REACTOR PHYSICS CALCULATIONS ON MOX FUEL IN BOILING WATER REACTORS (BWRs)

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Abstract

The loading of MOX (Mixed Oxide) fuel in BWRs (Boiling Water Reactors) is considered in this paper in a "once-through" strategy. The fuel assemblies are of the General Electric 8×8 type, whereas the reactor is of the General Electric BWR/6 type. Comparisons with traditional UOX (Uranium Oxide) fuel assemblies revealed that the loading of MOX fuel in BWRs is possible, but this type of fuel creates new problems that have to be addressed in further detail. The major ones are the SDM (Shutdown Margin) and the stability of the cores at BOC (beginning of cycle), which were demonstrated to be significantly lowered. The former requires a new design of the control rods, whereas a modification of the Pu isotopic vector allows improving the latter. Another issue with the use of the MOX fuel assemblies in a "once-through" strategy is the increased radiotoxicity of the discharged fuel assemblies, which is much higher than of the UOX fuel assemblies.

Introduction

In the past few years, the possibility of the use of MOX fuel in commercial nuclear power stations has been demonstrated in some western countries. Whereas MOX fuel has been loaded mostly in PWRs (pressurised water reactors), such a type of fuel is considered to be used also in BWRs (boiling water reactors).

This paper investigates the possibility of loading MOX fuel assemblies in BWRs, both in a partial or full MOX loading (the purpose of this paper is not to compare the possibility of loading MOX fuel assemblies in BWRs to the loading of such fuel assemblies in PWRs). These fuel assemblies are assumed to be used in a "once-through" strategy, i.e. the discharged MOX fuel assemblies are considered as nuclear wastes. The multirecycling of Pu is out of the scope of this paper. Several aspects are thus studied: the fuel inventory (major/minor actinides, and fission products), the fuel performance in terms of Pu utilisation, the reactivity coefficients (both at the assembly and the core level), the point-kinetic parameters and the stability of the core. For that purpose, a generic model of a General Electric BWR/6 has been chosen. The fuel assemblies are all of the General Electric 8×8 type, but three different lattices are presented in this paper: a UOX lattice used as a reference lattice, and a UOX and a MOX lattice used for the mixed UOX/MOX loading (the full MOX loading was carried out with the same type of MOX fuel assemblies).

In the first section of this paper, the assemblies and the core are described, and so are the hypotheses used for the assembly/core calculations. In the second section, an analysis of the fuel assemblies is carried out, and their main properties are summarised. In the third section, the core characteristics are analysed.

Description of the fuel assemblies and the core for the study

In this section, the fuel assemblies and the reactor, which were chosen for this study, are presented. The hypotheses used to perform the assembly and core calculations are also emphasised.

Fuel assemblies

In order to study the ability of BWR cores to be loaded with MOX assemblies and to develop a core equilibrium model, several types of assemblies are necessary. First of all, one needs fresh UOX fuel, which will constitute the first loading (cycle 1).[1] One also needs fresh UOX fuel that will be used for a full UOX loading and that will be loaded according to the same strategy from cycle to cycle until the equilibrium core is reached. These assemblies are referenced in the following as the *u00* lattice. Since the main aim of this paper is to compare the MOX cores to the UOX cores, it was decided to use typical UOX assemblies for these aforementioned ones. They are of the General Electric 8×8 type and are based on Ref. [1].

From this full UOX equilibrium, a MOX equilibrium must be reached, either in a partial MOX loading or a full MOX loading. This requires defining both a UOX and a MOX assembly (referenced in the following as the *u01* and *m01* lattices respectively), since the properties of one assembly depend on its environment.[2]

1. The characteristics of these bundles are not touched upon in the paper.

2. The same MOX bundle was used for both the partial and the full MOX loading.

For the MOX assembly, it is necessary to define the quality of plutonium, which is used in this study. The plutonium isotopic vector corresponds to the reactor-grade plutonium (64.1% fissile) that can be recovered from spent UOX assemblies. This quality is defined in Table 1 below.

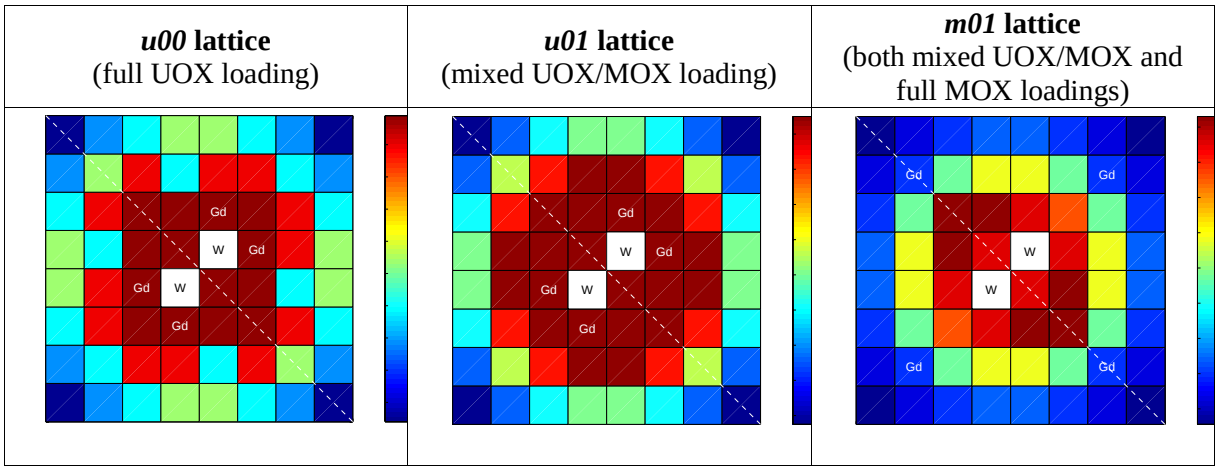
Table 1. **Nominal Plutonium isotopic vector**

