TRANSPORT AND STORAGE OF MOX FUELS
METHODOLOGY FOR CRITICALITY ASSESSMENT METHODS

E. Guillou(1), E. Lejeune(2), L. Aguiar(2)

(1) COGEMA La Hague, DSQ / IC / 50444 Beaumont-Hague, France
(2) SGN - 78182 St Quentin en Yvelines, France

INTRODUCTION

For spent fuel reprocessing in La Hague Plant, the authorization to unload these fuels into a pool is obtained further to analysis relative to their transport in cask and their storage in specific baskets.

One of the major risks arising during these operations is the criticality risk, present at each of the following steps :

- transport cask immersion
- fuel assembly handling
- loading, transfer and storage of fuel assembly baskets.

For UOX type assemblies, the criticality-safety of the foregoing operations is chiefly based on the fissile material limitation, characterized by the ^{235}U initial enrichment, which depends on geometric parameters such as its cross-section and characteristics of fuel rods and their number per assembly.

On the other hand, the criticality-safety analysis of MOX type assemblies, which can be made from mixtures of batches of plutonium from various sources, leads to a larger number of variables:

- Plutonium content
- Uranium and Plutonium isotopic composition
- assembly cross-section
- number of rods
- pellet diameter
- clad thickness
- rod diameter
- mixed oxide density.

Each value depends on the MOX concept used in the reactors. So far, transport and pool acceptance permits have been applied for by type of concept as a function of the scheduled reprocessing runs.

To optimize its approach, COGEMA has developed a criticality-safety demonstration aimed to cover the broadest possible range of MOX fuels with the minimum of constraints, in order to have a reduced number of permits. This overall approach is applied without accounting for a minimum burn-up.

The following two methods have been investigated and are proposed in this paper :

- the "optimal moderation" method, which ignores the number of missing rods to be guaranteed in the assembly, but which could lead to low permissible plutonium contents
- the "limited number of missing rods" method, which imposes a guaranteed number, but leads to higher permissible plutonium contents.

MAIN ASSUMPTIONS

Before distinguishing the specificities of the two methods, the assumptions common to the overall approach are set forth below.

The fissile medium is a lattice of rods of sintered mixed oxide UO_2 $+PuO_2$ clad with zirconium in water. The specific gravity of the oxide is taken as 11. The isotopic vector consists of the isotopes ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. The results presented below concern assemblies whose cross-sections do not exceed 214.5 x 214.5 mm^2. An equivalent study was conducted on assemblies with a cross-section of 230 x 230 mm^2.

These results also pertain to the handling of the assemblies and the pool storage of different types of baskets used in La Hague plant. For example, one of them has 9 cavities with a cross-section of 1010 x 1010 mm^2. Each cavity has an outside cross-section of 250 x 250 mm^2 and consists of stainless steel plate, jacketed with boron steel. The nominal centerline distance between the cavities is 305 mm.

The various configurations considered in these studies correspond to the normal operating conditions of the reception and storage facilities and to the incidents usually accounted for.

They are briefly described as follows:

a) normal configurations:

- infinite lattice of baskets with centered or off-centered assemblies
- transfer between pools.

b) incidental configurations:

- dropping of an assembly along a basket
- dropping of an assembly in the corner of a pool
- overturning of a basket fitted with its lid, with projection of the fuel assemblies outside the boron steel jackets.

Two acceptability criteria are set according to the type of situation considered.

In a normal situation, the acceptability criterion is set at keff + $3\sigma \leq 0.95$ and in an incidental situation at keff + $3\sigma \leq 0.97$.

The calculation scheme is the one recommended by the IPSN Criticality Safety Analysis Department: APOLLO 1 (with the CEA 86 library of cross-sections) and MORET III.

OPTIMAL MODERATION" METHOD

The first step is to carry out a parametric analysis, in infinite medium, as a function of the rod characteristics, in order to determine the neutron characteristics (infinite multiplication factor $k\infty$ and material buckling B^2m). This helps to select the most reactive rod (in the range of variation of PWR type rods), which is used throughout this second part of the study. The results show that the most reactive rod corresponds to:

- the largest pellet diameter
- the lowest clad thickness
- zero clad/pellet gap.

