

BS ISO 27468:2011



BSI Standards Publication

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

**bsi.**

...making excellence a habit.™

**National foreword**

This British Standard is the UK implementation of ISO 27468:2011.

The UK participation in its preparation was entrusted to Technical Committee NCE/9, Nuclear fuel cycle technology.

A list of organizations represented on this committee can be obtained on request to its secretary.

This publication does not purport to include all the necessary provisions of a contract. Users are responsible for its correct application.

© BSI 2011

ISBN 978 0 580 65210 3

ICS 27.120.30

**Compliance with a British Standard cannot confer immunity from legal obligations.**

This British Standard was published under the authority of the Standards Policy and Strategy Committee on 31 July 2011.

**Amendments issued since publication**

Date	Text affected
------	---------------

---

---

---

**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 conservatrice de crédit burnup*





**COPYRIGHT PROTECTED DOCUMENT**

© ISO 2011

All rights reserved. Unless otherwise specified, no part of this publication may be reproduced or utilized in any form or by any means, electronic or mechanical, including photocopying and microfilm, without permission in writing from either ISO at the address below or ISO's member body in the country of the requester.

ISO copyright office  
Case postale 56 • CH-1211 Geneva 20  
Tel. + 41 22 749 01 11  
Fax + 41 22 749 09 47  
E-mail [copyright@iso.org](mailto:copyright@iso.org)  
Web [www.iso.org](http://www.iso.org)

Published in Switzerland

## Contents

Page

<b>Foreword</b> .....	<b>iv</b>
<b>Introduction</b> .....	<b>v</b>
<b>1 Scope</b> .....	<b>1</b>
<b>2 Normative references</b> .....	<b>1</b>
<b>3 Terms and definitions</b> .....	<b>1</b>
<b>4 Methodology for criticality safety evaluations considering burnup of the fuel</b> .....	<b>3</b>
<b>5 Implementation of criticality safety evaluations considering burnup of the fuel</b> .....	<b>6</b>
<b>Annex A (informative) Validation of the depletion codes against post-irradiation examination data</b> .....	<b>7</b>
<b>Annex B (informative) Operational implementation of a burnup credit application</b> .....	<b>8</b>
<b>Bibliography</b> .....	<b>9</b>

## 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  $^{235}\text{U}$  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,  $k_{\text{eff}}$ , of a system involving irradiated fuel is possible with methods available to nuclear criticality safety specialists. Thus, the maximum estimated  $k_{\text{eff}}$  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  $k_{\text{eff}}$  calculation. Other approaches may be used (e.g. calculation of the average configuration with  $k_{\text{eff}}$  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,  $k_{\text{eff}}$ .

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  $k_{\text{eff}}$  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  $k_{\infty}$ ).

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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  are separated by regions with low  $k_{\text{eff}}$  importance

NOTE Convergence problems can occur when a Monte Carlo method is used for the  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$ .

### 4.2 Distribution of burnup

**4.2.1** The burnup distribution of the irradiated fuel assembly shall be evaluated because of its impact on  $k_{\text{eff}}$  (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  $k_{\text{eff}}$  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  $k_{\text{eff}}$ , 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  $k_{\text{eff}}$ ;
- 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  $k_{\text{eff}}$  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  $k_{\text{eff}}$ . Up to a cooling time of about 100 years, the  $k_{\text{eff}}$  decreases mainly due to the decay of the  $^{241}\text{Pu}$  (into  $^{241}\text{Am}$ ) plus the increase of  $^{155}\text{Gd}$  (from  $^{155}\text{Eu}$ ). For longer cooling time, the  $k_{\text{eff}}$  starts to increase again (as the  $^{241}\text{Am}$  and  $^{240}\text{Pu}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$ ;
- 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  $k_{\text{eff}}$  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  $k_{\text{eff}}$ .

**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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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  $k_{\text{eff}}$  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.



