SLOVENSKI STANDARD
oSIST prEN ISO 19226:2019

01-november-2019

Jedrska energija - Ugotavljanje pretoka nevtronov in premikov na atom (dpa) v reaktorski posodi in vgrajenih delov (ISO 19226:2017)

Nuclear energy - Determination of neutron fluence and displacement per atom (dpa) in reactor vessel and internals (ISO 19226:2017)

Kernenergie - Bestimmung der Neutronenfluenz und Verschiebungen pro Atom (dpa) im Reaktorbehälter und Einbauten (ISO 19226:2017)

Énergie nucléaire - Détermination de la fluence neutronique et des déplacements par atome (dpa) dans la cuve et les internes du réacteur (ISO 19226:2017)

Ta slovenski standard je istoveten z: **prEN ISO 19226**

ICS:

27.120.10	Reaktorska tehnika	Reactor engineering

oSIST prEN ISO 19226:2019 **en,fr,de**

INTERNATIONAL STANDARD

ISO 19226

First edition
2017-11

Nuclear energy — Determination of neutron fluence and displacement per atom (dpa) in reactor vessel and internals

Énergie nucléaire — Détermination de la fluence neutronique et du déplacement par atome (dpa) dans la cuve et les internes du réacteur

Reference number
ISO 19226:2017(E)

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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).

Attention is drawn to the possibility that some of the elements of this document may be the subject of patent rights. ISO shall not be held responsible for identifying any or all such patent rights. Details of any patent rights identified during the development of the document will be in the Introduction and/or on the ISO list of patent declarations received (see www.iso.org/patents).

Any trade name used in this document is information given for the convenience of users and does not constitute an endorsement.

For an explanation on the voluntary nature of standards, the meaning of ISO specific terms and expressions related to conformity assessment, as well as information about ISO's adherence to the World Trade Organization (WTO) principles in the Technical Barriers to Trade (TBT) see the following URL: www.iso.org/iso/foreword.html.

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 the ANSI/ANS 19.10-2009 but extends to cover the evaluation of irradiation damage due to neutron fluence.

Introduction

This document is intended for use by

a) those involved in the determination of exposure parameters for the prediction of irradiation damage to the vessel and to the internals of a nuclear reactor, where the exposure parameters can be neutron fluence and/or displacements per atom (dpa),

b) those involved in the determination of material properties of irradiated reactor vessel and reactor internals,

c) regulatory agencies in licensing actions such as the writing of Regulatory Guides, analysis of reports concerning the integrity and material properties of irradiated pressure vessels and reactor internals.

Nuclear energy — Determination of neutron fluence and displacement per atom (dpa) in reactor vessel and internals

1 Scope

This document provides a procedure for the evaluation of irradiation data in the region between the reactor core and the inside surface of the containment vessel, through the pressure vessel and the reactor cavity, between the ends of active fuel assemblies, given the neutron source in the core.

NOTE These irradiation data could be neutron fluence or displacements per atom (dpa), and Helium production.

The evaluation employs both neutron flux computations and measurement data from in-vessel and cavity dosimetry, as appropriate. This document applies to pressurized water reactors (PWRs), boiling water reactors (BWRs), and pressurized heavy water reactors (PHWRs).

This document also provides a procedure for evaluating neutron damage properties at the reactor pressure vessel and internal components of PWRs, BWRs, and PHWRs. Damage properties are focused on atomic displacement damage caused by direct displacements of atoms due to collisions with neutrons and indirect damage caused by gas production, both of which are strongly dependent on the neutron energy spectrum. Therefore, for a given neutron fluence and neutron energy spectrum, calculations of the total accumulated number of atomic displacements are important data to be used for reactor life management.

2 Normative references

The following documents are referred to in the text in such a way that some or all of their content constitutes requirements of this document. For dated references, only the edition cited applies. For undated references, the latest edition of the referenced document (including any amendments) applies.

ANSI/ANS 19.10, *Methods for determining neutron fluence in BWR and PWR pressure vessel and reactor internals*

ASTM E170-16a, *Standard Terminology Relating to Radiation Measurements and Dosimetry*

3 Terms and definitions

For the purposes of this document, the terms and definitions given in ANSI/ANS 19.10, ASTM E170-16a and the following apply.

