and physical adequacy of computer codes
C. Computer time
D. Convenience of data representation
E. Comparison of computer codes

The subgroup's conclusions, which were endorsed by the Panel, are listed below.

Subgroup Report

(Chairman: V. Benzi)

A. Availability of the existing computer codes

The subgroup felt that a list of existing nuclear model codes could improve greatly the present situation as far as availability is concerned. The subgroup therefore recommended that the IAEA take the necessary steps to produce an updated list of nuclear model codes (spherical optical model, statistical model, etc.) which are available on request. One of these steps might be the circulation of a questionnaire similar to the one shown in Table XVIII.

B. Quality and physical adequacy of computer codes

From the various lists of nuclear codes and from the literature it appears that programs exist for almost all the nuclear models developed so far. In addition, it seems that the majority of these codes are of good quality as far as programming is concerned. However, this does not mean that at present one is able to treat adequately the various aspects of nuclear reactions of

TABLE XVIII. PROPOSED QUESTIONNAIRE ON NUCLEAR MODEL CODES

GENERAL INFORMATION

1. Name of the program
2. Author(s)
3. Laboratory or institution
4. Documentation
5. Nature of physical problem solved
6. Physical model adopted
7. Computer for which the program was designed and others upon which it may be operable
8. Program language
9. Operating system under which the program is executed
10. Computer size and related topics (overlay, chains, links, peripheral devices, auxiliary codes and special subroutines, etc.)

QUESTIONS

1. Is the code available?
2. If yes, in what form? (cards, FORTRAN, binary, magnetic tapes, etc.)
3. On what basis? (privately, through international libraries, etc.)
4. Who should be contacted in order to gain access to the code?
5. Is a manual containing instructions for use and sample problems available?

interest to the evaluation work. In fact, from the physical point of view, a substantial effort is still needed in developing codes. For example, codes dealing with the fission process on the basis of the recently developed theories have not, as of now, been developed. In addition, it seems very important to establish the limits and region of validity of the various nuclear model codes. Such a validity mainly depends on:

(i) the physical adequacy of the adopted theory, and
(ii) the approximations or simplifications, adopted by the author of the code.[21]

The subgroup felt that in addition to the efforts devoted to the analyses of existing programs from the point of view of correctness in formulae, programming, etc., a parallel effort should be devoted to establish the region of validity of the adopted model.

After the adequacy of a model has been established in general, a large degree of arbitrariness still exists in selecting a particular formulation of the model and related parameters whenever a theoretical estimate of an unmeasured cross-section is required. For this reason it seems interesting to compare the results obtained by evaluators from different laboratories when estimating the unknown cross-sections of a given nucleus. Such a comparison should provide some feeling concerning the degree of convergence (or divergence) obtainable in estimating cross-sections on an almost purely theoretical basis and in guessing unknown parameters from systematics. Volunteers should therefore be asked, on a world-wide basis, to produce a data file calculated solely from models, together with a report describing the model and parameters adopted. An informal report comparing the obtained results should then be compiled by one of the relevant groups outside of the IAEA.

C. Computer time

Many of the existing nuclear model codes were developed to investigate fundamental nuclear physics problems. In many cases the physical adequacy of the adopted model can be demonstrated simply by comparing the calculated results with those obtained in a limited number of experiments. For this reason, for several codes no particular attention was devoted to the problem of reducing substantially the running time of the program. The situation is entirely different in the field of nuclear data evaluation where a large number of calculations could be required to fill gaps in the experimental data. The subgroup felt that in developing new codes for evaluation purposes much more effort should be devoted to the problem of saving computer time (e. g. by studying accurately the various numerical methods suitable for solving the mathematical problems contained in the program). As far as the existing codes are concerned, before using the code a careful examination should be carried out in order to establish the optimal mesh interval, the maximum angular momentum involved, etc. This should be done by considering the magnitude of the errors of other origins and the accuracy of the

[21] For example, the statistical model theory is probably adequate to describe the behaviour of the compound nucleus cross-sections for radiative capture in the region of the continuum. The theory requires a correction to take into account Porter-Thomas fluctuations. If such a correction is neglected, the program will be adequate for rough estimates only.

existing experimental data. In addition, in order to save computer time, it seems much better to have single-purpose codes, developed to solve a unique problem, rather than codes with a large number of options (for example, adiabatic and non-adiabatic calculations). In this way computer time losses due to clerical errors in the input cards could also be greatly reduced.

