CONSEQUENCES OF THE USE OF THE AVERAGE PU-CONTENT FOR CRITICALITY EVALUATION OF PWR MOX-FUEL TRANSPORT AND STORAGE PACKAGES

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ABSTRACT

Currently, criticality studies of Pressurized Water Reactor (PWR) Mixed OXide (MOX) (uranium and plutonium) fuel assemblies in transport and storage packages are based on conservative hypothesis by assuming that all rods have 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 assembly, with a plutonium content at the maximum value. As this calculation hypothesis is extremely conservative, Transnucléaire and Cogema have developed a new calculation method based on the use of the average Pu-content value in the criticality studies. The use of the average Pu-content, instead of the real Pu-content profiles, provides higher or equivalent reactivity values that make it globally conservative. This method can be applied for all MOX PWR complete or incomplete fuel assemblies, which Pu-content in mass does not exceed 15 %; it provides advantages which are discussed in the paper.

INTRODUCTION

Many designs of Pressurized Water Reactor (PWR) Mixed Oxide (MOX) (uranium and plutonium) fuel assemblies exist: assembly type (rod geometry, moderation, lattice), mapping, Pu-content profile vary from one type of fuel assembly to another.

The current method of calculation allows to overcome the difficulties due to mapping and Pu-content profile by assuming that all rods have the same Pu-content, corresponding to the maximum value. As this calculation hypothesis is extremely conservative, Transnucléaire has performed a study with a view to demonstrate that it is possible to use the average Pu-content values for criticality calculations of PWR fuel assemblies in transport and storage packages.

METHODOLOGY

To demonstrate that the use of the average Pu-content is acceptable, the impact on reactivity of each parameter which characterizes a fuel assembly, has been studied. The difference in reactivity values have been looked for by calculation using average Pu-content and those 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 broken down 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, Pu-content profile and type of basket (the assemblies are arranged in a basket),

- definition and validation of a calculation system, referenced TNRC MOX (paragraph 4.2), applicable to PWR MOX fuel assemblies, allowing their real heterogeneous mapping and Pu-content profile description,
- application of the calculation method using the average Pu-content value for two complete 17×17 assemblies loaded in a transport cask,
- comparison between the method using the average Pu-content and TNRC MOX while various parameters are varying (isotopic vector, Pu-content, mapping),
- extension of the comparison to the other types of fuel assemblies and casks,
- study of PWR MOX with missing rods.

IDENTIFICATION OF PARAMETERS AND GLOSSARY

The PWR fuel assemblies present different characteristics that make difficult their simplified description. Physical parameters that allow to differentiate and study them in details are:
- type of assembly (lattice): 14×14, 15×15, 16×16, 17×17, 18×18,
- mapping: term defining the location of fuel rods, water holes in a fuel assembly,
- Pu-content: term defining the total weight of plutonium of the assembly divided by the weight of plutonium plus the weight of uranium,
- Pu-content profile: term defining the radial distribution Pu-content in the rods of fuel assembly,
- isotopic vector: term characterizing the weight fractions of each isotope (in percent),
- missing rods: absence of several fuel rods in the assembly, which is in this case incomplete.

All these parameters and their role on the conservative aspects of average Pu-content in criticality studies, have been considered in details.

CALCULATION METHODS

Present calculation method

At present the criticality studies related to the transport and storage of PWR MOX assemblies are performed using the standard APOLLO1 (Option Super Cellule) – MORETIII calculation codes. The main hypothesis is to consider that all rods have the same Pu-content, corresponding to the maximum value among the rods of fuel assembly.

The APOLLO1 code (Option Super Cellule) [1] 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 homogenized in a 16-energy group representation of the entire fuel assembly. Then, the three dimensional criticality code MORETIII [2], using fuel cross sections (obtained from APOLLO1) and HANSEN&ROACH [3] cross sections (for structural parts), solves the transport equation by Monte Carlo method and provides the effective multiplication factor (K_{eff}).

The current method is easy to use but too much conservative.

TNRC MOX Reference calculation method

As criticality experiments and benchmarks for PWR MOX configurations studies are only a few, and no easy-to-use calculation method exists to describe the real assembly mapping, Transnucléaire defined a reference calculation method called TNRC MOX.

The reference method TNRC MOX uses the sequence of APOLLO1 and MORETIII codes, which are supplied with the CEA86 library (99-groups) and the following options:

APOLLO1:

- Rod-by-rod description of the fuel assembly, in two dimensions, and transport equation resolution by the probability of collision method,
- Entire assembly homogenization, 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,

MORETIII:

- Representation of the entirely homogenized fuel assemblies,
- Three-dimensional solving of the transport equation by the Monte Carlo method.

This method has been validated against the TRIPOLI4 polykinetic code [4], recognized 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 laborious input preparation, a big computer memory and needs a long time 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 method was used as the reference method to qualify the standard method and especially for the use of the average Pu-content.