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
accuracy of a measured/calculated value
difference between the “real” and the measured/calculated value, typically due to systematic errors in the measurement/calculation procedure

3.2
benchmark experiment
well-defined set of physical experiments with results judged to be sufficiently accurate for use as a calculational reference point

Note 1 to entry: The judgment is made by a group of experts in the subject area.

3.3
best-estimate fluence
most accurate value of the fluence based on all available measurements, calculated results, and adjustments based on bias estimates, least-squares analyses, and engineering judgment

3.4
calculational methodology
mathematical equations, approximations, assumptions, associated parameters, and calculational procedure 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 "calculational methodology."

3.5
code benchmark
comparison to the results of another code system that has been previously validated against experiment(s)

3.6
continuous-energy cross-section data
cross-section data that are specified in a dense point-wise manner that comprises the energy range

3.7
dosimeter reaction
neutron-induced nuclear reaction with a product nuclide having sufficient activity to be measured and related to the incident neutron fluence

3.8
displacements per atom (dpa)
mean number of times each atom of a solid is displaced from its lattice site during an exposure to displacing radiation, as calculated following standard procedures

3.9
least-squares adjustment procedure
method for combining the results of neutron transport calculations and the results of dosimetry measurements that provides an optimal estimate of the fluence by minimizing, in the least-squares sense, the calculation-to-measurement differences

3.10
multigroup cross-section data
cross-section data that have been determined by averaging the continuous-energy cross-section data over discrete energy intervals using specified weighting functions to preserve reaction rates

3.11
neutron fluence
time-integrated neutron fluence rate (i.e. the time-integrated neutron flux) as expressed in neutrons per square centimeter

3.12
precision of a measured/calculated value
standard deviation (if available from a set of repeated measurements/calculations) of the distribution of the measured or calculated physical value

3.13
reactor internals
reactor structure components that are within the pressure vessel such as the core baffle, core barrel, thermal shield, lower and upper core plates in PWRs and BWRs

3.14
solution variance
measure of the statistical variance of the Monte Carlo transport solution due to a finite number of particle histories

Note 1 to entry: Mathematically, it is the second central moment of the distribution about the mean value, which is used to measure the dispersion of the distribution about the mean.

4 Transport theory calculational models

4.1 General

4.1.1 Output requirements

The transport calculations need to be able to determine accurately the neutron flux or fluence distributions, and/or other response parameters such as reaction rates or dpa for the analysis of integral dosimetry measurements and for the prediction of irradiation damage to reactor pressure vessels and its internals.

Calculation methodologies described in this document focus on neutron fluence for determining radiation embrittlement of reactor vessel materials.

While neutron fluence (E > 1,0 MeV) (where neutron fluence (E > 1,0 MeV) represents the fluence of neutrons with energy above 1,0 MeV) has frequently been selected as the exposure parameter for determining radiation embrittlement of reactor vessel materials, the procedures in this document extend to include fluence spectrum above 0,1 MeV, in addition to thermal fluence below 0,625 eV.

Some parameters of the calculations would be determined based on

— direct use of the results: design or comparison to measurements (which imply envelope or best-estimate results, respectively),

— required response functions: (E > 1,0 MeV) neutron flux, (E > 0,1 MeV) neutron flux, thermal neutron flux (E < 0,625 eV), dpa/s, dosimeter reaction rates;

NOTE The figures for flux, given as examples of upper or lower limit, depend on the application.

— location(s) of interest: fineness of the spatial meshing.

4.1.2 Methodology: transport calculations with fixed sources

In the practice suggested in this document, a source distribution throughout the core is prepared using the results of core physics calculations; multidimensional transport theory calculations then are performed to propagate the neutrons to regions outside the core.

This document uses codes based on transport theory to determine multigroup three-dimensional flux distributions and to evaluate the reaction rates of dosimetry materials or dpa properties through proper use of response functions or cross sections.

Transport theory calculations should be performed using deterministic discrete ordinates (S_N)[2] or statistical Monte Carlo[3] approaches as discussed in 4.2.2 and 4.2.3, respectively. Other transport methods may be used if they are part of a benchmarked methodology.

