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# American Nuclear Society

## determination of steady-state neutron reaction-rate distributions and reactivity of nuclear power reactors

## an American National Standard

**WITHDRAWN**

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# Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors

Secretariat
**American Nuclear Society**

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# American National Standard

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Comments on this standard are encouraged and should be sent to Society Headquarters.

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# Foreword

(This Foreword is not a part of American National Standard for the Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors, ANSI/ANS-19.3-2005.)

It is the intent of this American National Standard to provide guidance for performing and validating the sequence of calculations leading to prediction of steady-state neutron reaction-rate spatial distributions and reactivity of nuclear reactors, and to provide guidelines by which the adequacy of design calculations may be demonstrated. This standard recognizes the diversity of the calculational procedures employed in reactor design. Consequently, the major thrusts of this standard are in the areas of verification, validation, and documentation. This standard is intended to cover reactor-physics calculations for the entire nuclear industry, from fast to thermal power reactors. Since many different kinds of calculations are performed, each having its own requirements for accuracy and validation, it is necessary that this standard be of a general nature. It should be noted that this standard does not endorse or exclude the application of any methodology that has been adequately verified, validated, tested, and demonstrated to yield, within acceptable errors, the desired reactor-physics parameters.

For illustrative purposes, a list of computer codes currently being used throughout the nuclear industry is presented in the Appendix. This Appendix, however, is not part of the Standard.

Compliance with the intent of this standard can be demonstrated for an intended area of applicability of the calculational system used by meeting the following requirements:

(1) Selection of models and methods.

   (a) Consideration of all conditions of reactor composition, temperature, and configuration that significantly affect the calculated quantities and justification of the resulting model approximations,

   (b) Preparation of multigroup constants, if employed, in conformance with American National Standard "Nuclear Data Sets for Reactor Design Calculations," ANSI/ANS-19.1-2002, through the use of an application-dependent energy-spectrum calculation,

   (c) Justification of geometric and neutronic transport approximations utilized in the spectrum calculation,

   (d) Inclusion of all important space and energy effects in the calculation utilized for the generation of few-group cross sections, if few-group cross sections are employed,

   (e) Demonstration of capability, as required by the application, to retrieve required neutron reaction rates in the physical reactor components from the computations and to justify any assumptions that need to be made in order to perform this retrieval,

   (f) Justification of the spectrum calculation interval used in depletion calculations and justification that the numerical integration time step is sufficiently small to ensure numerical stability and accuracy appropriate to the application;

(2) Calculational system validation. Establishment of the degree of agreement over a limited area of applicability by correlating experimental results or results of calculations from a more accurate model with results obtained from the system being validated;

(3) Evaluation of accuracy. Evaluation of accuracy and range of applicability of data and methods by establishment of biases and uncertainties, with confidence levels, for the calculations that include allowance for uncertainties in the comparison data;

(4) Documentation. Documentation of details of the preceding procedures.

It is the intent of this standard to require the individual to (a) give careful consideration to those physical and numerical effects that may contribute to the validity of results; (b) document the reasons for the choice of calculational path; and (c) validate the calculational system used over the intended range of applicability by testing it against appropriate experiments, numerical benchmarks, or previously validated methods.

The requirement for documentation is a crucial part of this standard and will provide an auditable path. Areas omitted due to proprietary consideration shall be noted where possible.

The most important ways in which this revision differs from its earlier version, ANSI/ANS-19.3-1995, are as follows:

(1) The passages on common practices for pressurized water reactors, boiling water reactors, and liquid metal reactors have been revised to reflect the significant advances in reactor-physics methods and computer codes made since the last revision. New passages have been added for heavy water reactor and high-temperature gas-cooled reactor methods;

(2) The Appendix, including the list of commonly used computer codes, has been updated. This revision reflects rapid development in two areas, namely,

    (a) significant advances in reactor-physics methods and calculational procedures as a result of a rapidly increasing experience base of operating power reactors,

    (b) computer hardware and operating software developments that have permitted many traditional approximate methods to be placed, with increased performance and user productivity as a result.

