American Nuclear Society

group-averaged neutron and gamma-ray cross sections for radiation protection and shielding calculations for nuclear power plants

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American National Standard
Group-Averaged Neutron and Gamma-Ray Cross Sections for Radiation Protection and Shielding Calculations for Nuclear Power Plants

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Foreword

(This Foreword is not a part of American National Standard "Group-Averaged Neutron and Gamma-Ray Cross Sections for Radiation Protection and Shielding Calculations for Nuclear Power Plants," ANSI/ANS-6.1.2-2013.)

A need for computer-readable standard reference neutron and gamma-ray cross-section data was identified by American Nuclear Society Standards Subcommittee ANS-6 in 1975. These cross sections are required for materials and energy ranges of importance in radiation protection and shielding calculations for nuclear power plants. It was observed at that time that data sets that did not meet desired standards of documentation and verification were nonetheless becoming de facto standards.

This standard provides guidance in the preparation and verification of neutron and gamma-ray cross-section sets and identifies several sets of standard reference data that meet the procedures specified herein. The identification of standard neutron and gamma-ray data is expected to improve the efficiency of shielding and radiation protection computations by reducing redundant verification and processing operations by each user. In addition, shielding computations are expected to become more accurate as a result of effort having been focused on the development and testing of nuclear data to be used as a standard. A coupled neutron-gamma multigroup cross-section set, referred to as BUGLE, was developed and tested for this purpose. A revised data set, BUGLE-80, was developed in 1980 on the basis of the BUGLE test results, and the BUGLE-80 data set was recommended for shielding and radiation protection calculations in ANSI/ANS-6.1.2-1983. A more detailed coupled neutron-gamma multigroup data set, VITAMIN-C, also was identified as meeting the requirements of ANSI/ANS-6.1.2-1983. The SAILOR broad-group averaged data set was added in ANSI/ANS-6.1.2-1989 as another recommended data set.

ANSI/ANS-6.1.2-1999 cited the BUGLE-96 broad-group cross-section library as the recommended set, replacing both the BUGLE-80 and the SAILOR sets. The more detailed VITAMIN-B6 set was also cited as a replacement for the VITAMIN-C set. Both multigroup cross-section sets were based on the most recent version of the evaluated point cross-section library available at the time the standard was prepared, i.e., ENDF/B-VI, Release 3. ENDF/B-VI contained numerous significant changes to nuclear data relative to earlier versions of ENDF/B. Improved experimental data and model predictions are included, and several format changes were made to provide for better representation of the underlying physics and the extension to higher energies.

The present edition of this standard cites the BUGLE-B7 broad-group cross-section library, which is an update of the BUGLE-96 library, and the corresponding fine-group cross-section library VITAMIN-B7, which is an update of the VITAMIN-B6 library, as good examples of data sets for use in radiation protection and shielding calculations. Both VITAMIN-B7 and BUGLE-B7 are based on ENDF/B-VII, Release 0. This edition of the standard also recommends generating problem-specific libraries if the group structure of the broad-group cross-section library BUGLE-B7 will not provide results with sufficient accuracy. An example of where this recommendation should be applied is for cases where thermal neutrons play an important role in calculations.

This standard is related to ANSI/ANS-19.1-2002 (R2011), "Nuclear Data Sets for Reactor Design Calculations." The scope of that standard includes data of importance for reactor core design, while ANS-6.1.2 covers radiation transport and shielding applications, especially for nuclear power plants. This standard is also related to ANSI/ANS-19.3-2011, "Steady-State Neutronics Methods for Power Reactor Analysis."

This standard is intended to prescribe recommended practices. The data sets identified are those a novice may use with some confidence and should be seriously considered by the expert. In cases where experts select to use other data sets, they would be expected to have strong reasons why the reference data sets provided herein were not used.

This revision references documents and other standards that may have been superseded or withdrawn at the time this standard is applied. A statement has been included in the references section that provides guidance on the use of references.

This standard does not incorporate the concepts of generating risk-informed insights, performance-based requirements, or a graded approach to quality assurance. The user is advised that one or more of these techniques could enhance the application of this standard.

The membership of Working Group ANS-6.1.2 at the time of the preparation and approval of this revision of the standard included the following:

F. A. Alpan (Chair), *Westinghouse Electric Company*

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Contents

Group-Averaged Neutron and Gamma-Ray Cross Sections for Radiation Protection and Shielding Calculations for Nuclear Power Plants

1 Scope

This standard specifies group-averaged neutron and gamma-ray cross sections and related group-averaged or derived data for the energy range and materials of importance in radiation protection and shielding calculations for nuclear power plants.

2 Acronyms and definitions

2.1 List of acronyms

ANS: American Nuclear Society

CSEWG: Cross Section Evaluation Working Group

LWR: light water reactor

NEA: Nuclear Energy Agency

NNDC: National Nuclear Data Center

ORNL: Oak Ridge National Laboratory

RSICC: Radiation Safety Information Computational Center

2.2 Shall, should, and may

shall, should, and may: The word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and the word "may" is used to denote permission, neither a requirement nor a recommendation.

2.3 Definitions

The following definitions apply for purposes of this standard. Other specialized terms conform to *Glossary of Terms in Nuclear Science and Technology* [1].[1)]

cross-section processing code: A computer code that converts data in ENDF-6 [2] format to a form that is appropriate for use in applications. A cross-section processing code performs calculations such as resonance reconstruction, Doppler broadening, and multigroup averaging.

evaluated nuclear cross-section data: Microscopic cross-section representations derived from basic experimental data, from nuclear models and systematics, and may include the consideration of integral measurements.

Evaluated Nuclear Data File: Evaluated nuclear data file stored using a specified format and procedure. Examples are ENDF/B-VII.1 [3], JEFF-3.1.2 [4], and JENDL-4.0 [5, 6, 7, 8].

Evaluated Nuclear Data File/B (ENDF/B): A U.S.-evaluated nuclear data file prepared and reviewed by subject matter experts that is coordinated and maintained by the Cross Section Evaluation Working Group (CSEWG) and the National Nuclear Data Center (NNDC) at Brookhaven National Laboratory.

experimental benchmark: An experiment for which conclusions can be drawn as to the accuracies of calculational models and the underlying nuclear data. An experimental benchmark contains the following:

- a complete description of the conditions under which the experiment took place, including input data such as reactor geometry, material compositions, core power distribution, relevant material temperatures, and experimental conditions specified in sufficient detail to model or to replicate the experiment;
- measured data and their associated uncertainties along with a complete specification of data correlations.

[1)] Numbers in brackets refer to corresponding numbers in Sec. 8, "References."