D. Convenience of data representation

As stated above, several nuclear model codes were developed by fundamental nuclear physicists who are not interested in some quantities which are relevant for evaluation purposes. The subgroup felt that the following quantities should be explicitly provided by the codes used for evaluation purposes:

(1) Statistical model

- (a) The numerical values of the fundamental constants adopted (e.g. neutron mass, radius constant, etc.)
- (b) For neutron radiative capture, the various components corresponding to the various values of angular momentum and the total capture cross-section as well
- (c) For inelastic scattering:
  - (i) The various excitation functions
  - (ii) The angular distributions (laboratory and centre-of-mass systems)
  - (iii) The energy spectrum of the emitted particle in the continuum
  - (iv) All the quantities specified below for the optical model, if the statistical calculations are directly linked with optical model calculations
- (d) For evaporation processes:
  - (i) The various excitation functions
  - (ii) The energy spectrum of emitted particles
  - (iii) The nuclear temperature(s), the level density parameters, <u>a</u>, and the spin cut-off factor (if these quantities are calculated by the program)

(2) Optical model

- (a) The numerical values of the adopted fundamental constants
- (b) All the values of the various parts of the optical potential adopted at the considered neutron energy (depths, radii, diffusenesses)
- (c) Transmission coefficients, strength functions, c-matrix coefficients for each partial wave
- (d) Total and differential cross-sections (centre-of-mass and laboratory systems)
- (e) Matching radius and mesh intervals (if kept constant)
- (f) If an automatic parameter search is adopted, the chi-square and potential values for each iteration.

In addition to the above-mentioned quantities, a detailed description of the output routine should be provided in order to allow for the adoption of a different kind of output (e.g. cards, display, plotter), if required by the user.

E. Comparison of computer codes

The subgroup felt that in general a reasonable agreement exists among the various codes when calculations are carried out unter the same conditions. However, it has often been pointed out that even the same code gives different results when used at different laboratories. The subgroup felt that such a situation is caused by an inadequate usage of the code or by errors contained in some particular version of the code itself. Hence, a very careful pre-examination should be made of the listing and of the sample problem before using a code. For this purpose, sample problems covering all the essential options should be provided with the code.

In addition, the subgroup recommended that the IAEA promote a series of standard nuclear model calculations in order to:

(i) Test the versions of the most frequently used codes existing in the various laboratories, under exactly the same input conditions

(ii) Compare the results given by different codes when applied to a particular physical problem under the same conditions (i. e. same form of adopted potential, same parameters, etc.). Such a comparison should be carried out for simple and pathological cases.

2.7. INTERNATIONAL CO-OPERATION IN, AND CO-ORDINATION OF, EVALUATION ACTIVITIES

The Panel's deliberations on this topic (Agenda item 6, Appendix B) were initially focussed on the question of the generation of multigroup cross-section sets. It was noted that in making use of microscopic evaluated nuclear data files for reactor physics calculations the procedure which is generally followed is essentially twofold. First, starting from the basic evaluated data file, multigroup cross-section sets are generated by using certain multigroup generation codes such as GALAXY, MC^2, etc. Secondly, these multigroup cross-section sets are used as input to reactor physics codes, like diffusion and transport theory, and with these one then calculates the parameters of the reactor system or critical assembly of interest. Though the Panel was only competent to deal with the basic evaluated nuclear data files and in particular the confidence level of these microscopic data, which have a direct bearing on the confidence level with which one may calculate reactor systems, it was recognized that the multigroup cross-section sets which are generated from the <u>same</u> evaluated data file using <u>different</u> multigroup generation codes are sometimes different. The existence of such discrepancies between these multigroup cross-section sets tends to introduce a further uncertainty (over and above the uncertainty due to the basic microscopic data) into the reactor calculations. This problem is not of a physical nature, but rather one of mathematical and computational origin, since each of the <u>different</u> multigroup generation codes presumably solves the same physical equations. In view of these factors, the Panel considered the desirability of an international comparison of multigroup cross-section sets to be generated from the <u>same</u> basic evaluated data file using <u>different</u> multigroup generation codes, and subsequently endorsed the following observation:

The Panel observed that in order to fully utilize evaluated neutron data libraries for reactor physics calculations, methods, techniques and codes are generally used to condense the evaluated microscopic data contained in the files into sets of multigroup cross-sections, which then constitute the nuclear data input to reactor physics calculations, and that technical problems exist in this work. Although the subject of multigroup cross-section sets was not discussed, being outside the scope of the meeting, the Panel considered it desirable that the Agency take appropriate steps towards collecting relevant information on discrepancies that are introduced by multigroup processing codes, as well as the documentation of these codes, and that the relevant problems be studied by the appropriate groups, both within and outside of the Agency.

