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# Nuclear criticality safety — Critical values for homogeneous plutonium-uranium oxide fuel mixtures outside of reactors

*Sûreté-criticité — Valeurs critiques pour oxydes mixtes homogènes de plutonium et d'uranium hors réacteurs*



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# Foreword

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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.

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ISO 11311 was prepared by Technical Committee ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 5, *Nuclear fuel cycle*.

# Introduction

This International Standard provides specifications to establish process and equipment limits for controlling the nuclear criticality hazard (e.g. choice of process monitoring modes, choice of equipment geometry) in facilities (outside of nuclear reactors) involving mixed uranium-plutonium oxide (MOX) fuel.

The criticality risk for this type of fuel results from the presence of the fissile nuclides $^{239}$Pu, $^{241}$Pu and $^{235}$U, and from other fissionable nuclides, such as $^{242}$Pu, $^{240}$Pu and $^{238}$U, more or less neutron absorbing.

The systems considered are uniform and homogeneous mixtures, moderated and reflected by water. The geometries concerned are single units of spheres, cylinders and slabs. A limited number of important safety parameter values are then selected.

Actually, regarding the field of MOX fuel, there are insufficient directly representative experiments of damp powders for establishing the bias between calculations and measurements. Therefore, an inter-code comparison is done to conservatively estimate critical values for different fissile material specifications.

Because the use of calculation codes can be associated with different nuclear libraries, the preceding comparison is extended to the results obtained with the most common nuclear data libraries.

Consequently, this International Standard provides reference critical values for the safety parameters selected. These values are determined by inter-code comparisons with an acceptable accuracy and are defined as the lowest calculated critical values of the selected safety parameters. These values will help nuclear criticality safety assessors during their analysis to make technical prescriptions for criticality risk prevention and for production purposes.

# Nuclear criticality safety — Critical values for homogeneous plutonium-uranium oxide fuel mixtures outside of reactors

## 1 Scope

This International Standard specifies common reference critical values (of which the effective neutron multiplication factor, $k_{eff}$ is equal to 1) for homogeneous water-moderated plutonium-uranium oxide mixtures based on an inter-code comparison of calculated critical values.

It is applicable to operations with unirradiated mixed uranium-plutonium oxide (MOX) outside nuclear reactors.

A classical validation approach for these systems is difficult because of the paucity of critical experiments for MOX fuel.

Various reference systems, in terms of isotopic compositions, thicknesses of water reflection, and densities of oxide are evaluated by different combinations of calculation codes and nuclear data libraries (i.e. different calculation schemes, see Annex B).



The critical values defined in this International Standard are the lowest of those calculated by each of these calculation schemes and accepted as credible.

The values in this International Standard are reference values and not absolute critical values.

## 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 921, *Nuclear energy — Vocabulary*

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

## 3 Terms and definitions

For the purposes of this document, the terms and definitions given in ISO 921 apply.

## 4 Reference systems concerned by this International Standard

### 4.1 Reference fissile media

#### 4.1.1 Description

The reference fissile media are homogeneous and uniform mixtures of uranium and plutonium dioxides in water.

 

#### 4.1.2 Plutonium content

The plutonium content in the mixture, $w_{\mathrm{Pu}}$, expressed as a percentage mass fraction, is defined by Equation (1):

$$w_{\mathrm{Pu}} = \frac{m_{\mathrm{Pu}}}{m_{\mathrm{U}} + m_{\mathrm{Pu}}} \tag{1}$$

where

$m_{\mathrm{Pu}}$ is the mass, in grams, of plutonium in the mixture;

$m_{\mathrm{U}}$ is the mass, in grams, of uranium in the mixture.

Plutonium contents used in the reference fissile media are:

a) $w_{\mathrm{Pu}} = 35{,}0\ \%$;

b) $w_{\mathrm{Pu}} = 12{,}5\ \%$.

#### 4.1.3 Oxide density ranges

Two ranges of oxide ($UO_2 + PuO_2$) density, expressed in grams per cubic centimetre, are considered:

— up to 3,50 g/cm$^3$, if the plutonium content is 35,0 % mass fraction;

— up to 11,03 g/cm$^3$, if the plutonium content is 12,5 % mass fraction.

NOTE The latter density is the theoretical dry density for this specific isotopic MOX composition.



#### 4.1.4 Isotopic composition

##### 4.1.4.1 Notation

In this subclause, the following notation is used:

$m_{i_{\mathrm{U}}}$ is the mass, in grams, of isotope $i$ of uranium;

$m_{\mathrm{U,total}}$ is the mass, in grams, of uranium;

$m_{i_{\mathrm{Pu}}}$ is the mass, in grams, of isotope $i$ of plutonium;

$m_{\mathrm{Pu,total}}$ is the mass, in grams, of plutonium.

##### 4.1.4.2 Uranium

The uranium composition considered corresponds to natural uranium:

$$m_{^{235}\mathrm{U}} / m_{\mathrm{U,total}} = 0{,}718\ \%$$

$$m_{^{238}\mathrm{U}} / m_{\mathrm{U,total}} = 99{,}282\ \%$$

NOTE The fissile systems with depleted uranium are bounded by the natural uranium systems considered in this International Standard.

