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The Development and Status of the Evaluated Nuclear Data File ENDF/B*

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The Evaluated Nuclear Data File (ENDF) is part of a system developed by the United States Atomic Energy Commission to improve the quality and facilitate the use of nuclear data in applications. The basic elements of the ENDF system are shown in Fig. 1. The widespread use of a single well defined format reduces the number of schemes necessary to process information or compare alternate data sets. The existing network of checking codes that examine the consistency of the data sets and perform physical checks against known nuclear properties helps to ensure the reliability of the file. There are large processing codes that use ENDF directly to prepare multigroup constants or pointwise data sets required for one- two- and three-dimensional diffusion theory, transport theory, or Monte Carlo calculations.

The entire ENDF library consists of two parts, A and B. ENDF/A contains a large and varied collection of evaluations from many sources. It is assembled by the National Neutron Cross Section Center. The contents of the present version of ENDF/A are shown in Fig. 2. The ENDF/B library, on the other hand, contains only one evaluation of the cross sections for each material in the library, and cross sections for all significant reactions that extend well into the MeV range. Versions A and B of the ENDF library use the same format. Only the contents of the two libraries differ. ENDF/B includes data important for fast reactor calculations, but the lack of experimental information sometimes requires reliance on nuclear models or systematics to complete the file. For certain applications ENDF/B is tested in carefully performed and documented benchmark experiments. The information from such integral experiments as well as the results of basic cross section measurements are used to determine ENDF revisions on a periodic basis.

*Research supported by U. S. Atomic Energy Commission.

ENDF/B has been developed and maintained by the Cross Section Evaluation Working Group (CSEWG). CSEWG was organized in 1966 by the AEC for the purpose of assessing data evaluation needs and capability among its contractors and other organizations. It consists of a group of working level representatives from national laboratories, industrial organizations, and academic institutions. Since its beginning CSEWG has provided a forum where data problems can be discussed and evaluations peformed. In addition to developing ENDF/B, CSEWG has achieved a significant level of cooperation of the producer, evaluator and user of data. The objectives of CSEWG listed in Fig. 3 are directed toward the widespread availability of nuclear data reviewed and revised on a regular basis.

The National Neutron Cross Section Center (NNCSC) at Brookhaven National Laboratory administers the activities of CSEWG, acting as secretariat and providing meeting facilities, personnel, preparation and distribution of material as well as other required services. (Fig. 4). CSEWG meetings are held two times each year at BNL. Attendance is limited in order to assure face-to-face discussion, but individuals from laboratories both inside and outside CSEWG are invited when needed. There is no attempt to include all experts on a problem but only as many as are judged essential to deal with technical questions under discussion. It is assumed that laboratory representatives will bring to CSEWG a consensus of the broad data interests of their laboratory even if the subject is outside their own specialty. In addition to general CSEWG meetings, there are subcommittee meetings organized by each subcommittee chairman which are held when CSEWG meets and at other times during the year.

The cooperative effort that began in 1966 produced Version 1 of the ENDF/B library in July 1968. The main emphasis at that time was the gathering of as much data as possible in the new format. The library consisted of some old evaluations borrowed in their entirety, some new evaluations, and some hastily conceived data sets which were in many ways deficient. Testing of the data showed discrepancies, particularly in the underprediction of K_{eff} in fast criticals. Version 2 of the ENDF/B library was issued in April of 1971. The distinctive characteristic of this version was the simultaneous evaluation of the fissile and fertile elements by a task force of evaluators. The data sets were consistent with the best microscopic differential data and cross section ratios. The results were still discrepant with integral benchmark calculations but discrepancies varied systematically with changes in isotopic concentrations and conclusions could be drawn as to which cross sections were playing an important role. The third version of ENDF/B appeared in February 1972. Within the experimental errors of the best microscopic cross section measurements, data were chosen that gave results consistent with carefully performed and documented integral benchmark experiments. In addition, attention was given to more careful specifications of benchmarks and to the intercomparison and subsequent revision of group-constant preparation codes. This latter effort helped to reduce uncertainties caused by disparate methods used at different laboratories.

