International Standard

ISO 18075

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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Contents

Page

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.

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.

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.

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.

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

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

3.2 Abbreviations

BWR boiling water reactor

CANDU CANada Deuterium Uranium (reactor)

HTGR high temperature gas cooled reactor

LMR liquid metal reactor

LWR light water reactor

PHWR pressurized heavy water reactor

PWR pressurized water reactor

4 Relation to other standards

The following American National Standards are related to this document:

— ANSI/ANS-19.1-2019[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 that 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.

— ANSI/ANS-19.4-2017 (R2022)[3] and ANSI/ANS 19.5-1995 (W2005)[4].

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

— 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–19.3-2022.

— ANSI/ASME-NQA 1-2022[6] deals with quality assurance, including that for computer programs.

— ANSI/ANS-10.4–2008 (R2021)[9] deals with requirements for verifying and validating computer codes, such as those used for neutronics calculations.

— ANSI/ANS-10.5–2006 (R2021)[10] 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 are typically performed in a sequence of steps. A typical sequence can be

a) utilization of averaged-data set cross sections, nuclide number densities, and geometrical information (usually repeating cells or supercells) to calculate an application-dependent many-group neutron spectrum and within-cell spatial distribution for each different reactor region and composition,

b) utilization of the spectra and spatial distributions from step a) to collapse and homogenize averaged-data set cross sections to few-group form,

c) utilization of the few-group cross sections from step b) and geometrical information about the reactor to calculate reactivity and few-group flux spatial distributions in the reactor,

d) utilization of the preceding information to compute reaction rates in physical reactor components (including the reactor vessel if needed), and

e) utilization of the preceding information to calculate changes in nuclide composition of fuel and possibly other reactor components with exposure.

In some applications, the collapse to few-group form is not used, and the multigroup cross sections and geometrical information about the reactor are used to calculate reactivity and multigroup spatial distributions in the reactor.

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 that 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 the following:

a) presence of control elements (rods, cruciforms or other forms), and degradation (or depletion) 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 boiling water reactor (BWR);

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

i) spatial variations in temperatures (e.g. fuel temperature, coolant temperature, and moderator temperature);

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