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Reactor technology — Power reactor analysis — Steady-state neutronics methods

Technologie du réacteur — Analyse des réacteurs de puissance — Méthodes stationnaires en neutronique

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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 document 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-19.3-2022[2].

This second edition cancels and replaces the first (ISO 18075:2018), which has been technically revised.

The main changes are as follows:

— addition of new terms in Clause 3;

— figures slightly revised;

— 5.8.5 was amended so that CANDU is only an example;

— correction of internal references;

— Table A.1 corrected;

— editorial amendments in the whole document.

Any feedback or questions on this document should be directed to the user’s national standards body. A complete listing of these bodies can be found at www.iso.org/members.html.

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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 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 improved calculation methods have become available, steady-state neutronic calculations have been utilized extensively for nuclear fuel and core designs and analyses and thus have become increasingly important.

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Reactor technology — Power reactor analysis — Steady-state neutronics methods

1 Scope

This document 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:

— 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, and

d) requirements for documentation of the preceding.

2 Normative references

There are no normative references in this document.

3 Terms, definitions and abbreviations

For the purposes of this document, the following terms and definitions apply.

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

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

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

3.1 Terms and definitions

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.

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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
averaged data set
data set prepared by averaging an evaluated data set or a processed continuous-energy 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 or application independent, e.g, light water reactors (LWRs), are dealt with in ANSI/ANS 19.1-2019[1].

3.1.4
cell
supercell
one or more reactor components with associated coolant (and possibly additional moderator and structural material) that, 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 that 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.

3.1.5
collapse
method by which a many energy group set of cross sections is reduced to fewer energy groups

3.1.6
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

3.1.7
dilution cross section
cross section, which refers to a background cross section that represents the effects of other isotopes in a material, particularly in the context of self-shielding in nuclear reactions

3.1.8
evaluated data set
data set (3.1.6) that 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 processes. An evaluated data set is intended to be independent of reactor composition, geometries, energy group structures, and spectra.

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3.1.9
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.

3.1.10
few-group
energy-group (typically two-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 other conditions such as pressure, temperature, xenon concentration, void fraction.

3.1.11
fine group
energy group structure (typically hundreds of groups for LWR, thousands for LFMR) that is adopted for a particular application.

Note 1 to entry: The fine-group constants are dependent on temperature.

3.1.12
lattice
array of cells (or a fuel-assembly cell) with its associated immediate environment, such as the volume of moderator associated with it

3.1.13
lumped pseudo fission product
fictitious fission product that is used in depletion calculations to represent a combination of certain minor fission products that are not tracked in an explicit fission product model

Note 1 to entry: A lumped pseudo fission product is characterized by effective neutron cross sections and decay constants.

3.1.14
methods of calculation
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".

3.1.15
multigroup
more than a *few-groups* (3.1.10), can be hundreds or thousands of energy groups

3.1.16
probability table parameters
parameters for a probability table that is used to process cross sections

3.1.17
processed continuous-energy data set
data set (3.1.6) prepared by expansion or compaction of an *evaluated data set* (3.1.8) using specified algorithms

Note 1 to entry: Such a data set is intended to be independent of reactor composition, geometries and spectra.

3.1.18
quasi steady-state condition
evolution of reactivity and flux is sufficiently low between two time intervals such that we can consider that the core is under a succession of static states

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3.1.19
reaction rate
rate at which neutrons interact with nuclei at a point in the reactor, a volume of the phase space or as an integral over the reactor, according to the context

EXAMPLE The reaction rate for absorption, scattering and fission.

3.1.20
reactivity
the quantity (1 minus the eigenvalue, λ, of the steady-state neutron balance equation), written as:

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

where

Φ is the neutron flux;
F is the neutron fission yield operator;
M is the scattering, absorption, and leakage operator.

Note 1 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.

3.1.21
time-average model (CANDU)
time-average cell-homogenized *few-group* (3.1.10) macroscopic cross sections at each bundle location in the core are calculated as flux-weighted quantities over the fluence range experienced by the fuel during its residence at that specific location

Note 1 to entry: Consequently, the flux, fluence range and macroscopic cross sections at each location are linked together and depend on the individual refuelling rate of the channel in which the fuel bundle resides. The resulting self-consistency problem is solved iteratively to determine all the results of the time-average model.

Note 2 to entry: The CANada Deuterium Uranium (CANDU) reactor time-average model is also known as the equilibrium-core model.

3.1.22
ultrafine group
energy group structure (typically ten thousand energy groups) that can allow to explicitly describe the resolved resonances

Note 1 to entry: The ultrafine-group constants can be dependent on temperature.

3.1.23
experimental 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

3.1.24
numerical validation
process of determining the degree to which a model is accurate by comparison to a previously validated code system

3.1.25
verification
process of determining that a model implementation accurately represents the developer's conceptual description of the model and the solution to the model

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