Calculation method proposed for PWR MOX criticality studies

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

- the main hypothesis is to consider all rods at the same Pu-content, corresponding to the average Pu-content of 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 one dimensional calculation of the cross-section in a 99-energy groups representation. Then these cross-sections are collapsed and homogenized over the whole fuel cell in a 16-energy representation. The three dimensional criticality code MORETIII using the fuel cross-sections provided by APOLLO1 and HANSEN&ROACH cross-sections for structural parts allows to obtain the multiplication effective factor.

Validation of the TNPC MOX method

The TNPC MOX method described above has been validated, against TNRC MOX method, within two configurations:

- infinite lattice of two 17×17 fuel assemblies,
- three transport casks : casks with one (FS65) or several lodgements which positions are non-symmetrical (TN12/2, MX8), several heterogeneous water holes (MX8), borated steel plates and complex compression systems (MX8, TN12/2).

The differences in reactivity between TNPC MOX and TNRC MOX are extremely small and are included in the code statistical uncertainties (-150 pcm as a maximum for casks loaded with the first type of fuel assemblies, and +230 pcm as a maximum with the second type of fuel assemblies, compared to the code statistical uncertainty of 200 pcm).

CALCULATIONS AND RESULTS

General calculation hypotheses

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

- all the fuel rods are considered with the average Pu-content,
- fuel rod Pu-content is that which corresponds to the maximum axial Pu-content,
- maximum Pu-content is limited to 15% in mass of Pu.

Influence of Pu-content, isotopic vector

The studies are based on the assessment of two fuel assemblies. Average Pu-content from 4,5% to 12%, and three isotopic vectors are considered.

The results obtained on the basis of average Pu-content cover, or are at least equivalent to, those obtained with real maps.

All these results are valid for both sets values of multiplication factors, infinite and effective ones.

The use of the average Pu-content is independent of the Pu-content degree and the isotopic vector.

Application to other PWR MOX assemblies

The examination of others cases of 14×14, 15×15, 16×16 and 18×18 (with different mapping, Pu-content profile, rod geometry, repartition and number of water holes) fuel assemblies has confirmed the above conclusions about the use of the average Pu-content for all PWR MOX fuel assemblies.

The assessment has confirmed that, with an infinite lattice of fuel assemblies, however heterogeneous it may be, mapping has a low or no effect on reactivity. It is indeed observed that differences in reactivity between the average Pu-content case (when the rods in the fuel assemblies are at the average Pu-content), and all cases where the assembly has a non-homogeneous radial profile in terms of Pu-content, are extremely small.

Application to different casks

Different kind of casks have been examined (FS65, TN12/2, MX8): casks with one or several lodgements which positions are non-symmetrical, several heterogeneous water holes, borated steel plates and complex compression systems.

Similar results are observed for both methods TNRC MOX and TNPC MOX. Therefore, the conservatism or equivalent aspect of average Pu-content is confirmed and does not depend on the type of cask considered.

Influence of missing rods

This study is made of two parts:

- determine the most reactive position of the missing rods by considering all the rods at the same Pu-content,

- study the effect on reactivity of missing rods for real mapping, by using the first part conclusion.

In infinite lattice of assemblies and in transport casks configurations, the most reactive position of the missing rods have been determined by removing one by one or two by two the rods on the symmetrical axes and on the crowns of the assemblies which contain all the rods at the same Pu-content. This assessment has proved that the most reactive position of the missing rods is situated at the external border of the assembly. This conclusion has been also obtained with the real mapping.

Compared to the reference method TNRC MOX, the production method TNPC MOX provides highest or equivalent reactivity values when the rods are removed. The difference between TNRC and TNPC MOX increases with the number of missing rods.

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 2300 pcm.

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 of the mapping and more precisely of the difference between the maximum Pu-content and the average Pu-content. It can reach 2.5% in term of Pu-content that means that, by using the average Pu-content, the transport and storage of PWR MOX fuel assembly can be realized with 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
- optimization of basket geometry for higher payload.

CONCLUSION

The works performed about the use of the average Pu-content as the basis for criticality safety studies of MOX pressure 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 (or equivalent). This conclusion is valid for all PWR assemblies and does not depend on the type of cask.

The new production method allows to gain reactivity margins (of 2300 pcm in the case of 8 17×17 PWR MOX assemblies loaded in a MX8 cask). The gain of reactivity margin obtained with equal safety, can be translated in terms either of higher average Pu-content of fuel assemblies in package approval, or of better and more adapted packaging designs. Concerning this last option, the two possible design optimizations are:

- selection of basket material with less neutron poison content
- new basket geometry to accommodate more fuel assemblies, for higher payload.

Compared to the reference method TNRC MOX, the production method TNPC MOX provides higher or equivalent reactivity values that makes this method globally conservative. It can be used for criticality studies of PWR 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 new method allows, at the design stage, to optimize the basket, materials or geometry for better payload, keeping the same reactivity.

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