## Bibliography

- [1] RABY, J., *et al.* Current studies related to the use of burnup credit in France. In: *Proceedings of Int. Conf. on Nuclear Criticality Safety, ICNC'2003*, Tokai Mura, Japan, October 2003
- [2] POULLOT, G., NOURI, A. "EXERCICE T" - PHASE II.B - Taux de combustion — Configuration emballage de transport; Effet de la distribution axiale sur la réactivité, note SEC/T/95.382, December 1995
- [3] *Topical report on actinide-only burnup credit for PWR spent nuclear fuel packages* — U.S. Department of Energy — DOE/RW — 0472 Rev.2, September 1998
- [4] ANNO, J., *et al.* French fission products burnup experiments performed in Cadarache and Valduc. results comparison., In: *ICNC'2003*, Tokai Mura, Japan, Oct. 20-24, 2003
- [5] ANNO, J., *et al.* *Status of the Joint French IPSN/COGEMA Validation Program of Fission Products*, ANS November 2001
- [6] THIOLLAY, N., *et al.* Burn-up credit for fission product nuclides in PWR (UO<sub>2</sub>) spent fuels, In: *ICNC'99*, Versailles, France, Sept. 20-24, 1999, pp. 612-621 (1999)
- [7] MENNERDAHL, D. BUC — A simple nuclear criticality safety concept that can be very difficult to implement. In: *Proceedings of IAEA Technical Committee meeting*, Madrid, 22-26 April 2002
- [8] RIFFARD, C., *et al.* International program REBUS experimental validation of spent fuel — Isotopic predictions for a 3.8 % UO<sub>2</sub> PWR with the DARWIN code system. In: *Proceedings of Int. Conf. on Nuclear Criticality Safety, ICNC'2007*, Saint Petersburg, June 2007
- [9] JUTIER, L., *et al.* "Latest studies related to the use of burnup credit in France". In: *Proceedings of International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing and Disposition (CSN/IAEA)*, Cordoba, Spain, October 2009
- [10] SANDERS, C.E., WAGNER, J.C. *Parametric study of the effect of control rods for PWR burnup credit*, NUREG/CR-6759, US Nuclear Regulatory Commission, Washington, DC., February 2002
- [11] SANDERS, C.E., WAGNER, J.C. *Study of the effect of integral burnable absorbers for PWR burnup credit*, NUREG/CR-6760, US Nuclear Regulatory Commission, Washington, DC 20555-0001, August 2001
- [12] WAGNER, J.C. PARKS, C.V. *Parametric study of the effect of burnable poison rods for PWR burnup credit*, NUREG/CR-6761, US Nuclear Regulatory Commission, Washington, DC 20555-0001, March 2002
- [13] DEHART, M.D. *Sensitivity and parametric evaluations of significant aspects of burnup credit for PWR spent fuel packages*, ORNL/TM-12973, Oak Ridge National Laboratory, Oak Ridge, TN, May 1996
- [14] WAGNER, J.C., PARKS, C.V. *Recommendation on the credit for cooling time in PWR burnup credit analyses*, NUREG/CR-6781, US Nuclear Regulatory Commission, Washington, DC, May 2002
- [15] PARKS, C.V., DEHART, M.D., WAGNER, J.C. *Review and prioritization of technical issues related to burnup credit for LWR fuel*, NUREG/CR-6665, US Nuclear Regulatory Commission, Washington, DC, February 2000
- [16] WAGNER, J.C., DEHART, M.D., PARKS, C.V. *Recommendations for Addressing axial burnup in PWR burnup credit analyses*, NUREG/CR-6801, US Nuclear Regulatory Commission, Washington, DC 20555-0001, March 2003

- [17] CABROL, E. *et al*, Determining an axial burn-up profile for BUC criticality studies by using French database of axial burn-up measurements. In: *Proceedings of Int. Conf. on Nuclear Criticality Safety, ICNC'2007*, Saint Petersburg, June 2007
- [18] ISO 921:1997, *Nuclear energy — Vocabulary*







# British Standards Institution (BSI)

BSI is the national body responsible for preparing British Standards and other standards-related publications, information and services.