Within CSEWG several subcommittees assist with the determination of the format and content of ENDF/B. At present CSEWG includes the subcommittees shown in Fig. 5. The two main entry points into the ENDF/B system are the Data Testing and Codes and Formats subcommittees. The final decisions regarding the balance between the physics and user needs are made in those subcommittees.

The Data Testing Subcommittee (1) through its Phase I (microscopic) and Phase II (integral) data testing procedures reviews data sets considered for ENDF/B, resolves all physics questions regarding data, and provides the NNCSC with guidelines for preparations of versions of the ENDF/B library. As part of the review data sets are checked to see whether recent experimental data were considered, whether data sets are complete over the required energy range and whether obvious errors have been eliminated. This committee is responsible for the acceptance, modification, or rejection of all data sets. Data sets approved by the subcommittee become part of the ENDF/B library. Another important responsibility of the Data Testing Subcommittee is the determination of the validity of ENDF/B data in certain applications. The major areas of Phase II testing are the following: a) fast criticals, b) thermal reactors, c) dosimetry, d) shielding, and e) burnup and decay heat of fission products. These results are used in making recommendations for further evaluations and basic data measurements. This Subcommittee searches for cases that can properly be termed "benchmark experiments" in order to test ENDF/B in many areas and aid its development into an application-independent data file.

The Codes and Formats Subcommittee (2) establishes forms in which nuclear data may be included in ENDF/B so as to achieve the best balance between the detail necessary for specifying physics information and the simplicity required to facilitate its use. In addition to these responsibilities the Codes and Formats subcommittee improves the accuracy of codes which use ENDF/B data directly and make recommendations for the development of additional processing or data manipulation codes. Numerous inaccuracies in code algorithms and simple coding errors have been discovered by processing "tedium" materials. Tedium materials are a set of idealized materials created in the ENDF/B format for the purpose of testing processing codes.

Other subcommittees or ad hoc committees are appointed as needed. The Normalization and Standards Subcommittee (3) has the responsibility for continual review of cross sections defined as standards. The Shielding Subcommittee (4) concerns itself with the improvement of data for shielding applications. It interacts with two other data centers besides the National Neutron Cross Section Center: The Radiation Shielding Information Center at Oak Ridge and the Photonuclear Data Center at the National Bureau of Standards. Many important problems have been discussed by the Resonance Region Subcommittee (5) including the use of the Probability Table method in the unresolved resonance region and the critical examination of the need for multilevel formalisms in reactor analysis. The Fission Products Subcommittee (6) has concerned itself with establishing a library of cross sections for burnup calculations and nuclear structure data for decay heat calculations. This portion of the ENDF library is expanding rapidly. In the evaluations nuclear model calculations are used to supplement experimental evidence and the Nuclear Model Codes Subcommittee (7) assists in the determination of which codes are useful and attempts to isolate problems connected with the adoption of these complex codes to different computers. A recent concern of CSEWG is the assignment of error quantities to the evaluations. The purpose of the Error Quantities Subcommittee (8) is to study and make recommendations on the means and procedures for including errors or confidence limits to be used in sensitivity studies that determine the importance of cross section accuracy in special applications. Initially, simplified procedures will be used since an exact specification of errors and their relation to other quantities is very complex. The Non-Neutron Data Subcommittee (9) has been successful in helping to form recommendations for including other than neutron-induced and photon-induced reaction data into the files. Charged-particle cross sections of interest to shielding and fusion applications may be included in a future ENDF/B library.

Version 3 of the ENDF/B library has now been used extensively. There is general confidence that ENDF/B-III is consistent with the best differential integral data available at the time of its development. The results of calculating several fast reactor benchmarks are shown in Fig. 6. There is significant improvement in the results over those obtained using ENDF/B-II data. The agreement between calculated and measured spectral indices is also quite satisfactory but there is not yet satisfactory agreement for all other parameters. Nevertheless, today in the U. S., ENDF/B-III is the chief source of data for fast reactor design.

Version 4 is due for release in early 1974. The starting point for the evaluations of the fissile and fertile elements is the fission cross section of ^{235}U. This cross section is chosen from the best microscopic data over the entire energy range to represent an accurate measurement standard. Using it as a reference, the remaining cross sections for ^{235}U and other fissile and fertile elements are chosen to be within probable error limits of the best differential cross section and ratio measurements and consistent with the best integral measurements.