Pu isotope	**Mass fraction** (%)
^{238}Pu	2.7
^{239}Pu	56.0
^{240}Pu	25.9
^{241}Pu	8.1
^{242}Pu	7.3

The lattices for the full UOX loading, for the mixed UOX/MOX and full MOX loadings are presented in Figure 1. The lattice for the full UOX loading is taken from Ref. [1], and represents a typical General Electric 8×8 UOX fuel assembly that will be considered as a reference fuel assembly for the rest of this study. The UOX and MOX lattices used in the mixed UOX/MOX loading (and the full MOX loading for the MOX fuel assembly) were determined via an iterative optimisation process (see Ref. [2] for further details) so that the desired cycle length was achieved, the power peaks inside the fuel assemblies and between the fuel assemblies (in case of a mixed UOX/MOX loading) were avoided, there was no mismatch between the Gadolinium and the fuel depletion, and finally the control rod efficiency was good enough.

It has to be pointed out that the fuel density has been reduced from 10,3 g/cm^3 for UOX assemblies to 9,78 g/cm^3 for MOX assemblies in order to keep the optimum of the moderator to fuel ratio.[3] This fuel density allows having an under-moderated lattice at EOL (end of life).

Figure 1. **Enrichment (%) of the lattices in ^{235}U and/or Pu**
(South-East corner of the control blades)



3. This optimum is larger for MOX fuel since the multiplication factor is higher due to ^{239}Pu.

Core

The nuclear reactor is supposed to be of the General Electric BWR/6 type. The cycle characteristics, assuming that they can be applied to each cycle in the equilibrium model, are described in Table 2 below.

Table 2. **Cycle characteristics for a general electric BWR/6**

Cycle length	369 days (≈ 12.3 months)
Capacity factor	85%
Cycle length at full power	314 EFPD (effective full power days)
Power density	50 kW/l
Core rated thermal power	3 380 MWth
Core rated flow	47.25×10^6 kg/hr
Heated fraction of flow	85%
Number of assemblies	764
Assembly nominal active fuel height	381 cm (cold)
Fraction of discharged assemblies	≈ 1/4
UOX assemblies (for both the full UOX and the mixed UOX/MOX loading) Fuel density EOC core-averaged burn-up	 10.3 g/cm^3 19 061 GWd/tHM
MOX assemblies (for both the mixed UOX/MOX and the full MOX loading) Fuel density EOC core-averaged burn-up	 9.78 g/cm^3 20 075 GWd/tHM
Number of MOX assemblies (for mixed UOX/MOX loading)	280

In order to be able to compare the different cores (core with a full UOX loading, core with a mixed UOX/MOX loading, core with a full MOX loading), one has to define a common "strategy" for them, i.e. one has to apply the same rules to reload the cores. Furthermore, only the cores that have reached equilibrium, i.e. cores that have the same properties from cycle to cycle if one applies the same reloading pattern, can be used for these comparisons.

The loading strategy was carried out according to the Control Cell Core (CCC).[4] Once the loading pattern has been chosen, it remains to determine how the fuel will be depleted. The depletion was carried out in accordance with the Haling principle (see Ref. [3]), which is a hypothetical depletion, but which can be applied to any loading.[5]

4. In the CCC, the control of the reactor is limited to a fixed group of control rods (rods in the A2 control cells) and low reactivity fuel assemblies surround the control rods.

5. The Haling principle states "The minimum peaking factor for a given fuel loading arrangement is achieved by operating the reactor so that the power shape (power distribution) does not change appreciably during the operating cycle".

Based on this Haling principle, an equilibrium model was developed for each loading.[6] The only constraint that was used in this model was to reach a target effective multiplication factor of 1.00000 (±100 pcm) at EOC (end of cycle). This was realised by modifying the reload batch size (approximately one fourth of the core) and by readjusting the loading pattern accordingly. Optimising the power distribution and readjusting the enrichment of the assemblies were clearly out of the scope of this study.

Codes

The codes used to perform all the calculations were CASMO-4, TABLES-3, and SIMULATE-3 from Studsvik Scandpower (see Refs. [4-6]). The neutron transport calculations were carried out with 70 groups, and the cross-section library was based on ENDF/B-IV. The CASMO-4 models rely on the method of the collision probabilities, the method of the response matrix, and the method of the characteristics, whereas the SIMULATE-3 models are based on the 2-group nodal diffusion equations (see Ref. [7]).

Fuel assembly analysis: fuel inventory/performance and reactivity coefficients

Although this paper focuses on the use of MOX fuel in BWRs, i.e. the characteristics of cores loaded with MOX fuel, it is essential to study such fuel at the assembly level via CASMO-4. The main properties of interest are of course the fuel inventory and the performance of the fuel assemblies. But the reactivity coefficients can also be studied for each assembly type assuming an infinite loading pattern constituted by identical fuel assemblies.