BSI is incorporated by Royal Charter. British Standards and other standardization products are published by BSI Standards Limited.

## About us

We bring together business, industry, government, consumers, innovators and others to shape their combined experience and expertise into standards-based solutions.

The knowledge embodied in our standards has been carefully assembled in a dependable format and refined through our open consultation process. Organizations of all sizes and across all sectors choose standards to help them achieve their goals.

## Information on standards

We can provide you with the knowledge that your organization needs to succeed. Find out more about British Standards by visiting our website at [bsigroup.com/standards](http://bsigroup.com/standards) or contacting our Customer Services team or Knowledge Centre.

## Buying standards

You can buy and download PDF versions of BSI publications, including British and adopted European and international standards, through our website at [bsigroup.com/shop](http://bsigroup.com/shop), where hard copies can also be purchased.

If you need international and foreign standards from other Standards Development Organizations, hard copies can be ordered from our Customer Services team.

## Subscriptions

Our range of subscription services are designed to make using standards easier for you. For further information on our subscription products go to [bsigroup.com/subscriptions](http://bsigroup.com/subscriptions).

With **British Standards Online (BSOL)** you'll have instant access to over 55,000 British and adopted European and international standards from your desktop. It's available 24/7 and is refreshed daily so you'll always be up to date.

You can keep in touch with standards developments and receive substantial discounts on the purchase price of standards, both in single copy and subscription format, by becoming a **BSI Subscribing Member**.

**PLUS** is an updating service exclusive to BSI Subscribing Members. You will automatically receive the latest hard copy of your standards when they're revised or replaced.

To find out more about becoming a BSI Subscribing Member and the benefits of membership, please visit [bsigroup.com/shop](http://bsigroup.com/shop).

With a **Multi-User Network Licence (MUNL)** you are able to host standards publications on your intranet. Licences can cover as few or as many users as you wish. With updates supplied as soon as they're available, you can be sure your documentation is current. For further information, email [bsmusales@bsigroup.com](mailto:bsmusales@bsigroup.com).

## BSI Group Headquarters

389 Chiswick High Road London W4 4AL UK

## Revisions

Our British Standards and other publications are updated by amendment or revision.

We continually improve the quality of our products and services to benefit your business. If you find an inaccuracy or ambiguity within a British Standard or other BSI publication please inform the Knowledge Centre.

## Copyright

All the data, software and documentation set out in all British Standards and other BSI publications are the property of and copyrighted by BSI, or some person or entity that owns copyright in the information used (such as the international standardization bodies) and has formally licensed such information to BSI for commercial publication and use. Except as permitted under the Copyright, Designs and Patents Act 1988 no extract may be reproduced, stored in a retrieval system or transmitted in any form or by any means – electronic, photocopying, recording or otherwise – without prior written permission from BSI. Details and advice can be obtained from the Copyright & Licensing Department.

## Useful Contacts:

### Customer Services

**Tel:** +44 845 086 9001

**Email (orders):** [orders@bsigroup.com](mailto:orders@bsigroup.com)

**Email (enquiries):** [cservices@bsigroup.com](mailto:cservices@bsigroup.com)

### Subscriptions

**Tel:** +44 845 086 9001

**Email:** [subscriptions@bsigroup.com](mailto:subscriptions@bsigroup.com)

### Knowledge Centre

**Tel:** +44 20 8996 7004

**Email:** [knowledgecentre@bsigroup.com](mailto:knowledgecentre@bsigroup.com)

### Copyright & Licensing

**Tel:** +44 20 8996 7070

**Email:** [copyright@bsigroup.com](mailto:copyright@bsigroup.com)



...making excellence a habit.™