The materials included in the ENDF/B-IV library can be thought of as fitting into four general categories (Fig. 7): general purpose, scattering-law, fission product, and dosimetry files. The file of general materials should be sufficient for thermal and fast reactor, shielding, space, and weapons applications. To complete the data requirements for specialized areas such as dosimetry, scattering-law, and decay heat studies the general file is supplemented by special purpose files. The data sets in the dosimetry files, for example, instead of being complete in all significant reactions as are materials in the general file, might only specify the activation cross section for a single isotope. New areas such as biomedical and fusion studies will require even more specialized evaluations.

In ENDF/B-IV the general purpose file will contain data for over 100 materials, the dosimetry file will contain 32 materials, the scattering-law file 10 materials, and the fission product file several hundred nuclides. Special attention has been given to data sets for shielding applications. Over 30 materials will have gamma-production data files that are consistent with the neutron data portion of the file. In addition there are photon-interaction data for all materials. Some of the thermal cross sections will be significantly improved in ENDF/B-IV but further reevaluation is expected in this somewhat neglected area after another round of data testing. Also, in ENDF/B-IV the library will contain uncertainty assignments to data but initially only for a limited number of materials.

In conclusion, ENDF/B is the result of a large national cooperative effort to develop and revise on a regular basis and up-to-date library that provides for all its users a highly accurate data base. Recently, an American Nuclear Society study group recommended ENDF/B be used as a standard reference file for fast reactor design and also recommended data be preprocessed into standard group libraries. The availability of a widely tested reference data base should increase the reliability of the cost analysis and the operational and safety design of fast reactors.

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FIGURES

1. Elements of the ENDF System

2. Objectives of CSEWG

3. Relationship between CSEWG and the User Community

4. Contents of ENDF/A

5. CSEWG Subcommittees

6. Benchmarks Results ENDF/B-III Data

7. Components of the ENDF/B-IV Library

Fig. 1

ELEMENTS OF THE ENDF SYSTEM

1. UNIQUELY DEFINED FORMAT
2. CHECKING CODES
3. PROCESSING CODES
4. COMPLETE DATA SETS
5. DATA TESTING
6. REGULAR REVISION OF DATA

Fig. 2

CONTENTS OF ENDF/A

LIBRARY NAME	DATE RECEIVED	NUMBER OF MATERIALS	TOTAL NUMBER OF RECORDS	DESCRIPTION
KEDAK* (KARLSRUHE)	10/26/70	41	--	FAST REACTOR CROSS SECTION SETS
UKNDL (VER. 2) (UNITED KINGDOM)	5/5/71	91	75,508	GENERAL CROSS SECTION DATA SETS
SAND-II (U.S.)	8/5/71	60	26,940	DOSIMETRY CROSS SECTIONS
A.A.E.C. (AUSTRALIA)	9/15/71	192	92,802	CROSS SECTION DATA FOR FISSION PRODUCT NUCLIDES
SPENG* (SWEDEN)	1/5/73	29	1,534	GENERAL CROSS SECTION DATA SETS
KONSHIN & NIKOLAEV (U.S.S.R.)	1/17/73	1	65	(n,f) REACTION FOR ^{235}U
STANDARD (U.S.)	1/25/73	5	264	MEASUREMENTS FREQUENTLY USED AS STANDARDS CONVERTED TO ENDF FORMAT FOR USE IN RENORMALIZATION
C.E.N. COMPILATION (FRANCE)	6/5/73	--	--	COMPILATION OF PROPERTIES OF FISSION PRODUCTS
UKNDL (VER.3) (UNITED KINGDOM)	8/7/73	65	--	GENERAL CROSS SECTION DATA SETS

* NOT CONVERTED TO THE ENDF FORMAT

Fig. 3

OBJECTIVES OF CSEWG

1. TO FORMULATE THE SCOPE AND CONTENTS OF THE REFERENCE LIBRARY (ENDF/B) AND ESTABLISH FORMATS AND PROCEDURES FOR ITS USE.

2. TO DEVELOP THE NECESSARY EVALUATIONS AND PERFORM DIFFERENTIAL AND INTEGRAL TESTING OF DATA AS PART OF A CONTINUAL REVIEW DIRECTED TOWARD THE UPGRADING OF EVALUATIONS.