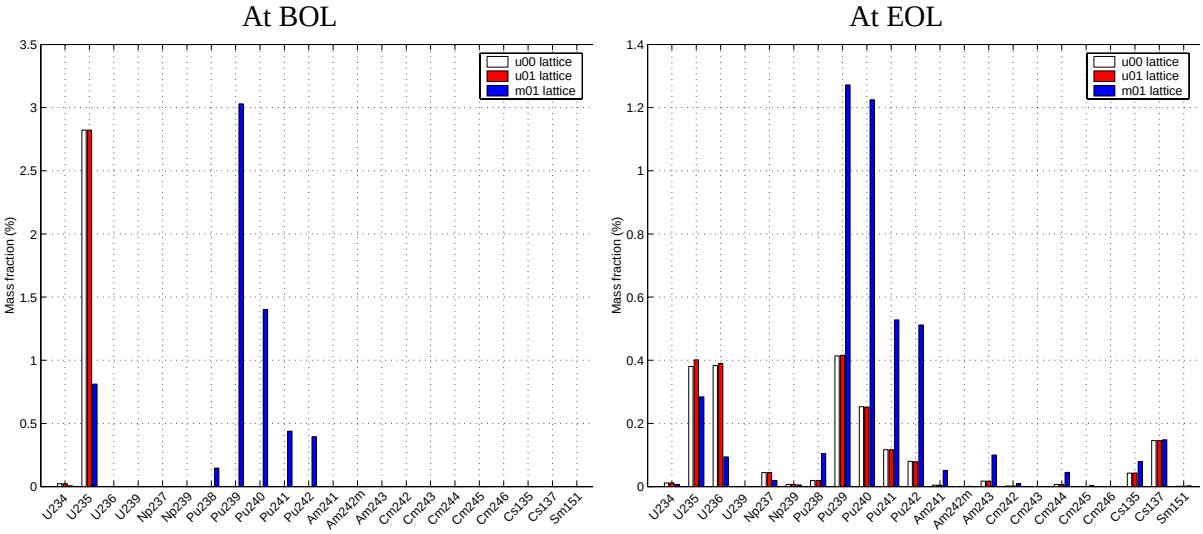
Regarding the fuel inventory, Figure 2 gives the mass distribution of the minor and the major actinides, ^{135}Cs, ^{137}Cs, and ^{151}Sm[7] at BOL (beginning of life) and EOL, for a void content of 40%. As can be seen on this Figure, the main difference between the UOX fuel assemblies (*u00* and *u01* lattices) and the MOX assembly (lattice *m01*) is the relatively high content of some of the minor actinides in the MOX fuel assemblies at EOL, the concentration of which is 5 to 15 times higher than for the UOX fuel assemblies. Concerning the U-Pu isotopes, one notices that roughly more than half the amount of ^{239}Pu and of ^{235}U has been burnt in the MOX fuel compared to their initial amount. The amount of the other Pu isotopes is larger at EOL than at BOL due to the Pu build-up by neutron capture/radioactive decay. The MOX fuel reduces considerably the amount of ^{236}U and ^{237}Np compared to the UOX fuel. Nevertheless, one notices also that the ^{135}Cs, and ^{151}Sm fission products have much higher concentrations at EOL for MOX assemblies than for UOX assemblies, and their

6. An equilibrium cycle consists of establishing a loading pattern using a fresh batch of fuel (fixed number of assemblies at a fixed enrichment) and then depleting and shuffling using the same fuel design and loading pattern for multiple cycles. Equilibrium is reached when exposure and power distributions converge to one solution, which does not change in succeeding cycles.

7. Regarding the radiotoxicity of the wastes, several fission products, which can be grouped into three categories (see Ref. [8]), have to be studied. The first group is constituted by ^{99}Tc, ^{129}I, ^{93}Zr, ^{107}Pd, and ^{135}Cs, and represent the long-lived isotopes that can be transmuted in a fast spectrum much faster than their natural decay (only ^{135}Cs is accessible in CASMO-4 using the ENDF/B-IV library). ^{90}Sr, ^{137}Cs, and ^{151}Sm represent isotopes that are not worth transmuting since their natural decay is similar or much faster than their transmutation (only ^{137}Cs and ^{151}Sm are accessible in CASMO-4 using the ENDF/B-IV library). Finally, the last group (^{79}Se, ^{126}Sn, and ^{94}Nb) is made of isotopes that cannot be transmuted rapidly because of their relatively small cross-sections (none of these isotopes is available in CASMO-4 using the ENDF/B-IV library). Fortunately, the radiotoxicity of these isotopes is rather small since their yield is limited.

concentration is roughly twice as high as for UOX assemblies (the amount of ^{137}Cs is almost identical in the UOX and MOX fuel assemblies). Regarding the radiotoxicity of these fuel assemblies, it can be seen on Figure 3 that the radiotoxicity of the MOX fuel assembly (lattice *m01*) is roughly one order of magnitude higher than the radiotoxicity of the UOX fuel assemblies (Figure 3 gives the radiotoxicity for the *u01* lattice, which is identical to the one for the *u00* lattice). The main actinides contributing to the radiotoxicity are of course the Pu isotopes,[8] the concentration of which is much higher in the MOX than in the UOX assemblies. The effect of the Am and Cm isotopes (mainly ^{241}Am and ^{244}Cm) is also roughly ten times higher in the MOX than in the UOX assemblies. The radiotoxicity of ^{237}Np, although lower in the MOX assembly than in the UOX assembly just after discharge, becomes higher a few hundreds of years after discharge due to the α decay of the accumulated ^{241}Am. Taking all these facts into consideration, the use of MOX fuel in BWRs as a means of burning the fissile Pu is possible since the amount of ^{239}Pu is reduced by half. Nevertheless, the radiotoxicity of such discharged fuel assemblies is roughly ten times higher than the one of conventional UOX fuel assemblies.