The Panel engaged in a wide-ranging exchange of views on the second major topic of Agenda item 6, the possible improvements in the international exchange of evaluated data libraries. In the course of this discussion, some panelists felt it was important to recognize clearly the background of evaluated neutron data libraries in order to avoid possible misunderstandings. The reason why evaluated neutron data libraries exist is evidenced by their application, this being the improvement of the nuclear energy program and not simply a nuclear physics program. Given this motivation it necessarily followed that commercial and political considerations had to be borne in mind. In addition it was suggested that, since tested evaluations are very expensive to produce, one would not get a free exchange of evaluated data libraries any more than one would get a free exchange of the computer codes necessary to use these data — it depended on how effectively the entire international community would co-operate.

Some panelists noted that there was already an exchange of evaluated data libraries between many countries, though it was not at present a completely free or general exchange. Others, however, remarked that it would be desirable to have a wider exchange of evaluated data libraries since such an exchange could help to accelerate progress in this field. After further discussion the Panel as a whole commended the view that the Agency has made a significant contribution towards furthering the exchange of evaluated data libraries and in this vein endorsed the following observation:

The Panel expressed its support and appreciation of the positive efforts which have been and are being made by the Agency, both within the framework of the International Nuclear Data Committee and the Agency's Nuclear Data Section, towards furthering the collection, dissemination and exchange of evaluated neutron nuclear data. The Panel further considered these efforts to be important for continued progress in this field.

The third and final major topic under Agenda item 6 which the Panel discussed was the co-ordination of evaluation activities on a regional and international scale. At the outset of this discussion the Panel noted that one of the major reasons why the co-ordination of evaluation activities on a more than national scale was often fraught with difficulties was the fact that evaluation activities are closely linked with the time-schedule and priorities of national nuclear energy programs and that these clearly differed from one country to another. The only approach towards a co-ordination of these activities which has, so far, been successful is the publication of the Neutron Nuclear Data Evaluation Newsletter (NNDEN) which is edited by Dr. P. Ribon of Saclay (France) on a four-monthly basis and which contains contributions from, and is distributed to, evaluators in Member States of the Organization for Economic Co-operation and Development (OECD). This Newsletter contains information on new evaluations and codes which either have been recently published, or are in progress, or are planned in the near future (i.e. in the next 4 months).

Dr. Ribon pointed out that the original idea of having such a Newsletter arose from the noticeable duplication of evaluation work which was evident in the OECD Member States and which seemed to be primarily due to evaluators in different countries being insufficiently aware of the corresponding evaluation efforts in progress or planned in other countries. Given this background, it was decided at the 5th Meeting (June 1970) of the Joint Sub-Committee on Evaluation of the European American Nuclear Data Committee (EANDC) and the European American Committee on Reactor Physics (EACRP) to establish the NNDEN starting from 1 July 1970.

It was emphasized that the most important characteristics of the NNDEN were its informal nature, as well as the frequency and speed with which it was produced, since evaluators were particularly responsive to the rapidly changing needs of their requesting agencies. This has the consequence that within a very short time-scale an evaluator may find that his efforts must be re-directed in order to satisfy the needs of his users.

The Panel then considered the desirability, usefulness and practicability of extending the present NNDEN to a fully international scale with regard to the contributions to and the distribution of this Newsletter. After an extensive exchange of views on this matter the Panel arrived at a consensus that the extension of such a Newsletter from both the distribution and contribution standpoint would be desirable but a number of questions must be decided first, such as how and within what time would the contributions be collected. Therefore, in the interim period, it would be preferable for the Agency to try to organize an international newsletter on neutron nuclear data evaluation, separate from but similar to the OECD's Newsletter, which would have a wider distribution than the NNDEN; if and when this international newsletter functions properly, then the NNDEN may no longer have a raison d'être and the parallel publication of these two newsletters may then cease. Reflecting this view the Panel subsequently endorsed the following observation:

The Panel showed general interest in the question of the international publication of a newsletter on neutron data evaluation activities such as the Neutron Nuclear Data Evaluation Newsletter (NNDEN) of the OECD area. Several of the necessary characteristics of such a publication were emphasized, such as its informal nature, the importance of regular and up-to-date contributions and expeditious distribution. It was further suggested that the Agency should explore the technical, practical and financial problems involved as a preliminary to the possible further discussion of this question with the INDC.

