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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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ISO copyright office
CP 401 • Ch. de Blandonnet 8
CH-1214 Vernier, Geneva
Phone: +41 22 749 01 11
Fax: +41 22 749 09 47
Email: copyright@iso.org
Website: www.iso.org

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Foreword

ISO (the International Organization for Standardization) is a worldwide federation of national standards bodies (ISO member bodies). The work of preparing International Standards is normally carried out through ISO technical committees. Each member body interested in a subject for which a technical committee has been established has the right to be represented on that committee. International organizations, governmental and non-governmental, in liaison with ISO, also take part in the work. ISO collaborates closely with the International Electrotechnical Commission (IEC) on all matters of electrotechnical standardization.

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

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