Figure 2. **Fuel inventory for the different lattices** (calculations performed at 40% of void)



Regarding the behaviour of the fuel assemblies in case of a mixed UOX/MOX loading, power discontinuities can occur at the boundaries between UOX and MOX assemblies because of the larger absorption cross-section of plutonium fuel. It is thus necessary to check the properties of the UOX and MOX segments (*u01* and *m01* respectively) in a checkerboard pattern, via the use of CASMO-4 (see Table 3 and Figure 4).

8. The radiotoxicity of some of the fission products such as ^{90}Sr is known to be actually larger than the one of the Pu isotopes just after the fuel discharge and to decrease significantly after a few hundreds of years, but these fission products are not present in the ENDF/B-IV library.

Figure 3. **Evolution of the radiotoxicity of the discharged fuel assemblies**

u01 *lattice* *m01* lattice

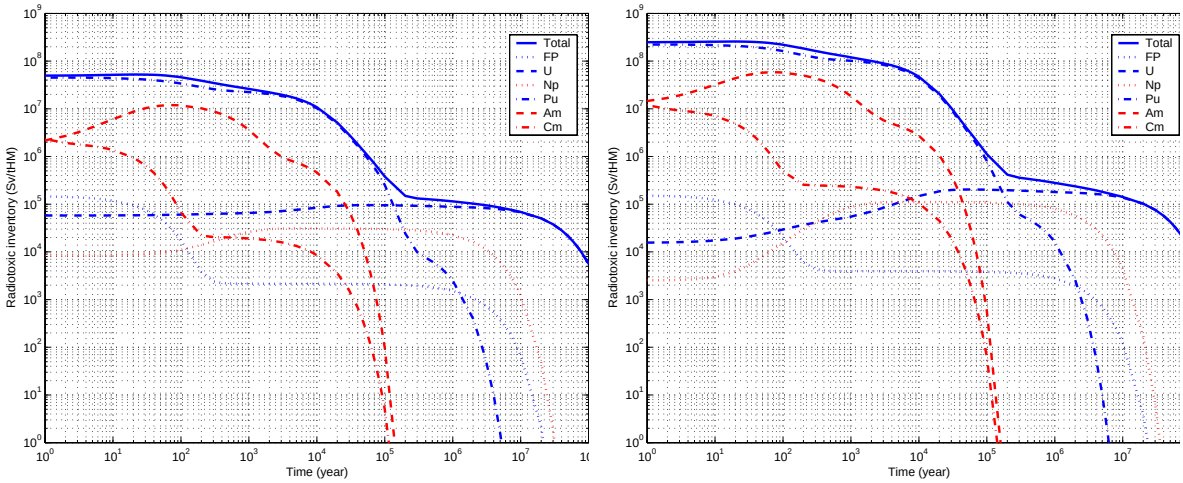
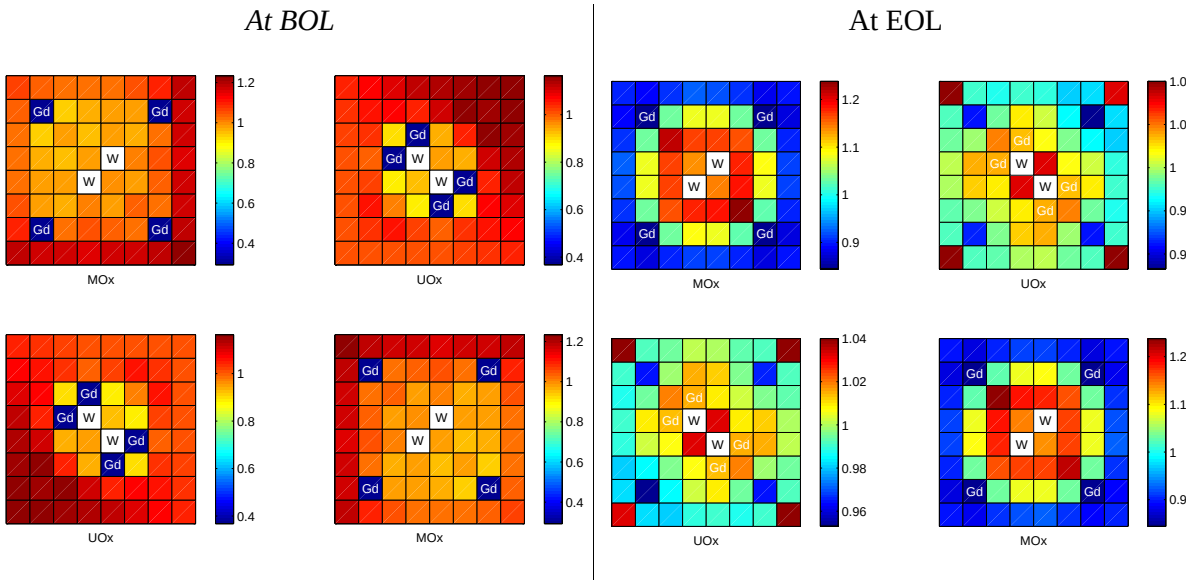