3. SUMMARY OF THE RECOMMENDATIONS AND OBSERVATIONS

For ease of reference, the recommendations and observations of the Panel which are contained in the subgroup reports given in Sections 2.3 to 2.7 have been extracted and are listed below. However, each recommendation and observation should be viewed within the context of the specific subgroup report.

Basic rules of neutron nuclear data evaluation
(See Section 2.3)

(1) The Panel remarked that because of the limitations of time and manpower many of the existing evaluations fall some way short of the ideal, both in the analysis and selection of data and as regards documentation. For the most important materials, the most important reactions and the most important parts of the energy range the evaluations are being steadily improved depending on the differing needs of the various requesting agencies. This degree of progress has been much expedited through collaboration and exchange between existing evaluation groups. For the less important materials, reactions and energy ranges there is still much room for improvements.

(2) The Panel felt that at the present time the U-235 fission cross-section below 100 keV, the B-10(n, α) or B-10(n, $\alpha\gamma$) cross-sections above 100 keV and the gold capture cross-section in the keV energy range are not satisfactory as reference standards.

(3) The Panel considered that the use of consistent sets of data, such as the periodically updated IAEA set of thermal values for the main fissile nuclides, is generally preferred.

(4) Regarding the assessment of the uncertainties of evaluated data, the Panel recommended that, despite the difficulties, at least ad hoc uncertainty estimates (over broad energy regions) should be attempted whenever time allows.

(5) The Panel considered that adequate documentation, such as a formal report, should be produced for each evaluated data file placed in a library.

Establishment of computer libraries of evaluated data and associated computer programs
(See Section 2.4)

(1) An index of the contents of each library tape should appear at the beginning of each tape and should contain the name of the materials and the number of card images in each material, similar to the procedure followed in the UKNDL.

(2) Materials should be transmitted with the Z, A or material identification number in ascending order.

(3) Within a material, all numbers denoting reaction types should appear in ascending order.

(4) It should be possible to identify a material as an isotope, element or mixture.

(5) Standardization of units among formats is recommended.

(6) It was agreed that specification of angular distributions in only one representation within a material offered programming advantages, but the Panel members had differing views on whether this procedure should be recommended to the evaluator.

(7) The beginning of any section of a data file describing a reaction type should contain book-keeping information which establishes the number of points or the length of the file.

(8) All components of a reaction type such as the total inelastic scattering cross-section and partial cross-section for the excitation of levels should be given sequentially within the file.

(9) If energy-dependent average resonance parameters are specified, the energy range over which they are applicable should also be specified.

(10) If the tabulated values are derived from resonance parameters, the resonance parameters should also be specified because of their importance.

(11) The physics information entered into a data file should not be restricted by format. The number of points used to describe a cross-section should not exceed the number of points necessary to convey the physics information. Because of the problems of small computers mentioned elsewhere, there were different opinions on how or whether the format should allow section subdivisions or limited array sizes.

(12) Care should be taken to avoid the loss of significant figures in the transmission of data, such as the transmission of a 6-significant digit number in an E-11.4 format.

(13) Each distinct data set in the library should have a unique identification number to facilitate the retrieval of data, and this identification number should appear on each record.

(14) Checking, updating, plotting and other handling codes should be made available to other users of the formats. They should be as software and hardware independent as feasible.

(15) The feasibility of using a common format for the transmission of evaluated data should be investigated.

(16) In the revision and development of formats for evaluated data, small computers should be considered whenever practical.

(17) When translating from one format to another a faithful translation should be attempted, but it should be realized that the pursuit of high fidelity may sometimes result in an overly lengthy file, and some relaxation of fidelity may be necessary on practical grounds. In testing the degree of fidelity attained it is useful to compare integral quantities such as Maxwellian averages, resonance integrals and fission spectrum averages between the original and the translated version.

(18) Increased simplification and greater accuracy could be achieved if common laws for secondary energies could be used in the evaluated data libraries.

