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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

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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;

— the moderator density change from the bottom to the top of the core;

— radial leakage of the neutrons, which depend on the environment of the assembly, on its position in the reactor during irradiation and on the presence of burnable poisons;

— radial absorption of the neutrons.

WARNING — The axial burnup distribution is not sufficient to determine the axial variation of the composition of the irradiated fuel: the neutron spectrum of the irradiation flux also varies axially and has an impact on the fuel nuclide concentrations that are determined from the depletion calculation. Guidelines on the effect on fuel nuclides concentration of the fuel depletion parameters are given in 4.3.

4.2.2 Each fuel assembly may be divided into regions or zones in which the burnup is assumed to be uniform. The division into such regions or zones shall be justified for each application and may be different to what is usually used in the reactor core calculations.

4.2.3 The axial burnup profile(s) considered in the criticality safety evaluation shall ensure a conservative approach with regard to:

— the range of fuel assemblies (each of them with a different axial burnup profile) considered in the evaluation;

— the partial insertion of control rods within the fuel assembly during its irradiation.

4.2.3.1 The axial burnup profile considered in the criticality safety evaluation may come from determining the most limiting profile among calculated profiles and/or measured profiles. When the axial burnup profile is obtained from calculations, the evaluator shall account for uncertainties from code validation. When the axial burnup profile is obtained from measurements, the evaluator shall account for uncertainties due to the measurement devices and from the validation of the measurement method.

NOTE Measurement methods of burnup require calculation steps to convert the raw measure into a burnup value.

4.2.3.2 If a uniform axial distribution of burnup (commonly called “flat profile”) is used, then the burnup value selected shall ensure a bounding approach with regard to the end effect.

NOTE Considerations about determining an axial profile are given in References [16] and [17].

4.2.4 The significance of the effect on k_{eff} of a radial/horizontal shape of burnup should be evaluated. This shape can lead to having at least one fuel assembly side less irradiated than the mean burnup. For any application where the proximity of lower irradiated faces of the fuel assemblies being transported, stored, disposed or processed may lead to an increase in the k_{eff}, the effect on reactivity of the radial/horizontal shape of burnup shall be considered.

NOTE The radial/horizontal gradients of burnup are calculated as a function of burnup in Reference [3]. An example of the potential influence of horizontal gradients in PWR fuel is provided in Reference [7] (based on Reference [3]).

4.3 Nuclide concentration calculation

4.3.1 The calculation of the irradiated fuel nuclide concentrations shall consider:

— the fresh fuel characteristics;

— the fuel irradiation parameters and cooling time which lead to bounding nuclide concentrations in terms of k_{eff};

— the validation of the depletion code used (e.g. against post-irradiation examination of fuel compositions; see Annex A).

4.3.2 The range of possible variations of the irradiation parameters shall be known in order to define the bounding values for the depletion calculations. The values of each irradiation parameter considered in the depletion calculation shall be justified and documented. The main irradiation parameters are listed in 4.3.2.1.

4.3.2.1 The irradiation parameters leading to a neutron flux spectrum hardening shall be considered. These parameters are (see References [1], [9], [12], [13] and [15]):

— boron concentration in the reactor coolant;

— temperature and density of the reactor coolant;

— presence of burnable poisons;

— control rods insertion;

— presence of mixed oxide (MOX) fuels and/or poisoned fuels around the fuel assembly of interest.

4.3.2.2 The other irradiation parameters required for the depletion calculation (e.g. specific power, fuel temperature, shutdown periods) shall be assessed.

4.3.3 For a given set of irradiated fuels, a bounding value of cooling time shall be assumed for the depletion calculation (i.e. the cooling time considered in criticality assessments shall ensure that the k_{eff} value of the application never exceeds the calculated value).

WARNING — The minimal value of the cooling time that can be justified by the operators among all the fuel assemblies may not necessarily lead to the maximum value of the k_{eff}. Up to a cooling time of about 100 years, the k_{eff} decreases mainly due to the decay of the ^{241}Pu (into ^{241}Am) plus the increase of ^{155}Gd (from ^{155}Eu). For longer cooling time, the k_{eff} starts to increase again (as the ^{241}Am and ^{240}Pu decay) up to 30 000 years (see Reference [14]).

EXAMPLE 1 For applications involving fuels up to 100 years after their irradiation in a reactor, any value of cooling time may be assumed.

EXAMPLE 2 For applications involving fuels up to 200 years after their irradiation in a reactor, the maximal value of cooling time which may be assumed is 40 years (see Reference [14]).

4.3.4 Due to the complexity of the depletion calculations, the different options used for the validation of computer codes (e.g. definition of time intervals for recalculation of the cross-sections during the depletion calculations, self-shielding) shall be evaluated and documented.

4.4 Nuclides selection

4.4.1 The nuclides (actinides and fission products) included in the evaluation of k_{eff} shall be determined for each criticality safety evaluation considering burnup of the fuel.

4.4.2 The list of nuclides under consideration shall take into account:

— the characteristics of each nuclide (e.g. fission and absorption cross-sections, concentration) and its impact on k_{eff};

— the assurance of the continued existence of each nuclide in the evaluated application (with regard to, for example: decay, release of entrained gases, chemical separation);

— the accuracy with which their concentration is predicted by the depletion code used.

4.4.2.1 All nuclides with a significant positive contribution to k_{eff} should be accounted for. Omission of such nuclides shall be justified.