Table 3. **Comparison between the UOX and the MOX assemblies in a checkerboard pattern**

	At 0 GWd/tHM		**At 40 GWd/tHM**	
	UOX	MOX	UOX	MOX
Burn-up (GWd/tHM)	0.000	0.000	38.302	41.788
$\nu\Sigma_f/\Sigma_a$ (1)	1.13827	1.11642	0.87403	0.93249
2-group k_∞ (1)	1.14816	1.10338	0.87952	0.92650
2-group M^2 (cm^2)	86.11	83.77	87.69	87.84
Fission fraction (1)	1.023	0.977	0.941	1.059
Absorption fraction (1)	0.943	1.057	0.959	1.041
Nu*fission fraction (1)	0.952	1.048	0.927	1.074
Power fraction (1)	1.009	0.991	0.938	1.062

One notices that the UOX assembly contributes to most of the energy release at BOL, whereas the contribution of the MOX assembly becomes more significant during the depletion and even larger than the UOX one. This is mostly due to the higher macroscopic absorption cross-section in the MOX segment at BOL compared with the UOX segment. Consequently, the depletion rate is higher in the MOX segment; this is why the corresponding burn-up is larger at EOL. Since the infinite multiplication factor for the MOX segment at hot conditions without control rod is larger than the one for the UOX segment at high burn-up, the relative importance of the MOX assembly becomes larger during the depletion. As expected also, the flux is harder in the MOX assembly because the thermal absorption is much larger (the corresponding macroscopic absorption cross-section is larger). Finally, the power distribution is relatively good during the depletion. The power peak for the MOX assembly is moving from the periphery at BOL towards the centre of the assembly at EOL, since the depletion rate is larger along the (wide) water gaps. For the UOX assembly, the power peak located along the water gaps at BOL totally vanishes during the fuel depletion.

Figure 4. **Power distribution in the UOX and MOX segments in a checkerboard pattern** (40% of void)



Concerning the reactivity coefficients of each assembly modelled in an infinite lattice, the Doppler and the void coefficients of reactivity are presented in Figures 5 and 6, respectively. If one relies on the classical four-factor formula, these reactivity coefficients can also be analysed in a qualitative manner. Regarding the fuel temperature coefficient, the most important effect is due to the decrease of the resonance escape probability in the resonance group when the fuel temperature increases (increase of the resonance integral). This is a negative reactivity effect. On the other hand, there is a positive contribution of ^{239}Pu and ^{241}Pu for MOX fuel. This positive effect is triggered by the larger increase of the fission rate than the capture rate when the fuel temperature rises. ^{240}Pu contributes also to the Doppler coefficient but in a negative manner (strongly at BOL, slightly at EOL) to the resonance escape probability because of the increase of the capture rate solely when the fuel temperature increases. It thus results that the Doppler coefficient is slightly more negative for MOX assemblies than UOX assemblies. The analysis of the void coefficient is slightly more complicated since many competing effects are involved when the void content changes. When the void content increases, the neutron mean free path increases, the fraction of neutrons captured in the resonances during the slowing-down increases, and so does the neutron age. All these effects have a negative reactivity contribution. On the other hand, two main effects induce a positive contribution when the void content rises: the decrease of the fraction of thermal neutrons captured in the moderator, and the increase of the fast fission factor due to the increase of the flux around the fission energies. In case of MOX fuel, the harder spectrum (higher flux in the resonance region) gives an increase in the change of the capture rates, so that the variation of the resonance escape probability becomes more negative. For the same reason, this spectrum hardening strengthens the changes in the fast fissions, so that the fast fission factor becomes more positive. Nevertheless, the effect on the resonance escape probability is larger than the effect on the fast fission factor, and consequently the void coefficient is lowered (more negative). It thus results that the void coefficient is more negative for MOX assemblies than for UOX assemblies (except at high void fraction, where the effect on the fast fission factor becomes larger for MOX fuel than for UOX fuel).

Core analysis: reactivity coefficients and stability

In the previous section, the reactivity coefficients have been calculated for each assembly via CASMO-4. In this Section, the reactivity coefficients are calculated for the whole core via SIMULATE-3. The following Table 4 summarises the main core characteristics for the full UOX core, the mixed UOX/MOX core, and the full MOX core.

In summary, one can say that the uniform Doppler coefficient is almost unchanged in case of MOX assemblies, whereas the void coefficient (most of the MTC – Moderator Temperature Coefficient) is more negative, as expected. The power coefficient is slightly larger (more negative), and so is the flow coefficient (more positive). The pressure coefficient is also slightly smaller at BOC (Beginning Of Cycle), but larger at EOC.

The most significant changes are the variation of the effective fraction of delayed neutrons (due to ^{239}Pu and ^{240}Pu, 210 pcm and 490 pcm respectively, compared with ^{235}U, 650 pcm), the variation of the prompt neutron lifetime, and the reduction of the control rod efficiency. This reduction is so strong that the shutdown margin becomes smaller than 1% at BOC. The high value for the thermal absorption cross-section of the MOX fuel causes a reduction in the mean free path of thermal neutrons, which reduces the thermal neutron absorption probability for the control rod.

