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Nuclear criticality safety — Evaluation of systems containing PWR UOX fuels — Bounding burnup credit approach

Sûreté-criticité — Évaluation des systèmes mettant en œuvre des combustibles REP UOX — Approche conservative de crédit burnup

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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.

International Standards are drafted in accordance with the rules given in the ISO/IEC Directives, Part 2.

The main task of technical committees is to prepare International Standards. Draft International Standards adopted by the technical committees are circulated to the member bodies for voting. Publication as an International Standard requires approval by at least 75 % of the member bodies casting a vote.

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.

ISO 27468 was prepared by Technical Committee ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 5, *Nuclear fuel cycle*.

Introduction

For many years, criticality evaluations involving irradiated uranium oxide (UOX) fuels in pressurized water reactor (PWR) considered the fuel as un-irradiated. Information on and consideration of the fuel properties after irradiation could usually have resulted in considerable criticality safety margins.

The use of PWR UOX fuel with increased enrichment of ^{235}U motivates evaluation of burnup credit in existing and new applications for storage, reprocessing or transport of irradiated fuel. A more realistic estimation of the actual effective neutron multiplication factor, k_{eff}, of a system involving irradiated fuel is possible with methods available to nuclear criticality safety specialists. Thus, the maximum estimated k_{eff} value during normal conditions and incidents can be reduced compared with the assumption of an un-irradiated fuel.

Moreover, the safe use of burnup credit can reduce the overall risk (fewer cask moves, etc.).

Therefore, for the safe use of the burnup credit, this International Standard highlights the need to consider new parameters in addition to those that need evaluation for un-irradiated fuel. It presents the different issues that should be addressed to support evaluations of burnup credit for systems with PWR fuels that are initially containing uranium oxides and then irradiated in a PWR.

This International Standard identifies a bounding approach in terms of k_{eff} calculation. Other approaches may be used (e.g. calculation of the average configuration with k_{eff} criteria covering credible variations/bias/uncertainties) especially if there are additional mechanisms to control the subcriticality (e.g. use of boron, gadolinium or dry transport).

Overall criticality safety evaluation and eventual implementation of burnup credit are not covered by this International Standard. However, the burnup credit evaluation in this International Standard should support use of burnup credit in the overall criticality safety evaluation and an eventual implementation of burnup credit.

Nuclear criticality safety — Evaluation of systems containing PWR UOX fuels — Bounding burnup credit approach

1 Scope

This International Standard establishes an evaluation methodology for nuclear criticality safety with burnup credit. It identifies important parameters and specifies requirements, recommendations, and precautions to be taken into account in the evaluations. It also highlights the main important technical fields to ensure that the fuel composition or history considered in calculations provides a bounding value of the effective neutron multiplication factor, k_{eff}.

This International Standard is applicable to transport, storage, disposal or reprocessing units implying irradiated fissile material from pressurized water reactor (PWR) fuels that initially contain uranium oxide (UOX).

Fuels irradiated in other reactors (e.g. boiling water reactors) and fuels that initially contain mixed uranium-plutonium oxide are not covered in this International Standard.

This International Standard does not specify requirements related to overall criticality safety evaluation or eventual implementation of burnup credit.

2 Normative references

The following referenced documents are indispensable for the application 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.

ISO 1709, *Nuclear energy — Fissile materials — Principles of criticality safety in storing, handling and processing*

ISO 14943, *Nuclear fuel technology — Administrative criteria related to nuclear criticality safety*

3 Terms and definitions

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

3.1
actinide
element with atomic number in the range from 90 to 103

NOTE Many actinides are produced during the irradiation due to neutron capture on other actinides and/or decay of other actinides and/or by (n,2n) reactions, etc. The corresponding nuclides are all neutron producers and some are net (considering neutron production and absorption) neutron producers in a slow neutron energy spectrum.

3.2
axial burnup profile
real or modelled axial distribution of the burnup in the fuel assembly

NOTE The axial distribution of the burnup is caused by axial neutron leakage, axial variations in the fuel enrichment, moderator temperature rise through the core, non-full length burnable poison and partial insertion of control rods.