The second step consists to simulate the 3-D geometric configurations previously identified, and to establish the relations between the different components of the isotopic vector (U and Pu) and the Pu content, to obtain the acceptability range.

The ratios r1 $\left(= \frac{^{241}Pu}{^{240}Pu}\right)$ and r2 $\left(= \frac{^{242}Pu}{^{241}Pu}\right)$ firstly retained in this type of study are 0.65 and 0.09. Based on a fixed ^{240}Pu content, these ratios are used to determine the ^{241}Pu and ^{242}Pu contents, the complement to 100% being provided by ^{239}Pu. These values were determined with the aim of covering the widest number of isotopic compositions of plutonium obtained from the light water fuels identified at La Hague. These ratios proved to be highly limiting for the MOX fuels actually fabricated. A more detailed analysis of these fuels helped to relax the constraint on the ratios r1 and r2.

The pairs (r1 = 0.43 ; r2 = 0.3) and (r1 = 0.5 ; r2 = 0.5) were thus observed. Since some plutonium (particularly from UGR fuels) does not fit into these ranges, a rule, allowable by IPSN, of compensation of ^{242}Pu by an excess of ^{240}Pu was used.

The table below gives the permissible plutonium contents for different ^{240}Pu contents and various uranium matrices (case of the pair (r1 = 0.43 ; r2 = 0.3)):

U matrix (% of ^{235}U)	$\frac{^{240}Pu}{Pu_{total}}$ %	$\frac{Pu_{fissile}}{U_{total} + Pu_{total}}$ %
0.3	17	3.3
	20	3.7
	23	4.15
	25	4.4
0.72	17	3
	20	3.25
	23	3.6
	25	3.95

LIMITED NUMBER OF MISSING RODS" METHOD

With this method, the study in infinite medium showed that for a given type of assembly, the most reactive rod corresponds to:

- the smallest clad outside diameter
- the smallest oxide diameter.

The clad thickness fills the gap between the clad outside diameter and the oxide diameter. Note that in the ranges of variation of the oxide and clad thicknesses corresponding to a given type of assembly, the differences obtained in reactivity are slight. This helps to limit the number of guaranteed geometric parameters of the rod to one, the outside diameter.

The envelope ratios r1 = 0.65 and r2 = 0.09 have been retained.

On the other hand, the assemblies are assumed to be complete or nearly complete. We give below the permissible plutonium contents for assemblies containing 254 rods (complete assembly less 10 rods), uniformly distributed in the assembly cross-section.

U matrix (% of ^{235}U)	$\frac{^{240}Pu}{Pu_{total}}$ %	$\frac{Pu_{fissile}}{U_{total} + Pu_{total}}$ %
0.3	17	5.45
	20	6.5
	23	7.6
	25	8.3
0.72	17	5.
	20	6.15
	23	7.1
	25	7.9

CONCLUSIONS

To be rapidly usable and transcribed for permit applications, this set of results was processed to develop linear formulas of the type:

$$\frac{\text{Pu fissile}}{U_{total} + Pu_{total}} \leq k1 \times \frac{^{240}Pu}{Pu_{total}} + k2$$

In the "optimal moderation" method, these equations are accompanied by constraints on the ratios r1 and r2 (for example, $r1 \leq 0.43$ and $r2 \geq 0.30$). In the "limited number of missing rods" method, they are accompanied by constraints on the rod diameter ; the pair ($r1 = 0.65$; $r2 = 0.09$) covers all the Plutonium produced by La Hague Plant.

The overall approach recommended by COGEMA has given rise to generic studies which yielded a series of permissible values of plutonium contents as a function of the isotopic vector and for different types of assembly.

The reactivity of these MOX assemblies can be compared with that of UOX assemblies, in order to determine what is called a MOX-UOX "equivalence formula".