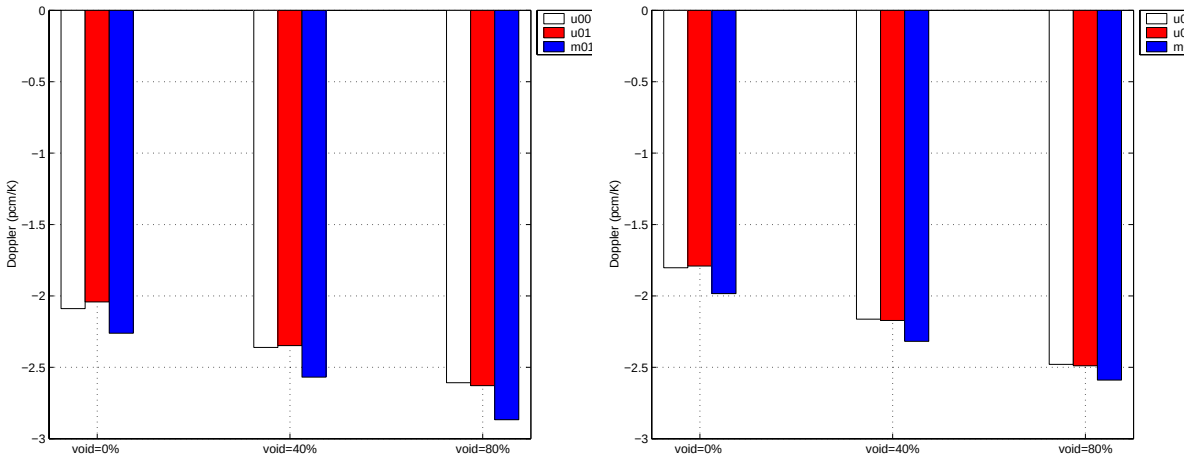
It was decided, for the UOX/MOX core exclusively, to model a variation of the Pu isotopic vector in order to study the evolution of the previous parameters. A variation of 50% of each Pu isotope (relative to their nominal value) was carried out. A variation of the Pu-2*xx* isotope is referenced as *mxx* and *pxx*, for -50% and +50% of variation respectively in the following Figures. All the absolute results are given, and also their variation relative to the full UOX core (for the UOX/MOX core and the full MOX core) or relative to the nominal UOX/MOX core (for a Pu isotopic vector variation). Regarding the variation of the Pu quality, one notices that there is no significant change with the Pu isotopic vector (only a significant reduction of ^{239}Pu gives a decrease of the power coefficient) except for the SDM (Shutdown Margin), as depicted in Figure 7. The SDM can become greater than 1% if one reduces the ^{239}Pu amount and/or one increases the ^{240}Pu amount. This is the result of the large absorption cross-section of ^{239}Pu (≈1 035 barn in the thermal region), whereas the ^{240}Pu absorption cross-section is relatively small (≈197 barn in the thermal region). Consequently, any reduction in the absorption cross-section tends to increase the control rod efficiency, as explained before. The influence of the other Pu isotopes is lower, since their nominal proportion is relatively small compared with ^{239}Pu and ^{240}Pu.

Another parameter that is essential to estimate in case of BWRs is the stability of the reactor. It is well known that three main types of instability may occur in a BWR (see Refs. [9,10]): the fundamental mode or in-phase oscillation (in which the fundamental flux is oscillating over the whole core), the first azimuthal mode or out-of-phase oscillation (in which the first azimuthal flux is oscillating over the whole core), and the channel instability or density wave oscillation (in which the flux is oscillating only in a certain channel). In this study, only the first type of instability is investigated in a qualitative manner via the so-called March-Leuba model (see Ref. [11]). This model allows estimating the transfer function of the reactor (from reactivity perturbation to normalised neutron density variation), which can be reduced to a second-order model in the frequency range of interest for the in-phase type of oscillations. From this transfer function, it is possible to calculate the Decay Ratio (DR), defined as the ratio between two consecutive maxima of the impulse response of the normalised neutron density (the DR gives a measurement of the damping of the system). The full derivation of the DR ratio from the March-Leuba model is given in Ref. [2].

The DR was found to be sensitive to several parameters used in the March-Leuba model. The first one is related to the fuel density, and it was shown that a lower fuel density renders the core slightly more unstable. As pointed out previously (see Table 2), the fuel density of MOX fuel assemblies is lower than the one of UOX fuel assemblies, and thus the reactor is expected to be less stable in case of a mixed UOX/MOX loading and a full MOX loading. Likewise, it was demonstrated that the DR was larger (core more unstable) when the effective fraction of delayed neutron was smaller, due to the proximity of prompt-criticality. MOX cores are thus expected to be less stable since the effective fraction of delayed neutrons is smaller than full UOX cores (see Table 4). Finally, the DR was proven to be sensitive to a parameter called in Ref. [2] the amplification factor of the void response, the parameter *C*. This parameter is a destabilising factor if it is larger since the void response is a large delayed negative response that could be out of phase with the initial reactivity perturbation, and thus could enhance it. MOX cores are thus less stable since the void coefficient is larger (less negative). Core calculations revealed that *C* was very sensitive to the Pu quality (mostly ^{239}Pu and ^{240}Pu) at BOC. As can be seen on Figure 8, the DR decreases significantly at BOC if one reduces the ^{240}Pu content or/and one increases the ^{239}Pu content. This is due to the fact that in these cases, the void coefficient and consequently the *C* parameter are less negative, thus improving the stability. This stability improvement disappears during the cycle because of the fuel depletion.[9]

Figure 5. **Comparison between the Doppler coefficients of reactivity for the different assemblies**

At BOL At EOL



9. It was also shown that the DR was dependent on the residence time of steam bubbles in the fuel channels. This does not depend on the fuel type loaded in the core (UOX or MOX), but more on the thermal-hydraulic design of the fuel assemblies.