3.3
burnable poison
nuclide neutron absorber added to the fuel assembly to control reactor reactivity and power distribution

NOTE 1 As the reactor operation progresses, the amount of neutron absorbing material is depleted, or ''burned''. Then, if the presence of burnable poisons (fixed or removable) is considered in a criticality safety evaluation, the most reactive condition may not be for the fresh fuel.

NOTE 2 See also ISO 921:1997, entry 135.

3.4
burnup
average energy released by a defined region of the fuel during its irradiation

NOTE 1 This region could be a complete fuel assembly or some part of the assembly. Burnup is commonly expressed as energy released per mass of Initial fissionable actinides (uranium only for this International Standard). Units commonly used are expressed in megawatt day per metric tonne of initial uranium (MWd/t) or gigawatt day per metric tonne of initial uranium (GWd/t).

NOTE 2 See also ISO 921:1997, entry 1156.

3.5
burnup credit
margin of reduced k_{eff} for an evaluated system, due to the irradiation of fuel in a reactor, as determined with the use of a structured evaluation process

3.6
cooling time
time following the final irradiation of the fuel in a reactor

NOTE During this period, the radioactive decay results in changes in the fuel nuclide concentrations.

3.7
depletion calculation
calculation performed to determine the concentrations of individual nuclides in the fuel at the end of irradiation in a reactor; that is a cooling time equal to zero

NOTE 1 Other fuel properties can usually be determined by depletion calculations (e.g. flux-weighted macroscopic cross-sections or lattice cell k_∞).

NOTE 2 Radioactive decay between reactor irradiation periods and after final shutdown is usually included in the same calculation procedure.

3.8
end effect
impact on k_{eff} of the less irradiated parts of the fuel assembly (upper and lower ends of the assembly)

NOTE The end effect is commonly defined as the difference between the k_{eff} for the two following systems:

— a system containing irradiated fuel assemblies having a constant fuel composition corresponding to the average burnup and irradiation energy spectrum of the fuel,

— the same system containing irradiated fuel assemblies having an axially varying fuel composition corresponding to the modelled axial burnup profile, with consideration of the neutron energy spectrum during irradiation.

3.9
fission product
nuclide produced from nuclear fission

NOTE 1 During this reaction two or more fission products are produced together with neutrons and radiations (gamma, etc.). The fission products can be a direct result of the fissions or can be created after the decay of (or neutron absorption with) other fission products. Often only a selection of fission products is accounted for as neutron absorbers in burnup credit, but consideration of all fission products absorption is required to simulate fuel irradiation during reactor operation.

NOTE 2 See also ISO 921:1997, entry 478.

3.10
loosely coupled system
system in which two or more regions with high “local” values of k_{eff} are separated by regions with low k_{eff} importance

NOTE Convergence problems can occur when a Monte Carlo method is used for the k_{eff} calculation of such systems where neutron interaction between the highly fissile regions is weak.

3.11
validation
documented determination that the combination of models, methods and data as embodied in a computer code methodology is an appropriate representation of the process or system for which it is intended

NOTE This documented determination is accomplished by comparing code results to benchmark experimental results to define code bias and areas of applicability of a calculation method.

4 Methodology for criticality safety evaluations considering burnup of the fuel

IMPORTANT — The application of this clause requires evaluators to know the initial composition of each fuel and its history of irradiation.

4.1 General

The bounding approach identified in this International Standard consists of the main following steps, for a given application (e.g. a given transport, storage, reprocessing, disposal) and for a given range of irradiated fuels:

- to choose and justify a burnup distribution to model in the fuel assemblies (see 4.2);
- to calculate the irradiated fuel nuclide concentrations for each burnup assessed, with considerations for the cooling time (see 4.3);
- to select the nuclides to be included in the evaluation of k_{eff} for the application (see 4.4);
- to perform the criticality calculations of the evaluated application (see 4.5).

For each step where a calculation code is used, the validation of these calculation tools shall be justified and documented. Such validation may consist of a global validation of the resulting k_{eff}.

4.2 Distribution of burnup

4.2.1 The burnup distribution of the irradiated fuel assembly shall be evaluated because of its impact on k_{eff} (see References [1], [2], [9], [15] and [16]). The axial and radial/horizontal burnup gradients, due to the neutron flux distribution during the irradiation, are mainly related to:

- neutron leakage at the top and the bottom of the fuel assembly;
- neutron absorption within partially inserted control rods at the top of the fuel assembly;