(19) While it is recognized that certain redundancies are desirable for checking purposes and some applications, the Panel recommended avoiding them as far as possible. An example is the pointwise presentation of angular distributions which may be either normalized probabilities or differential cross-sections. The latter can be derived from the former and the appropriate interaction cross-section.

Role and efficiency of nuclear theory in evaluation: Resolved and unresolved resonances

(See Section 2.5)

(1) The Panel recommended that the Agency take the necessary steps to produce an updated list of the existing resonance codes used, to analyse experimental data, or to compute cross-sections from given parameters. Such a list should contain the relevant information regarding the characteristics of these codes, their availability and specifications about the Doppler and resolution broadening procedures adopted.

(2) The Panel considered that users should avoid calculating cross-sections with a certain formalism using parameters which are obtained by another one; otherwise, a different average cross-section may be obtained by calculation.

(3) To try to understand the large discrepancies between the various sets of resonance parameters for one and the same isotope, the Panel endorsed the suggestion that the various institutes concerned with resonance parameter determination should analyse the same simulated data.

(4) The Panel recommended that the institutes already involved in work on nuclear systematics should continue this activity which requires a long time and wide experience.

(5) The Panel felt that more rigorous studies of cross-section fluctuations in the unresolved resonance energy range are necessary in order to achieve a better understanding of the phenomena involved.

(6) The Panel supported the view that it would be useful to further investigate the influence of the interference effects and the existence of intermediate structures on the calculation of the Doppler effect.

Role and efficiency of nuclear theory in evaluation: Statistical, optical and direct interaction models

(See Section 2.6)

(1) The Panel recommended that the Agency take the necessary steps to produce an updated list of the existing nuclear model codes (spherical optical model, statistical model, etc.) which are available on request.

(2) The Panel recommended that the Agency should take the appropriate steps to promote a series of standard nuclear model calculations in order to:

(i) Test the versions of the most frequently used codes existing in the various institutes, under exactly the same input conditions

(ii) Compare the results given by different codes when applied to a particular physical problem under the same conditions (i.e. same form of adopted potential, same parameters, etc.). Such a comparison should be carried out for simple and pathological cases.

(3) The Panel supported the view that volunteers should be asked, on a world-wide basis, to produce a data file calculated solely from models, together with a report describing the model and the parameters adopted. An informal report comparing the obtained results should then be compiled by one of the relevant groups outside of the Agency.

International co-operation in, and co-ordination of, evaluation activities
(See Section 2.7)

(1) The Panel expressed its support and appreciation of the positive efforts which have been and are being made by the Agency, both within the framework of the International Nuclear Data Committee and the Agency's Nuclear Data Section, towards furthering the collection, dissemination and exchange of evaluated neutron nuclear data. The Panel further considered these efforts to be important for continued progress in this field.

(2) The Panel observed that in order to fully utilize evaluated neutron data libraries for reactor physics calculations, methods, techniques and codes are generally used to condense the evaluated microscopic data contained in the files into sets of multigroup cross-sections, which then constitute the nuclear data input to reactor physics calculations, and that technical problems exist in this work. Although the subject of multigroup cross-section sets was not discussed, being outside the scope of the meeting, the Panel considered it desirable that the Agency take appropriate steps towards collecting relevant information on discrepancies that are introduced by multigroup processing codes, as well as the documentation of these codes, and that the relevant problems be studied by the appropriate groups, both within and outside of the Agency.

(3) The Panel showed general interest in the question of the international publication of a newsletter on neutron data evaluation activities such as the Neutron Nuclear Data Evaluation Newsletter (NNDEN) of the OECD area. Several of the necessary characteristics of such a publication were emphasized, such as its informal nature, the importance of regular and up-to-date contributions and expeditious distribution. It was further suggested that the Agency should explore the technical, practical and financial problems involved as a preliminary to the possible further discussion of this question with the INDC.

