Criticality evaluation of BWR MOX fuel transport packages using average Pu content

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Abstract

Currently in France, criticality studies in transport configurations for Boiling Water Reactor Mixed Oxide fuel assemblies are based on conservative hypothesis assuming that all rods (Mixed Oxide (Uranium and Plutonium), Uranium Oxide, Uranium and Gadolinium Oxide rods) are Mixed Oxide rods with the same Plutonium-content, corresponding to the maximum value. In that way, the real heterogeneous mapping of the assembly is masked and covered by a homogeneous Plutonium-content assembly, enriched at the maximum value. As this calculation hypothesis is extremely conservative, COGEMA LOGISTICS has studied a new calculation method based on the average Plutonium-content in the criticality studies. The use of the average Plutonium-content instead of the real Plutonium-content profiles provides a highest reactivity value that makes it globally conservative. This method can be applied for all Boiling Water Reactor Mixed Oxide complete fuel assemblies of type 8×8, 9×9 and 10×10 which Plutonium-content in mass weight does not exceed 15%; it provides advantages which are discussed in our approach.

With this new method, for the same package reactivity, the Pu-content allowed in the package design approval can be higher. The COGEMA LOGISTICS' new method allows, at the design stage, to optimise the basket, materials or geometry for higher payload, keeping the same reactivity.

1. Introduction

Many designs of Boiling Water Reactor (BWR) Mixed Oxide (MOX) (uranium and plutonium) fuel assemblies exist: assembly type (rod geometry, moderation, lattice), mapping (MOX, UO_2 and $UO_2Gd_2O_3$ rods position in the assembly), axial and radial Pu-content profiles vary from one type of fuel assembly to another.
The current method of calculation allows to overcome the difficulties due to mapping, axial and radial Pu-content profiles by assuming that all rods (MOX, UO_2, $UO_2Gd_2O_3$ rods) are MOX rods with the same Pu-content, corresponding to the maximum value. As this calculation hypothesis is extremely conservative, Cogema Logistics has performed a study in order to demonstrate that it is possible to use the average Pu-content values for criticality calculations of BWR MOX fuel assemblies in transport and storage packages.

2. Methodology

To demonstrate that the use of the average Pu-content is acceptable, the impact on the reactivity of each parameter, which characterises a fuel assembly, has been studied. The differences in reactivity values have been examined by performing calculations using the average Pu-content and calculations using the real heterogeneous representations. The comparisons made cover two configurations:

- infinite lattice of fuel assemblies of the same type,
- fuel assemblies loaded in a transport cask.

The study is divided into different steps, the main ones being:

- identification of the main parameters to be taken into consideration, such as assembly type (rod geometry, moderation and lattice), mapping, axial and radial Pu-content profiles and type of basket (the assemblies are arranged in a basket),
- definition and validation of a calculation system, referenced "Transn[1]ucléaire Référence Criticité MOX REB (TNRC MOX REB)" (section 4.2), applicable to BWR MOX fuel assemblies, allowing their real heterogeneous mapping and Pu-content profile description,
- definition of a calculation method using the average Pu-content called 'Transnucléaire Production Criticité MOX REB (TNPC MOX REB)" and validation of this method against TNRC MOX REB while various parameters are varying (isotopic vector, Pu-content, mapping).

3. Identification of parameters and glossary

The BWR MOX fuel assemblies present different characteristics that make difficult their simplified description. Physical parameters that allow differentiating and studying them in details are:

- type of assembly (lattice): 8×8, 9×9, 10×10, mapping: term defining the location of fuel rods, poisoned rods, UO_2 rods and water holes in a fuel assembly,
- Pu-content: term defining the total weight of plutonium of a fuel rod divided by the weight of plutonium plus the weight of uranium,
- Pu-content profile: term defining the radial or the axial distribution of the Pu-content in the rods of the fuel assembly,
- isotopic vector: term characterising the weight fractions of each isotope (in percent),
- All these parameters and their role on the conservative aspects of the average Pu-content in criticality studies have been considered in details.

4. Calculation methods

4.1. Present calculation method

Currently the criticality studies related to the transport and storage of BWR MOX assemblies are performed using the standard APOLLO1 ("Option Super Cellule") – MORET3 (or APOLLO2 –MORET4) calculation codes. The main hy-

[1] TRANSNUCLEAIRE™ , trademark of COGEMA LOGISTICS

pothesis is to consider that all rods (MOX, UO_2, $UO_2Gd_2O_3$ rods) are MOX rods with the same Pu-content, corresponding to the maximum value among the rods of the fuel assembly.
The APOLLO1[1)] code (Option Super Cellule) solves the transport equation by the probability of collision method in a one dimension space and considers 99-energy groups. Then cross sections are collapsed and homogenised in a 16-energy groups representation of the entire fuel assembly. Finally, the three dimensional criticality code MORET3[2)], using fuel cross sections (obtained from APOLLO1) and HANSEN&ROACH[3)] cross sections (for structural parts), solves the transport equation using the Monte Carlo method and provides the effective multiplication factor (K_{eff}).
The current method is easy to use but too much conservative .