#### 4.1.4.3 Plutonium

Three plutonium compositions, P0, P5 and P20, are considered:

c) composition P0 is defined by:

$$m_{^{239}\mathrm{Pu}} / m_{\mathrm{Pu,total}} = 100{,}000\ \%$$

d) composition P5 is defined by:

$$m_{^{239}\mathrm{Pu}} / m_{\mathrm{Pu,total}} = 95{,}000\ \%$$

$$m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} = 5{,}000\ \%$$

e) composition P20 is defined by:

$$m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} = 20{,}000\ \%$$

$$m_{^{241}\mathrm{Pu}} / m_{^{240}\mathrm{Pu}} = \mathrm{R}_1 = 11/17$$

$$m_{^{242}\mathrm{Pu}} / m_{^{241}\mathrm{Pu}} = \mathrm{R}_2 = 1/11$$

$$\begin{aligned} m_{^{239}\mathrm{Pu}} / m_{\mathrm{Pu,total}} &= 1 - m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} - m_{^{241}\mathrm{Pu}} / m_{^{240}\mathrm{Pu}} - m_{^{242}\mathrm{Pu}} / m_{^{241}\mathrm{Pu}} \\ &= 1 - m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} - (m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} \times \mathrm{R}_1) - (m_{^{240}\mathrm{Pu}} / m_{\mathrm{Pu,total}} \times \mathrm{R}_1 \times \mathrm{R}_2) \end{aligned}$$



### 4.1.5 Resulting fissile media

The six reference fissile media resulting from these physical and chemical forms and from these isotopic compositions are presented in Annex A.



## 4.2 Moderation conditions

Two water moderation degrees are considered:

a) a limited moderation corresponding to a water mass fraction less than or equal to 3,0 %, according to Equation (2):

$$w_{\mathrm{H_2O}} = \frac{m_{\mathrm{H_2O}}}{m_{\mathrm{H_2O}} + m_{\mathrm{PuO_2}} + m_{\mathrm{UO_2}}} \leq 3\% \qquad (2)$$

where

$m_{\mathrm{H_2O}}$ is the mass, in grams, of water in the mixture;

$m_{\mathrm{PuO_2}}$ is the mass, in grams, of plutonium dioxide in the mixture;

$m_{\mathrm{UO_2}}$ is the mass, in grams, of uranium dioxide in the mixture.

NOTE This degree of moderation is selected because MOX fuel is usually fabricated from mixtures of nearly dry powder and hydrogenated additives.

b) optimum moderation (minimal critical values obtained whatever the moderator-to-fuel ratio is).

These moderations are considered homogeneous in the fissile medium.

The mixture of MOX and water leads to a decrease in the MOX density from the theoretical density (full crystal dry density) as the water content increases in the mixture. For each mixture of MOX and water, the sum of their volume fractions (actual density divided by theoretical density) is unity.

In the case of a MOX density up to 3,50 g/cm$^3$, an initial void fraction is defined by this maximal density divided by the theoretical MOX density. The initial void fraction then allows a certain water content in the mixture with a constant MOX density. A further increase in the water content leads to a decrease in the MOX density. For each mixture of MOX and water, the sum of their volume fractions and the void fraction is unity.

### 4.3 Geometrical models

Critical values are given for the three following simple geometries of the fissile material:

- sphere;
- infinite length cylinder;
- infinite section slab.

### 4.4 Reflecting conditions

The critical values are given for a 2,5 cm and a 30,0 cm water reflector. The water reflector is close-fitting around the fissile material, with a free boundary beyond the reflector.



## 5 Critical values

### 5.1 Presentation of the results

Annex C (for a 30 cm water reflection) and Annex D (for a 2,5 cm water reflection) specify the lowest values of critical dimensions for a sphere, an infinite length cylinder, and an infinite section slab. These critical dimensions are the radius, in centimetres, and the volume, in litres, of the sphere, the diameter, in centimetres, of the infinite cylinder, and the thickness, in centimetres, of the infinite slab geometry. The critical volume of a sphere is the minimal critical volume whatever the credible fissile material geometry is.

Annex E (for a 30 cm water reflection) and Annex F (for a 2,5 cm water reflection) specify the lowest values of the critical parameters for the three reference geometries, in terms of mass, in kilograms, of actinide (U and Pu) for a sphere, linear density, in grams per centimetre, of actinide for a cylinder, and surface density, in grams per centimetre squared, of actinide for a slab geometry.

Each of these results is the lowest value resulting from the comparison of 12 to 17 values calculated with different calculation routes among those given in Annex B. For each value, at least four different computer codes and four different data libraries were used.

NOTE 1 All the computational results are extracted from References [1] to [7]. These calculations were performed with a temperature of 293 K.

NOTE 2 The results from References [1] to [7] show that the critical values for MOX with depleted uranium are not notably lower than the critical values in Annexes C to F for MOX with natural uranium.

### 5.2 Requirements

**5.2.1** Nuclear criticality safety assessors preparing specifications relative to fissile systems described in Clause 4 shall compare their own critical values with the critical values presented in Annexes C to F.

 