APPENDICES

APPENDIX A

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APPENDIX B

AGENDA OF IAEA PANEL ON NEUTRON NUCLEAR DATA EVALUATION

1. Evaluation activities in Member States, important evaluation needs and the assessment of these needs

2. Status and quality control of evaluations:

   A. Status of existing evaluated data libraries
   B. Quality control – format, consistency and physical checks – influence of macroscopic experiments and adjustments to evaluated data sets

3. Basic rules of neutron nuclear data evaluation:

   A. Comparison of experiments and the criteria used to characterize agreement
   B. Handling of discrepant experimental data
   C. Weighting and fitting procedures
   D. Reference standards used in evaluation
   E. Documentation of evaluations
   F. Assessment of the errors of evaluated data

4. Establishment of computer libraries of evaluated data and associated computer programmes:

   A. Formats, editing and user programmes
   B. Practical problems of representation of evaluated data
   C. Technical problems connected with the exchange of evaluated data libraries – conversion from one format to another

5. Role and efficiency of nuclear theory in evaluation

   I. <u>Resolved and unresolved resonances:</u>

      A. Present status and formalisms used
      B. Experiences, limitations and achievements in the application of theory to resolved resonance evaluation
      C. Energy and spin dependence and systematics of average resonance parameters
      D. Importance of resonance interference and intermediate structure in fission on the Doppler effect
      E. Representation of resolved and unresolved resonance data

   II. <u>Statistical, optical and direct interaction models:</u>

      A. Availability, quality of and estimated computer time for computer codes
      B. Physical adequacy and convenience of data representation
      C. Comparison of computer codes – possible reasons for discrepant results

6. International cooperation in evaluation, coordination of evaluation activities and possible improvements in the international exchange of evaluated data.

APPENDIX C

LIST OF CONTRIBUTED PAPERS

These are gathered together in the unpublished document IAEA-153, available from the Nuclear Data Section of the Agency

Review of the neutron data evaluation activities, the important evaluation needs and the assessment of these needs in IAEA Member States
T.A. Byer, J.J. Schmidt

Neutron data evaluation studies in the United Kingdom in the second half of 1971
J.S. Story

Application of evaluated nuclear data files available at IAEA for reactor calculations
P. Vértes

Review of nuclear data work for reactors in India
B.P. Rastogi, H.C. Huria

Neutron nuclear data evaluation activities in Japan
S. Igarasi

Nuclear data activities in Romania
S. Râpeanu

Автоматизация процесса проверки информации для библиотеки рекомендованных ядерных данных, программа посошок
В.Е.Колесов, А.С.Кривцов, Н.А.Соловьев

Comparative analysis of evaluated nuclear data files/philosophy
S. Yiftah

Preliminary graphical analysis of selected plutonium isotope cross sections of the ENDF/B-I and II, KEDAK and UKNDL files
D. Ilberg, S. Yiftah

The role of physical trends in neutron data evaluation
J. Csikai, I. Angeli, Z.T. Bödy

Recommended values for (n, 2n) and (n, total) cross sections
Z.T. Bödy, J. Csikai, I. Angeli

Interpretation of trends in total neutron cross sections
I. Angeli, J. Csikai

Establishment of computer libraries of evaluated data and associated computer programs
S. Pearlstein

Technical comments and remarks about ENDF/B, KEDAK and UKNDL
Y. Gur, S. Yiftah

Remarks about UKNDL, ENDF/B-II and KEDAK libraries
G.C. Panini

Формат библиотеки рекомендованных ядерных данных для расчета реакторов
В.Е.Колесов, М.Н.Николаев

Планирование оптимальной совокупности микроскопических экспериментов и оценок, обеспечивающей заданную точность расчета реакторных параметров
Л.Н.Усачев, Ю.Г.Бобков

Role and efficiency of nuclear theory in evaluation - resolved and unresolved resonances
P. Ribon

Energy region of resolved and unresolved resonances – the activity of the CNEN Nuclear Data Group in Bologna, Italy
M. Motta

Research of the best running conditions of nuclear codes for calculation of neutron interaction with heavy deformed nuclei
Y. Kikuchi

Comparison of spherical optical model codes and standard values for nuclear data evaluations
Y. Kikuchi

On the computer codes based on statistical, optical and direct interaction models
S. Igarasi

Nuclear model codes used by the CNEN Nuclear Data Group
V. Benzi

Notes on the availability, quality and comparison of nuclear model codes
V. Benzi

Fast neutron data for titanium
E. Barnard, J.A.M. de Villiers, D. Reitmann, A.B. Smith, E.M. Pennington, J.F. Whalen

Fast neutron cross sections of ^{240}Pu; measured results and a comparison with an evaluated file
A.B. Smith, P.P. Lambropoulos, J.F. Whalen

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