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Steady-state neutronics methods for power-reactor analysis

Méthodes stationnaires en neutronique pour l'analyse des réacteurs de puissance

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Foreword

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The procedures used to develop this document and those intended for its further maintenance are described in the ISO/IEC Directives, Part 1. In particular the different approval criteria needed for the different types of ISO documents should be noted. This document was drafted in accordance with the editorial rules of the ISO/IEC Directives, Part 2 (see www.iso.org/directives).

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This document was prepared by Technical Committee ISO/TC 85, *Nuclear Energy, Nuclear Technologies, and Radiological Protection*, Subcommittee SC 6, *Reactor Technology*. This document is based on a standard developed by the American Nuclear Society (ANS) of which the current version is ANSI/ANS-19.3-2011 (R2017)[2].

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 become available and more refined calculation methods are developed, the reliability of the results of the steady-state power reactors will be considerably enhanced.

Steady-state neutronics methods for power-reactor analysis

1 Scope

This document provides guidance for performing and validating the sequence of steady-state calculations leading to prediction, in all types of operating UO_2-fuel commercial nuclear reactors, of:

— reaction-rate spatial distributions;

— reactivity;

— change of nuclide compositions with time.

The document provides:

a) guidance for the selection of computational methods;

b) criteria for verification and validation of calculation methods used by reactor core analysts;

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

d) requirements for documentation of the preceding.

2 Normative references

There are no normative references in this document.

3 Terms and definitions

ISO and IEC maintain terminological databases for use in standardization at the following addresses:

— ISO Online browsing platform: available at https://www.iso.org/obp

— IEC Electropedia: available at http://www.electropedia.org/

3.1 Terms

3.1.1
application-dependent multigroup
discrete energy-group structure that is intermediate between the application-independent multigroup structure and a few-group structure

Note 1 to entry: The application-dependent multigroup structure can 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.

3.1.2
application-independent multigroup
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

Note 1 to entry: The application-independent multigroup structure can be employed directly in reactor-design spectrum calculations, or it can be employed to generate group constants in an application- dependent multigroup structure. An application-independent multigroup data set is one type of averaged data set.

3.1.3
cell
one or more reactor sections with associated coolant (and possibly additional moderator and structural material) which, for computational purposes, are assumed to form a spatially repeating array in the reactor

Note 1 to entry: The simplest example of a cell is the "pin cell" in which a single fuel rod or pin is surrounded by coolant (e.g. light water, heavy water, or sodium). Another example is a bundle of fuel rods cooled by heavy water within a housing, surrounded by a heavy water moderator space.

Note 2 to entry: More complex geometric configurations are also used for some applications. These are often referred to as "supercells", or sometimes "(fuel) assembly cells", although the exact definition of the term varies greatly between reactor types and is even somewhat subjectively defined for a particular reactor type. Supercells, in the context of this document, represent more complex "cell" configurations which involve a collection of contiguous cells forming an assumed repeating array within the reactor, or augmented cells incorporating additional regions to serve as a computational artifice, e.g. to account for significant spectrum effects due to compositions outside the cell, or cell configurations including a reactivity device in addition to fuel, coolant, moderator and poison.

3.1.4
data set
collection of microscopic cross-sections and nuclear constants encompassing the range of materials and reaction processes needed for the application area of interest

3.1.4.1
averaged data set
data set prepared by averaging an evaluated data set or a processed continuous data set with a specified weighting function over a specified energy group structure

Note 1 to entry: The group structure and weighting functions may be selected to be application dependent. Application-independent averaged data sets for a wide range of reactor analysis, e.g. light water reactors, are dealt with in American National Standard Nuclear Data Sets for Reactor Design Calculations, ANSI/ANS-19.1-2002 (R2011)[1].

3.1.4.2
evaluated data set
data set which is completely and uniquely specified over the ranges of energy and angles important to reactor calculations

Note 1 to entry: Such a data set is based upon available information (experimental measurement results and nuclear theories) and employs a judgment as to the best physical description of the interaction process.

Note 2 to entry: An evaluated data set is intended to be independent of reactor composition, geometries, energy group structures, and spectra.

3.1.4.3
processed continuous data set
data set prepared by expansion or compaction of an evaluated data set using specified algorithms

Note 1 to entry: Such a data set is intended to be independent of reactor compositions, geometries, energy-group structures, and spectra.

3.1.5
experimental data
any experimentally measured quantity or quantities

Note 1 to entry: As such it is applied herein to both differential cross-section measurements and integral measurements (e.g. control-rod worth) obtained from reactor experiments or operations.