3. TO PROVIDE RECOMMENDATIONS FOR NEW EVALUATIONS, EXPERIMENTS, AND COMPUTER CODES NEEDED TO IMPROVE THE ACCURACY OF ANALYSES USING NUCLEAR DATA.

Fig. 4

RELATIONSHIP BETWEEN NNCSC-CSEWG and the USER COMMUNITY

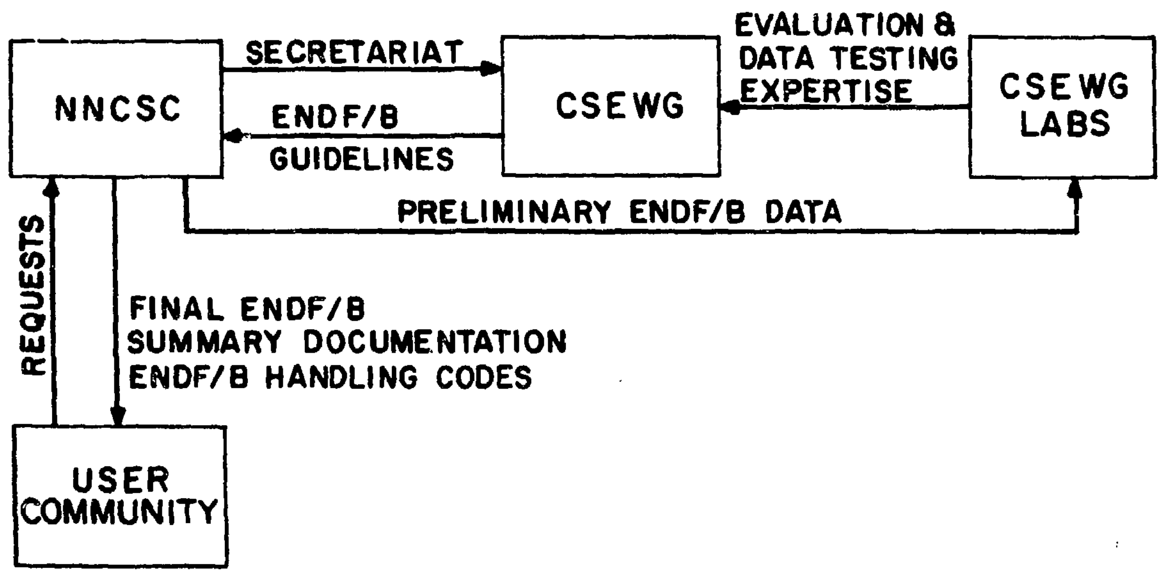

Fig. 5

CSEWG SUBCOMMITTEES

1. DATA TESTING
2. CODES AND FORMATS
3. NORMALIZATION AND STANDARDS
4. SHIELDING
5. RESONANCE REGION
6. FISSION PRODUCTS
7. NUCLEAR MODEL CODES
8. ERROR QUANTITIES
9. NON NEUTRON DATA

Fig. 6

BENCHMARK RESULTS USING ENDF/B-III DATA

ASSEMBLY	ENDF/B-II	ENDF/B-III	MEASURED VALUE
JEZEBEL	0.987	0.999	1.000 ± 0.003
VERA IIA	0.968	0.994	1.000 ± 0.004
ZPR-3-48	0.969	0.998	1.000 ± 0.004
ZEBRA - 3	0.948	0.984	1.000 ± 0.003
GODIVA	0.998	1.004	1.000 ± 0.003
VERA 1B	0.992	0.999	1.000 ± 0.003
ZPR-3-6F	0.992	1.002	1.000 ± 0.002
ZPR-3-11	0.966	0.988	1.000 ± 0.003
ZPR-3-12	0.981	0.997	1.000 ± 0.002
ZEBRA-2	0.966	0.982	1.000 ± 0.004

Fig. 7

COMPONENTS OF THE ENDF/B-IV LIBRARY

1. GENERAL PURPOSE FILE

2. SCATTERING LAW DATA FILE

3. FISSION PRODUCT FILE

4. DOSIMETRY FILE