4.2. Reference calculation method

As criticality experiments and benchmarks for BWR MOX configurations studies are only few, and as no easy-to-use calculation method exists to describe the real assembly mapping, Cogema Logistics has defined a reference calculation method called "TNRC MOX REB".
The reference method TNRC MOX REB uses the sequence of APOLLO1 and MORET3 codes, which are supplied with the CEA86 library (99-groups) and the following options:

<u>APOLLO1 code</u>

- rod-by-rod description of the fuel assembly, in two dimensions, and transport equation resolution by the probability of collision method,
- entire assembly homogenisation, using the transport-transport equivalence module,
- production of the cross-sections for the entire assembly,
- specific calculation of resonant nuclei self-shielding,
- 16-energy groups collapse,

<u>MORET3 code</u>

- representation of the entirely homogenised fuel assemblies,
- three-dimensional solving of the transport equation using the Monte Carlo method.

This method has been validated by comparing it to the TRIPOLI4 polykinetic code[4)], recognised as one of the reference codes for neutron transport, and which has been selected as the numerical standard.
Using a detailed description of the assembly, the TRIPOLI4 code requires a laborious preparation of the input files, a big computer memory and needs a long time of computer calculation. For these reasons, this method can not be adopted as an every day calculation method. On the contrary, after validation the TNRC MOX REB method was used as the reference method to qualify the standard method and especially for the use of the average Pu-content.

4.3. Calculation method using the average Pu-content

The studies performed in order to use the average Pu-content in criticality calculations of BWR MOX fuel assemblies in transport and storage packages, have shown that the most appropriated method is the method called "TNPC MOX REB". This method is based on the APOLLO1 (Option Super Cellule) – MORET3 (or APOLLO2 – MORET4) codes applied with the following options:

- $UO_2Gd_2O_3$ and UO_2 rods are replaced by MOX rods with the maximum Pu-content,
- the main hypothesis is to consider all rods at the same Pu-content, corresponding to the average Pu-content of the fuel assembly,
- the axial distribution of Pu-contents is not taken into account; the greatest value of rods Pu-content defines the fuel section to be considered,
- the water holes are "diluted" around the fissile part of the fuel. It means that the moderation of the fuel assembly will be added in the same proportion as the number of water holes.

The APOLLO1 code (Option Super Cellule) using the CEA86 neutron library makes a one dimensional calculation of the cross-sections in a 99-energy groups representation. Then these cross-sections are collapsed and homogenised over the whole fuel cell in a 16-energy representation. The three dimensional criticality code MORET3 using the fuel cross-sections provided by APOLLO1 and HANSEN&ROACH cross-sections for structural parts allows to obtain the effective multiplication factor (K_{eff}).

5. Calculations and results

The results presented in the following sections give $\Delta K_i / K_i$ with

$$\frac{\Delta K_i}{K_i} = \left(\frac{K_i^{TNPC\ MOX\ REB} - K_i^{TNRC\ MOX\ REB}}{K_i^{TNRC\ MOX\ REB}} \right) \qquad (1)$$

where "K_i" means "$K_{¥}$" for an infinite lattice of fuel assemblies and "K_{eff}" for the case of loaded transport casks.

5.1. General calculation hypothesis

The calculation hypothesis adopted for the validation of the average Pu-content method include the following ones:

- all fuel rods are considered with the average Pu-content,
- Pu-content of the fuel rod is the one which corresponds to the maximum axial Pu-content,
- the maximum Pu-content is limited to 15% in mass of Pu,
- the assembly has no missing rods and respects the following assessments shown in Table 1.

Table 1 -Assessments on the fuel assembly

Lattice	**8´ 8**	**9´ 9**	**10´ 10**
Maximal Enrichment of the UO_2 rods [%]	3	3	3
Minimum number of UO_2-Gd_2O_3 rods	11	14	6
Minimum Gd_2O_3 content per UO_2-Gd_2O_3 rods [%]	2	2	1,25
Maximal ^{235}U content per UO_2-Gd_2O_3 rods [%]	4,9	4,9	3,95
Maximal area of the water channel given in water cell number	4	7	9

5.2. TNPC MOX REB validation for infinite lattice of fuel assemblies

The TNPC MOX REB is validated by comparing it to TNRC MOX REB for infinite lattices of fuel assemblies. The results of this study for $\Delta K_\infty / K_\infty$ 1) given in pcm (1 pcm is equal to $10^{-5} \times \Delta K_\infty / K_\infty$) are shown on Figure 1. while two parameters are varying: Pu-content and mapping.