3.1.6
few-group
energy-group (typically 2-group) structure that is adopted for a particular application

Note 1 to entry: The few-group constants for a region are dependent on a specific reactor composition and geometry through a calculated energy spectrum, and are also dependent on temperature.

3.1.7
lattice
lattice cell
normally refers to a fuel-assembly cell with its associated immediate environment, such as the volume of moderator associated with it

3.1.8
calculation method
mathematical equations, approximations, assumptions, associated numerical parameters, and calculational procedures that yield the calculated results

Note 1 to entry: When more than one step is involved in the calculation, the entire sequence of steps comprises the "calculation method".

3.1.9
reactivity
property of the whole reactor, not just of a given material composition, is the ratio of the net production rate of neutrons (excess of neutrons produced by fission over those absorbed) to the production rate due to fissions

Note 1 to entry: Quantitatively, the core reactivity, ρ, can be represented as:

$$\rho = (\lambda - 1)/\lambda = 1 - (1/k_{\text{eff}})$$

where

λ is the eigenvalue of the steady-state neutron balance equation;

k_{eff} is the effective neutron multiplication constant.

Note 2 to entry: quantity (1 minus the eigenvalue of the steady-state neutron balance equation, written as:

$$M\Phi = \lambda F\Phi$$

where

Φ is the neutron flux;

F is the neutron yield operator;

M is the scattering, absorption, and leakage operator.

Note 3 to entry: The effective multiplication factor k_{eff} is the inverse of λ. Reactivity is a unitless, pure number. It is, however, often written in terms of smaller "units", such as milli-k = 0,001, pcm = 0,000 01 = 10^{-5} or "dollars" (and cents), where 1 dollar is taken as the value of the delayed-neutron fraction in the system of interest.

3.1.10
validation
process of determining the degree to which a model is an accurate representation of the real world from the perspective of the intended uses of the model

3.1.11
verification calculation
process of determining that a model implementation accurately represents the developer's conceptual description of the model and the solution to the model

3.2 Abbreviations

BWR boiling water reactor

HTGR high temperature gas cooled reactor

HWR heavy water reactor

LMR liquid metal reactor

PWR pressurized water reactor

4 Relation to other standards

The following American National Standards are related to this document:

— *Nuclear Data Sets for Reactor Design Calculations*, ANSI/ANS-19.1-2002 (R2011)[1], defines the criteria to be employed in the preparation of application-independent cross-section data files from experimental data and theoretical models. This document covers subsequent space and energy averaging processes which may be employed to prepare cross-sections for use in the representation of the core and its environment, and the subsequent calculation of the spatial distribution of neutron reaction rates in the core and of the core reactivity. There may be many ways of carrying out the space and energy averaging to obtain few-group cross-sections, and no unique path for the preparation or use of cross-sections employed in design calculations is defined, required, or recommended by this standard.

— *Guide for Acquisition and Documentation of Reference Power Reactor Physics Measurements for Nuclear Analysis Verification*, ANSI/ANS-19.4–2017[3]; and *Requirements for Reference Reactor Physics Measurements,* ANSI/ANS 19.5-1995; W2005[4].

Validation of calculation systems requires comparison with available integral experimental results. The preceding standards contain criteria for performing and documenting such experiments, in order to be most useful for this purpose.

— *Determination of Thermal Energy Deposition Rates in Nuclear Reactors*, ANSI/ANS 19.3.4–2002 R2017[5], provides criteria for the establishment of the thermal energy deposition rate distribution within a nuclear reactor core. Since the accuracy with which this can be done is dominated by the accuracy with which neutron reaction rates can be calculated, ANSI/ANS-19.3.4–2002; R2017 is closely related to ANSI/ANS–19.3–2011; 2017.

— *Quality Assurance Program Requirements for Nuclear Facility Applications*, ANSI/ASME-NQA 1 2015[6]. This standard deals with quality assurance, including that for computer programs.

— *Guidelines for the Documentation of Computer Software*, ANSI/ANS 10.3 1995 W2005[8]. This standard includes requirements for computer programs.

— *Verification and Validation of Non-Safety Related Scientific and Engineering Computer Programs for the Nuclear Industry,* ANSI/ANS-10.4–2008; W2016[9]. This standard deals with requirements or verifying and validating computer codes, such as those used for neutronics calculations.

— *Accommodating User Needs in Scientific and Engineering Computer Software*, ANSI/ANS-10.5–2006; R2016[10]. This standard deals with methods to respond to users' requirements in computer programs.