This standard for reactor-physics calculations will undergo review and revision within 5 years. Suggestions for the improvement of this standard will be welcome. They should be sent to the attention of the Standards Department, American Nuclear Society, 555 N. Kensington Avenue, La Grange Park, IL 60526. Questions arising from the use of this standard or requests for clarification of portions of the standard should be sent in writing to the above address.

This standard was developed and later revised by Working Group ANS-19.3 of the American Nuclear Society, which at the time of this revision had the participation of the following members:

B. Rouben, *Atomic Energy of Canada Limited*
R.-T. Chiang, *General Electric*
D. M. Cokinos, *Brookhaven National Laboratory*
P. Finck, *Argonne National Laboratory*
G. Hobson, *Framatome Technologies*
B. R. Moore, *General Electric*
R. D. Mosteller, *Los Alamos National Laboratory*
C. T. Rombough, *CTR Technical Services*
Y. Shatilla, *Westinghouse Electric Company*
S. B. Thomas, *Duke Power*

The membership of Subcommittee ANS-19 at the time of its review and approval of this standard was the following:

D. M. Cokinos, Chair, *Brookhaven National Laboratory*
C. T. Rombough, Secretary, *CTR Technical Services*

S. Baker, *Transware Enterprises*
M.C. Brady Raap, *Pacific Northwest National Laboratory*
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B. Rouben, *Atomic Energy of Canada Limited*
A. Weitzberg, *Retired*

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

T. M. Raby (Chair), *National Institute of Standards and Technology*
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(Alt. A. Adams, *U.S. Nuclear Regulatory Commission*)
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# Contents

# Determination of Steady-State Neutron Reaction-Rate Distributions and Reactivity of Nuclear Power Reactors

## 1 Introduction

The design and operation of nuclear reactors require knowledge of the conditions under which a reactor will be critical, as well as the degree of subcriticality or supercriticality when these conditions change. In addition, knowledge is required of the spatial distribution of neutron reaction rates in reactor components as a prerequisite, for example, for inferring temperature distributions and thus the satisfaction of thermal limit requirements. Both reaction-rate spatial distributions and reactivity can be and have been measured by suitable experimental techniques, either in mock-ups or in the operating reactors themselves. These quantities can also be calculated by various techniques. As nuclear cross sections have become more accurate and as calculational methods have been refined, the tendency has been to rely more heavily on calculations. Available reactor experimental data have been used to validate the calculations within reasonable margins.

## 2 Scope

This standard provides guidance for performing and validating the sequence of steady-state calculations leading to prediction, in all types of commercial nuclear reactors, of

(1) reaction-rate spatial distributions;

(2) reactivity;

(3) change of isotopic compositions with time.

The standard provides

(1) guidance for the selection of computational methods;

(2) criteria for verification and validation of calculational methods used by reactor core analysts;

(3) criteria for evaluation of accuracy and range of applicability of data and methods;

(4) requirements for documentation of the preceding.

## 3 Definitions

### 3.1 Limitations

The following definitions are of a restricted nature for the purpose of this standard. Other specialized terms are defined in *Glossary of Terms in Nuclear Science and Technology* [1][1)] and in the definition sections of standards specified in Section 9, "References."

### 3.2 Glossary of terms

**application-dependent multigroup:** A discrete energy-group structure that is intermediate between the application-independent-multigroup structure and a few-group structure. The application-dependent-multigroup structure may be such that the group constants are dependent on reactor composition through an estimated neutron energy spectrum. An application-dependent-multigroup data set is one type of averaged data set.

**application-independent multigroup:** A discrete energy-group structure that is sufficiently detailed that the group constants may be considered as being independent of reactor composition, geometry, or spectrum for a wide range of reactor analysis. The application-independent-multigroup structure may be employed directly in reactor-design spec-

[1)]Numbers in brackets refer to corresponding numbers in Section 9, "References."