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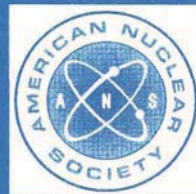
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**neutron and gamma-ray cross sections  
for nuclear radiation protection  
calculations for nuclear power plants**

**an American National Standard**

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**ANSI/ANS-6.1.2-1999**

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

Secretariat  
**American Nuclear Society**

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**Foreword** (This Foreword is not a part of American National Standard Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants, ANSI/ANS-6.1.2-1999.)

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 nuclear radiation protection and shielding calculations for nuclear power plants. It was observed at that time that data sets not meeting high standards of documentation and verification were 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 which meet the procedures specified. The identification of standard neutron and gamma-ray data is expected to improve the efficiency of shielding and radiation protection computations by reducing redundant validating and processing operations by each user. In addition, shielding computations are expected to become more accurate as a result of the focusing of effort 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 identified as meeting the requirements of the standard. The BUGLE-80 data set uses a multigroup energy structure which permits useful shielding and radiation protection calculations. A more detailed coupled neutron-gamma multigroup data set, VITAMIN-C, also was identified as meeting the requirements of the standard. The SAILOR cross section set was added to the standard in 1987-88.

The present edition of this standard cites the BUGLE-96 broad-group cross section library as the recommended set, replacing both the BUGLE-80 and SAILOR sets. The more detailed VITAMIN-B6 set is also cited as a replacement for the VITAMIN-C set. Both are based on the most recent version of the evaluated cross-section library, ENDF/B-VI, Release 3. ENDF/B-VI contains numerous significant changes to available 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.

This standard is related to American National Standard Nuclear Data Sets for Reactor Design Calculations, ANSI/ANS-19.1-1989. 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 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. The expert might be expected to provide strong reasons why he did not use the reference sets if he selects other data. The working group was unanimous in its decision to recommend specific data sets.

The membership of Working Group ANS-6.1.2 at the time it prepared this standard was:

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Subcommittee ANS-6 had the following membership at the time of its approval of this standard:

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Consensus Committee N17, Research Reactors, Reactor Physics, and Radiation Shielding, had the following membership at the time it reviewed and approved this standard:

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A. D. Callihan ..... Individual  
R. E. Carter ..... Individual  
D. Cokinos ..... Brookhaven National Laboratory  
A. De La Paz ..... Vista Technologies  
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This is a preview of "ANSI/ANS-6.1.2-1999 ...". [Click here to purchase the full version from the ANSI store.](#)



# Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power Plants

## 1. Scope

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

## 2. Definitions

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

**evaluated data.** Microscopic cross section representations derived from basic experimental data, from nuclear models and systematics, and from consideration of integral measurements.

**Evaluated Nuclear Data File/B (ENDF/B).** An evaluated nuclear data file prepared and reviewed by specialists in the field and coordinated and maintained by the Cross Section Evaluation Working Group (CSEWG) and the National Nuclear Data Center at Brookhaven National Laboratory.

**experimental benchmark.** Integral experiment for which measurements are of sufficient accuracy and for which experimental conditions are specified in sufficient detail so that conclusions may be drawn as to the accuracies of calculational models and cross section data.

**group-averaged data.** Evaluated data averaged over energy groups (intervals) as weighted by specified functions.

**integral experiment.** Experiment carried out for measurement of quantities proportional to energy and space-integrated radiation fields in bulk matter representing shielding configurations.

**neutron and gamma-ray cross sections.** Microscopic cross sections for the interactions of

neutrons and gamma-rays with matter, including cross sections for emission of neutrons and gamma-rays as well as cross sections for the material effects of neutrons and gamma-rays. The cross sections may be averaged over energy intervals (groups) for purposes of application.

**numerical benchmark.** Specification of composition and geometry of bulk material and radiation sources, and of the objects of calculation, in a detail that is required to determine the accuracies of various calculational methods, usually by comparison with an accepted method.

**reference data.** Published and readily available tables of values of physical constants. These data may be available in the form of computer readable media.

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

**standard reference data.** Reference data which have been reviewed by a standards organization and found to meet minimum requirements for specified purposes. For purposes of this standard, standard Evaluated Data Sets and standard Group-Averaged Data Sets are standard reference data.

## 3. Preparation and Verification of Neutron and Gamma-Ray Cross Sections

**3.1 Evaluated Microscopic Cross Sections.** Evaluated microscopic cross sections shall be derived from documented and reviewed information, including basic experimental data, nuclear models and systematics, and integral experiments. The evaluated cross sections shall be expressed as unique physical parameters and piecewise-continuous functions of incident particle energy, of secondary particle energy, and of secondary particle angle with respect to the incident particle direction. The evaluation shall be in sufficient detail for shielding applications

<sup>1</sup> Numbers in brackets refer to corresponding numbers in Section 8, References.