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steady-state neutronics methods for power reactor analysis

an American National Standard

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American National Standard Steady-State Neutronics Methods for Power Reactor Analysis

Secretariat
American Nuclear Society

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American National Standard

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(3) the purposes of the inquiry;

(4) the inquiry stated in a clear, concise manner;

(5) a proposed reply, if the inquirer is in a position to offer one.

Inquiries should be addressed to

American Nuclear Society
ATTN: Standards Administrator
555 N. Kensington Avenue
La Grange Park, IL 60526

or standards@ans.org

Foreword

(This Foreword is not a part of American National Standard "Steady-State Neutronics Methods for Power Reactor Analysis," ANSI/ANS-19.3-2011.)

The intent of this American National Standard is to provide guidance for developing, validating, and utilizing steady-state neutronics methods to calculate neutron reaction-rate spatial distributions, power distributions, and effective neutron multiplication constants of nuclear power reactors and to provide guidelines by which the adequacy of design calculations may be demonstrated. This standard recognizes the diversity of the calculation procedures employed in reactor design. Consequently, the major thrusts of this standard are in the areas of methodology, 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 neutronics methods have been utilized for analyzing power reactors, and each has its own validation requirements for accuracy, it is necessary that this standard be of a general nature. Furthermore, this standard does not endorse or exclude the application of any methodology that has been adequately verified, validated, tested, and demonstrated to yield reliable 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 calculation system used, if the following requirements are met:

(1) Selection of models and methods

(a) consideration of reactor configuration, composition, and all conditions of operation that significantly affect the calculated quantities and justification for the resultant model approximations,

(b) preparation of multigroup cross sections and other parameters, if employed, in conformance with ANSI/ANS-19.1-2002 (R2011), "Nuclear Data Sets for Reactor Design Calculations," through the use of an application-dependent energy spectrum estimate,

(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 these 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 nuclide depletion calculations and justification that the numerical integration time step is sufficiently small to ensure numerical stability and accuracy appropriate to the application;

(2) Calculation system verification and validation

Establish degree of agreement between results obtained with the system being verified with results of experiments or of calculations using a more accurate model, over the intended area of applicability for the system being verified;

(3) Evaluation of accuracy

The accuracy and range of applicability of data and methods should be evaluated by establishment of biases and uncertainties, with degree of confidence, for the calculations that include allowance for uncertainties in the comparison data;

(4) Documentation.

The intent of this standard is 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 selecting a specific calculation path; and (c) validate the calculation system used over the intended range of applicability by testing it against appropriate experiments, numerical benchmarks, and/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-19.3-2005, 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 calculation 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 replaced, with increased performance and user productivity as a result.

This standard might reference documents and other standards that have been superseded or withdrawn at the time the standard is applied. A statement has been included in Sec. 9, “References,” 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.

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.

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:

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R.-T. Chiang, *AREVA NP*
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S. B. Thomas, *Duke Power*
P. Yarsky, *U.S. Nuclear Regulatory Commission*

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, Inc.*

A. Attard, *U.S. Nuclear Regulatory Commission*
S. Baker, *Transware Enterprises, Inc.*
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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:

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(Alt. R. D. Zimmerman, *American Institute of Chemical Engineers*)
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Steady-State Neutronics Methods for Power Reactor Analysis

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 proper power and temperature distributions to ensure the satisfaction of thermal-limit and safety-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. Available reactor experimental data have been used to validate the steady-state neutronic calculations within reasonable margins. As more accurate nuclear cross sections and more significantly improved calculation methods are available, steady-state neutronic calculations have been utilized extensively for nuclear fuel and core designs and analyses and thus become increasingly important.

2 Scope

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

(1) reaction-rate spatial distributions;

(2) reactivity;

(3) change of nuclide compositions with time.

The standard provides the following:

(1) guidance for the selection of computational methods;

(2) criteria for verification and validation of calculation 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.

Note that the use of mixed uranium-plutonium oxide (MOX) fuel has been taken as out of scope for this revision of the standard. It will be taken into account in the next revision.

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 Sec. 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 spectrum

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