Figure 6. **Comparison between the void coefficients of reactivity for the different assemblies**

At BOL At EOL

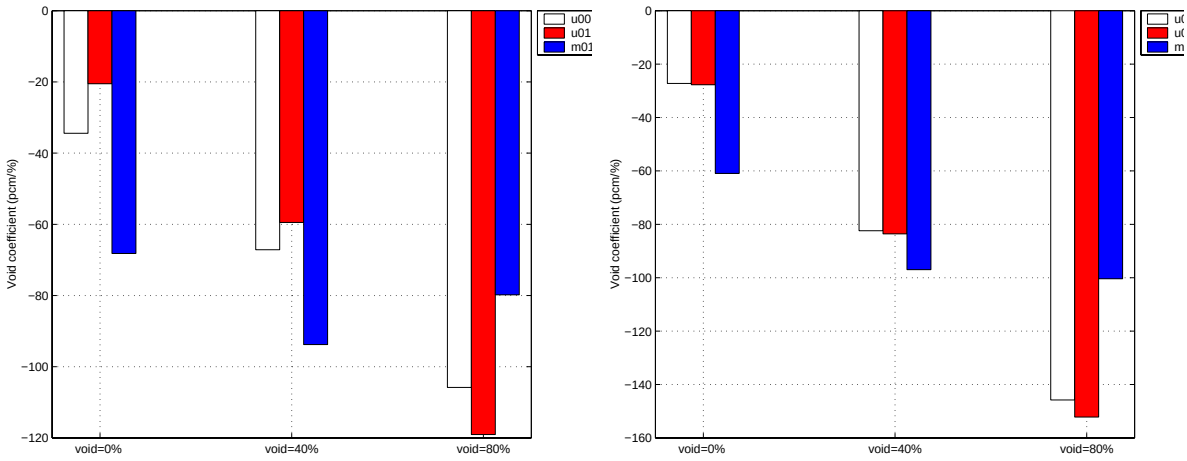


Table 4. **Summary of the reactivity coefficients and the associated parameters for the three types of cores**

	Full UOX core		**Mixed UOX/MOX core**		**Full MOX core**	
	BOC	EOC	BOC	EOC	BOC	EOC
Moderator temperature coefficient (pcm/°F)	-55.44	-57.01	-57.69	-60.37	-64.46	-62.84
Uniform doppler coefficient (pcm/°F)	-1.12	-1.08	-1.10	-1.15	-1.19	-1.16
Power coefficient (pcm/%power)	-41.80	-45.81	-42.66	-47.37	-45.52	-46.85
Flow coefficient (pcm/%flow)	31.63	36.57	32.53	37.66	34.72	37.22
Pressure coefficient (pcm/psia)	9.32	8.43	7.96	8.82	8.78	9.00
Effective fraction of delayed neutrons (pcm)	604	532	536	491	408	417
Prompt neutron lifetime (μs)	39.6	41.4	34.1	35.9	24.5	26.5
Control rod worth (pcm)	35 023	35 965	32 132	33 251	26 854	28 164
Shutdown margin (%)	2.426	3.756	0.737	3.039	0.705	2.343
Average discharge burn-up (GWd/tHM)	–	28.3	–	29.9/35.6 UOX/MOX	–	33.2

Figure 7. **Analysis of the shutdown margin**

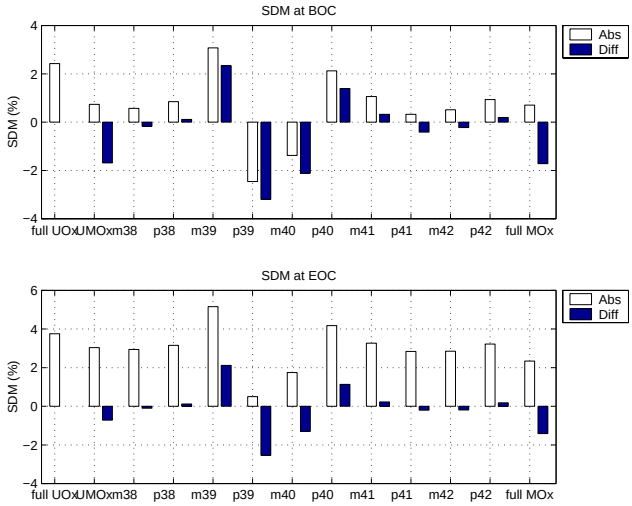
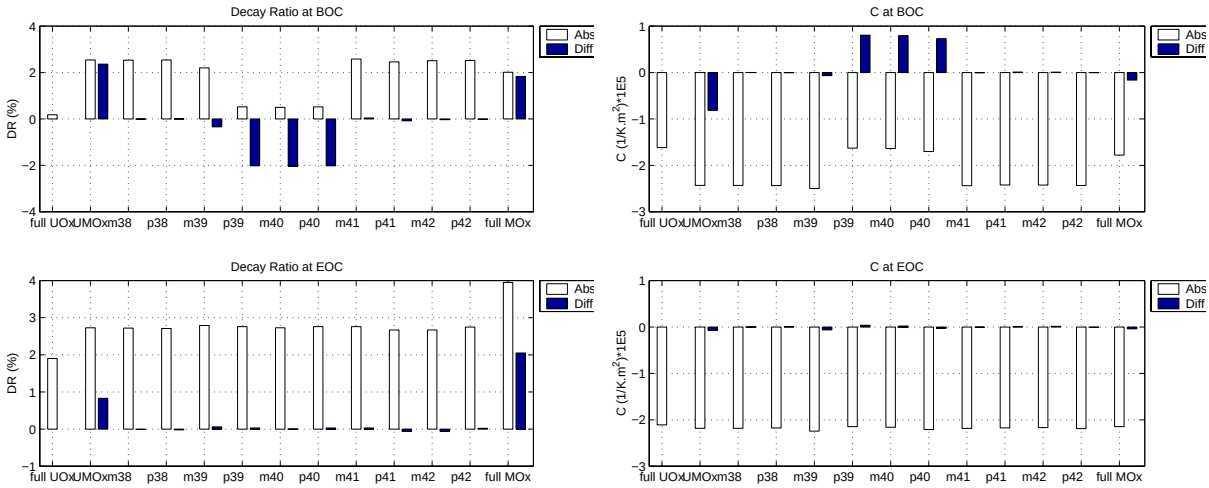