NOTE A positive reactivity contribution of one nuclide can be balanced by a larger negative reactivity contribution by another nuclide and both can thus be omitted, especially if their presence is correlated.

4.4.2.2 Only nuclides with a constant concentration (relative to the timeframe of interest of the specific application) should be accounted for as neutron absorbers in the burnup credit application. A fission product that decays in the timeframe of interest may be accounted for only if the decay product (i.e. the daughter nuclide) complies with the requirements of 4.4 and has a higher negative worth in k_{eff}.

4.4.2.3 Each of the accounted nuclides shall be justified to be present in the fuel with regard to the normal and fault conditions of the studied application (e.g. deviation in process conditions and accident conditions of transport). An application may necessarily limit the number of nuclides that can be safely accounted for and this number may vary depending on the application (e.g. storage, transport, reprocessing and disposal).

NOTE This has additional importance for applications of fuel reprocessing (e.g. dissolution).

4.5 Criticality safety calculations

4.5.1 For storage, handling, and processing fissile materials within the boundaries of nuclear establishments, the basic principles and limitations specified in ISO 1709 in relation to criticality safety calculations shall be applied.

4.5.2 In addition to ISO 1709 requirements, the items below shall be more particularly considered in criticality safety calculations for burnup credit applications. In particular, the criticality safety evaluation shall take into account the effect on k_{eff} of these items:

— geometrical characteristics of irradiated fuel (e.g. expansion of pellets, distortion of the rod pitch);

— validation of the uncertainties in nuclide cross-sections for the nuclear data used, especially for the nuclides accounted for in the irradiated fuel composition and that are not commonly used in criticality calculation with a fresh fuel;

  EXAMPLE The discrepancies between calculated and measured values of neutron-absorption rates may lead to the taking into account of penalizing factors in the criticality calculations performed (see References [4], [5], [6]).

— loosely coupled systems when a Monte Carlo method is used to perform the k_{eff} calculation; the Monte Carlo code report shall be checked to confirm that there has been sufficient sampling in the high reactivity zones (end zones of the fuel assemblies);

— interaction or mixture with other fissile materials, especially fuel assemblies with different properties (un-irradiated fuel, different geometry, etc.).

  EXAMPLE In a regularly spaced lattice of fuel assemblies, such as a storage rack or a spent fuel cask basket, a mixture of different types of assemblies can result in a significantly higher value of k_{eff} than if all assemblies are identical (see Reference [7]).

5 Implementation of criticality safety evaluations considering burnup of the fuel

5.1 For storage, handling, and processing fissile materials within the boundaries of nuclear establishments, the basic principles and limitations which govern such operations specified in ISO 1709 and ISO 14943 shall be applied.

5.2 Concerning operations and storage involving only irradiated fuel, the operational (on-site) implementation of burnup credit requires specific evaluations and/or verifications.

These evaluations are required due to a more realistic estimation of the system k_{eff} compared to the fresh fuel assumption and due to the multiple parameters possibly affecting the reactivity of a fuel assembly for a given value of burnup.

NOTE Annex B and Reference [7] give guidelines for implementation of a burnup credit method.

Annex A
(informative)

Validation of the depletion codes against post-irradiation examination data

The different stages of the experimental validation of the depletion codes with samples of irradiated fuels (or post-irradiation examination, PIE) may be as follows (e.g. see Reference [8]).

a) To define the measured values of burnup (and their associated uncertainties), derived from the concentration of a given nuclide indicator of burnup. This requires knowing the relation between the burnup and the concentration of this nuclide. Any dependencies of this function with the irradiation parameters should be minimized and the uncertainties due to remaining dependences should be accounted for.

   NOTE An indicator of burnup is a nuclide (or a ratio of nuclides) for which a reliable relationship between the burnup and its concentration can be established.

b) To calculate, with the depletion code, the concentrations of the nuclides of interest for the measured value of burnup and to compare them to their measured values, in order to determine the calculation versus experiments (C/E) values.

c) To propose a method to set penalizing factors on the calculated nuclides concentration. Those penalizing factors should take into account the C/E values, the number of experimental values, and the level of confidence in the experimental data. Note that those factors only correspond to a given domain of burnup and of irradiation histories.

   WARNING — If the penalizing factors resulting from this method are large, the criticality safety assessor should reconsider the depletion calculation method used and/or the choice of the nuclides leading to such factors.

Annex B
(informative)

Operational implementation of a burnup credit application

The controls required for implementation of a burnup credit application in operations or storages may be as follows.

a) Regarding the conservative evaluation of the nuclide composition of the irradiated fuel, the real ranges of variation of the parameters related to the irradiation history should be checked to comply with the domain of applicability of the criticality safety evaluation.

b) Regarding the conservative evaluation of the distribution of the burnup, it should be checked that the real burnup shape is bounded by the burnup shape assumed in the criticality safety evaluation.

   EXAMPLE 1 By a measurement (detection and the associated calculations) of the burnup shape done for each assembly (or every N assemblies when justified by a probability assessment).

   EXAMPLE 2 By the in-core measurements and/or calculations of the neutron flux in a reactor, to define a bounding burnup shape for every assembly of a given reactor.

c) An independent verification of the compliance between the irradiated fuel and the assessed domain of the criticality evaluation (e.g. estimated burnup versus minimum required burnup).

The quantity and quality of the associated means of control, defined by the safety evaluation of the operations, depends on the importance of burnup credit (reactivity worth) in ensuring the sub-criticality of these operations.

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