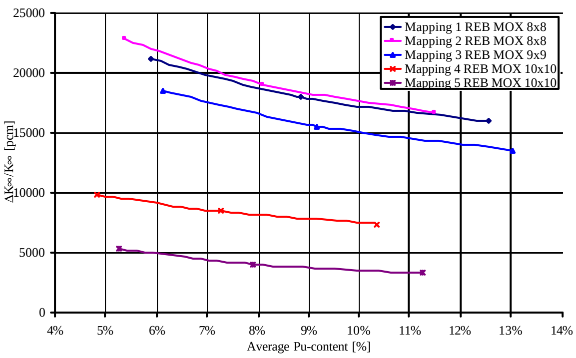

Fig.1 Comparison between TNPC MOX REB and TNRC MOX REB for infinite lattices of fuel assemblies.

The results obtained on the basis of average Pu-content cover those obtained with real mappings.

5.3. TNPC MOX REB validation for transport casks

Two kinds of casks have been examined: the MX6 and the TN12/2BWR packagings. These packagings present different characteristics: casks with several housings which positions are symmetrical or non-symmetrical, several heterogeneous water holes, borated steel plates and complex plates systems.
The results of this study for $\Delta K_{eff}/K_{eff}$ [1)] given in pcm (1 pcm is equal to $10^{-5}\times\Delta K_{eff}/K_{eff}$) are shown on Figures 2 and 3 while two parameters are varying: Pu-content and mapping.

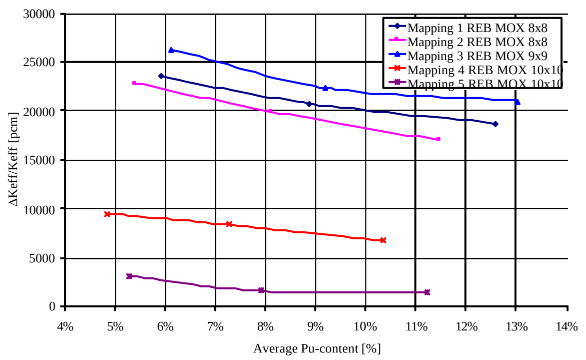

Fig.2 Comparison between TNPC MOX REB and TNRC MOX REB for an isolated MX6 package.

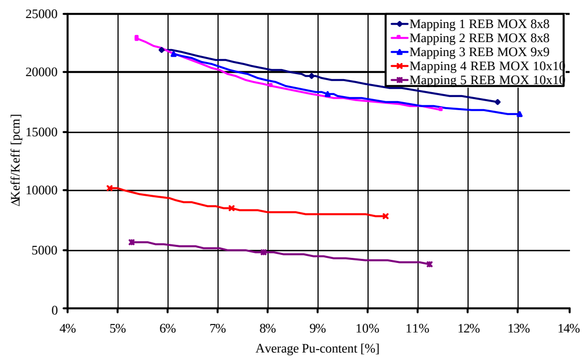

Fig.3 Comparison between TNPC MOX REB and TNRC MOX REB for an isolated TN12/2BWR package.

Compared to the reference method TNRC MOX REB, the production method TNPC MOX REB provides highest reactivity values for isolated transport casks.
All these results are valid while the isotopic vector is varying.

6. Consequences of using the average Pu-content

The reactivity difference observed between the calculation method using the average Pu-content and the present calculation method may reach 5100 pcm (1 pcm is equal to $10^{-5}\times\Delta K_{eff}/K_{eff}$).
The gain of reactivity margin obtained without any safety risk can be translated in terms

- either of plutonium content,
- or of design options for baskets.

The gain obtained by using the average Pu-content instead of the maximum value of Pu-content depends on the mapping and more precisely on the difference between the maximum Pu-content and the average Pu-content. It can reach 4.7% in term of Pu-content which means that, by using the average Pu-content, the transport and storage of BWR MOX fuel assemblies can be realised with an higher Pu-content for the same design.
As for design baskets options, two ways of improvements are proposed:

- selection of basket materials with less boron content for reduction of costs,
- optimisation of the basket geometry for higher payload.

7. Conclusion

COGEMA LOGISTICS' work about the use of the average Pu-content as the basis for criticality safety studies of MOX boiling water reactor fuel assemblies in transport and storage packages, have shown that the use of the average Pu-content instead of the real Pu-content profiles is conservative. This conclusion is valid for BWR MOX assemblies defined in the available scope and does not depend on the type of cask.

The new production method allows to gain reactivity margins (of 5100 pcm in the case of 16 10×10 BWR MOX assemblies loaded in a MX6 cask). The gain of reactivity margin obtained with equal safety, can be translated in terms either of higher average Pu-content of the fuel assemblies in the package approval, or of better and more adapted packaging designs. Concerning this last option, the two possible design optimisations are:

- selection of basket material with less neutron poison content,
- new basket geometry to accommodate more fuel assemblies, for higher payload.
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Compared to the reference method TNRC MOX, the production method TNPC MOX provides higher reactivity values that makes this method globally conservative. It can be used for criticality studies of BWR MOX fuel assemblies, in transport and storage packages defined in the available scope, without any safety risk.

With this new method, for the same package reactivity, the Pu-content allowed in the package design approval can be higher. The COGEMA LOGISTICS' new method allows, at the design stage, to optimise the basket, materials or geometry for higher payload, while keeping the same reactivity.

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