5 Methods of calculation

5.1 General

Calculations within the scope of this document would typically be performed in a sequence of steps. A typical sequence might be:

a) *Spectrum Calculation*. Averaged-data set cross-sections, nuclide number densities, and geometrical information (usually repeating cells or supercells) are used to calculate an application-dependent neutron spectrum for each different reactor region or composition.

b) *Cross-Section Collapsing*. Collapsing averaged-data set cross-sections to few-group form, using spectrum calculated in a) above.

c) *Cross-Section Homogenization*. The spectra obtained above are used to homogenize cross-sections and number densities over pin cells and assemblies.

d) *Flux Distributions and Reactivity*. The broad-group cross-sections and geometrical information about the reactor obtained above is used to calculate reactivity and few-group flux spatial distributions in the reactor.

e) *Reaction Rates Calculations*. The preceding information is used to compute reaction rates in physical reactor components.

f) *Exposure*. Calculation of changes in nuclide composition of fuel and possibly other reactor components with exposure are obtained based on the above data.

Not all steps in the sequence would normally be executed for a given problem. It is not a requirement of this document that a particular sequence of calculations, such as the one previously listed, be used. Similarly, the use of the preceding sequence does not, in itself, demonstrate compliance with this document. The use of a specific calculation procedure shall be justified by the procedure presented in Clause 6. However, the preceding sequence does provide an adequate framework within which most of the problems in steady-state reactor physics calculations can be discussed. Therefore, each of the aforementioned steps will be discussed in later passages of this subclause.

A summary of the requirements of this document is given in Clause 8.

5.2 Conditions to be considered

Consideration shall be given to all conditions which significantly affect the calculated quantities. The method of calculation shall be capable of treating the reactor composition or configuration under the conditions being studied.

Important conditions that may be significant include, but are not limited to:

a) presence of control elements (rods, cruciforms, or other forms), and degradation of the effectiveness of control elements;

b) presence and spatial distribution of burnable or soluble absorbers;

c) presence of adjacent, unlike fuel assemblies;

d) composition and geometric layout of fuel in an assembly;

e) dependence of coolant or moderator density upon conditions, or their spatial dependence;

f) depletion dependent conditions, including previous power history, coolant-density history, control-element history, and soluble-absorber history of fuel assemblies;

g) presence of materials or conditions, or both, outside the core, such as the core shroud in a BWR;

h) presence of sources, detectors, structural materials, and experimental devices;

i) spatial variations in temperatures;

j) fuel temperature;

k) spatial and temporal variations of important nuclides, e.g. xenon, samarium, and actinides.

5.3 Fine-group cross-sections

5.3.1 Basic data

The primary sources of basic nuclear data that are used for the generation of fine-group constants are evaluated data sets. Examples of these are the ENDF, JENDL, BROND, JEFF, and CENDL evaluated data sets. The properties and criteria for selecting these sources of basic nuclear data are specified in ANSI/ANS 19.1-2002 R2011[1].

5.3.2 Preparation of fine-group cross-sections

The preparation of application-dependent fine-group constants from existing application-independent fine-group constants shall entail use of an application-dependent energy spectrum estimate (see 5.3.3 and 5.4). This procedure employs a weighting spectrum that is selected to preserve important system-dependent characteristics during the averaging process. These characteristics usually include reaction rates, and may include other quantities.

5.3.3 System dependent spectrum calculations

The fine-group cross-section set (5.3) should be used in the calculation of the neutron energy spectra in the system under investigation. The energy spectra are established by the geometry, material composition, and the operating conditions of the reactor in an interplay of neutron leakage with reactions such as absorptions and scattering. The neutron energy spectrum may vary from one region of the core to another and it may be necessary to compute the spectra for several representative regions of the reactor core.

5.3.4 Weighting function

The fine-group constants can be sensitive to the selection of an energy dependent weighting spectrum, and to the choice of group structure. The smaller the number of energy groups, the greater the sensitivity will be. Therefore, an estimate of the reactor spectrum is needed and should be obtained from measurements in identical or similar reactors, or from analytical models of neutron slowing down or source spectra. It should be noted that results may be sensitive to the modelling of the spectra and the choice of group structure.

5.4 Preparation of broad-group libraries

5.4.1 General

There are three distinct steps for generating broad-group libraries:

a) processing of continuous or point-wise cross-sections, accounting for self-shielding and Doppler broadening effects and collapsing these data into a fine-group library using an appropriate spectrum;

b) performance of fine-group transport calculation for a simplified model of the reactor to obtain a fine-group spectrum;

c) utilization of the fine-group spectrum to obtain broad-group libraries.