Figure 8. **Analysis of the decay ratio and of the amplification factor C of the void response**



The following Table 5 summarises the results of the stability calculations for the mixed UOX/MOX core (with the nominal Pu isotopic vector) and the full MOX core. Only the comparison with the full UOX core is presented since the simple model used in this study does not allow an absolute determination of the Decay Ratio, i.e. only a qualitative analysis is possible.

One notices that the mixed and full MOX cores are much less stable at BOC than at EOC in comparison with the full UOX core. This is mostly due to the fuel depletion and the production of Pu isotopes by the fresh UOX assemblies in the full UOX cores, so that the differences in the core characteristics die out with burn-up. Although the MOX cores (mixed UOX/MOX and full MOX cores) are more than ten times less stable at BOC, full power, and full core flow than the full UOX core, it is difficult to give an absolute value for the DR since the absolute DRs are too low to be realistic. Furthermore, it is not granted either that the same ratio can be applied to any part of the power-flow map (this study only investigated the full power and full core flow case). Nevertheless, it is very likely that the same tendency can be noticed for other points of the power-flow map, i.e. that the MOX cores are much less stable than the UOX cores. Therefore, other points in the power-flow map (where the stability of the full UOX core is known to be very low) might have a DR very close to instability or might even show limit-cycle oscillations for MOX cores.

Table 5. **Stability of the mixed UOX/MOX and full MOX cores** (at full power and full core flow)

	Ratio between the UOX/MOX core DR and the full UOX core DR (absolute DRs)	**Ratio between the full MOX core DR and the full UOX core DR** (absolute DRs)
BOC	14.11 (2.54%/0.18%)	11.17 (2.01%/0.18%)
EOC	1.44 (2.73%/1.90%)	2.08 (3.95%/1.90%)

Conclusions

The main goal of this study was to determine the possibility of loading MOX fuel in BWRs. In order to be able to compare the core characteristics, it was necessary to develop three equilibrium core models: a full UOX core, a mixed UOX/MOX core (approximately 2/3 of UOX assemblies and 1/3 of MOX assemblies), and a full MOX core.

The assembly/core calculations showed that it seems to be possible to load MOX assemblies in BWRs (either in a mixed UOX/MOX pattern, or a full MOX pattern), since most of the core characteristics are comparable with the ones of a full UOX core. Nevertheless, three main problems arise: the shutdown margin at BOC is lower than 1%, the value requested by the safety authorities; the cores with MOX fuel are less stable; finally and most importantly, the radiotoxicity of the discharged MOX fuel is significantly larger than of UOX fuel.

One way of solving the two first problems could be to modify the Pu isotopic vector: a variation of the ^{239}Pu and/or the ^{240}Pu content affects very much both the SDM and the DR. Unfortunately, these variations are not compatible: if one increases the ^{239}Pu content for instance, both the SDM and the DR decrease, whereas the opposite behaviour occurs with ^{240}Pu. Consequently, one needs to modify both the Pu quality for the stability and also the control rod efficiency for the SDM, i.e. a new control rod design is necessary. The same conclusion regarding the SDM was also drawn in Ref. [12] (where it was demonstrated that increasing the B-10 content of the control rods allowed improving the SDM). This may be a major obstacle to the loading of MOX fuel, unless the SDM safety limit is lowered. Considering the SDM, the stability, and the radiotoxicity, MOX fuel assemblies in BWRs raise major concerns that have to be solved prior to their extensive use in commercial nuclear reactors.

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Nomenclature

BOC	Beginning of Cycle
BOL	Beginning of Life
BWR	Boiling Water Reactor
CCC	Control Cell Core
DR	Decay Ratio
EFPD	Effective Full Power Day
EOC	End of Cycle
EOL	End of Life
FP	Fission Products
MOX	Mixed Oxide
MTC	Moderator Temperature Coefficient
PWR	Pressurised Water Reactor
SDM	Shutdown Margin
UOX	Uranium Oxide