EXPERIMENTAL VALIDATION OF EFFECTIVE DELAYED NEUTRON FRACTION IN LWR-MOX CORE

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ABSTRACT

Integral measurements of the effective delayed neutron fraction were performed during the MISTRAL experimental programme launched in the EOLE facility at Cadarache center (France). Two homogeneous core configurations are studied in this paper. MISTRAL1 is a 3.7% UOX homogeneous core considered as a reference for the second homogeneous core MISTRAL2 (7% MOX). These measurements consist in the determination of the effective delayed neutron fraction by the noise analysis method coupled with the determination of the absolute total fission integral of the core. The kinetic parameters of the two cores are calculated from direct and adjoint transport calculations with the french neutronic deterministic code APOLLO2.5 and the CEA93.V6 library based on the european JEF2.2 nuclear data evaluation. Cell and core calculation are performed. Accurate core calculations are in good agreement with experiment, except the UOX core calculation which seems to slightly overpredict the β_{eff} value: C/E factors are +2.4% $\pm$ 1.6% (1σ) in the UOX case and +0.1% $\pm$ 1.6% (1σ) in the MOX case.

1 INTRODUCTION

Integral measurements of the effective delayed neutron fraction were performed during the MISTRAL experiment carried out in the EOLE zero-power reactor at Cadarache center (France). MISTRAL [1] was an experimental programme launched by the french Atomic Energy Commission (CEA) within the framework of a collaboration with its industrial partners and the japanese Nuclear Power Energy Corporation (NUPEC) [2]. It consisted in four configurations. The experiment which started in 1996 and ended mid-July 2000, was devoted to Advanced Light Water Reactor studies with high moderation lattices and loaded with 100% MOX fuel.

Two homogeneous core configurations are studied in this paper (Figure 1). The first one, MISTRAL1, is a 3.7% UOX homogeneous core considered as a reference for the second core MISTRAL2 loaded with 7% MOX fuel. The first regular and homogeneous core (MISTRAL1) consists in about 750 UO_2 fuel pins with a lattice pitch of 1.32 cm, leading to a moderation ratio of about $V_{mod}/V_{fuel} = 1.7$ and $H/HM = 5.1$. MISTRAL2 is an homogeneous MOX core (7% w/o Pu) involving about 1600 fuel pins and the same square pitch than MISTRAL1.

Several physical parameter were investigated with the four configurations on this programme as core reactivity, reactivity temperature coefficient, absorber worth and so on. Effective delayed neutron fraction (β_{eff}) was determined in the two first cores MISTRAL1-UOX and MISTRAL2-MOX.

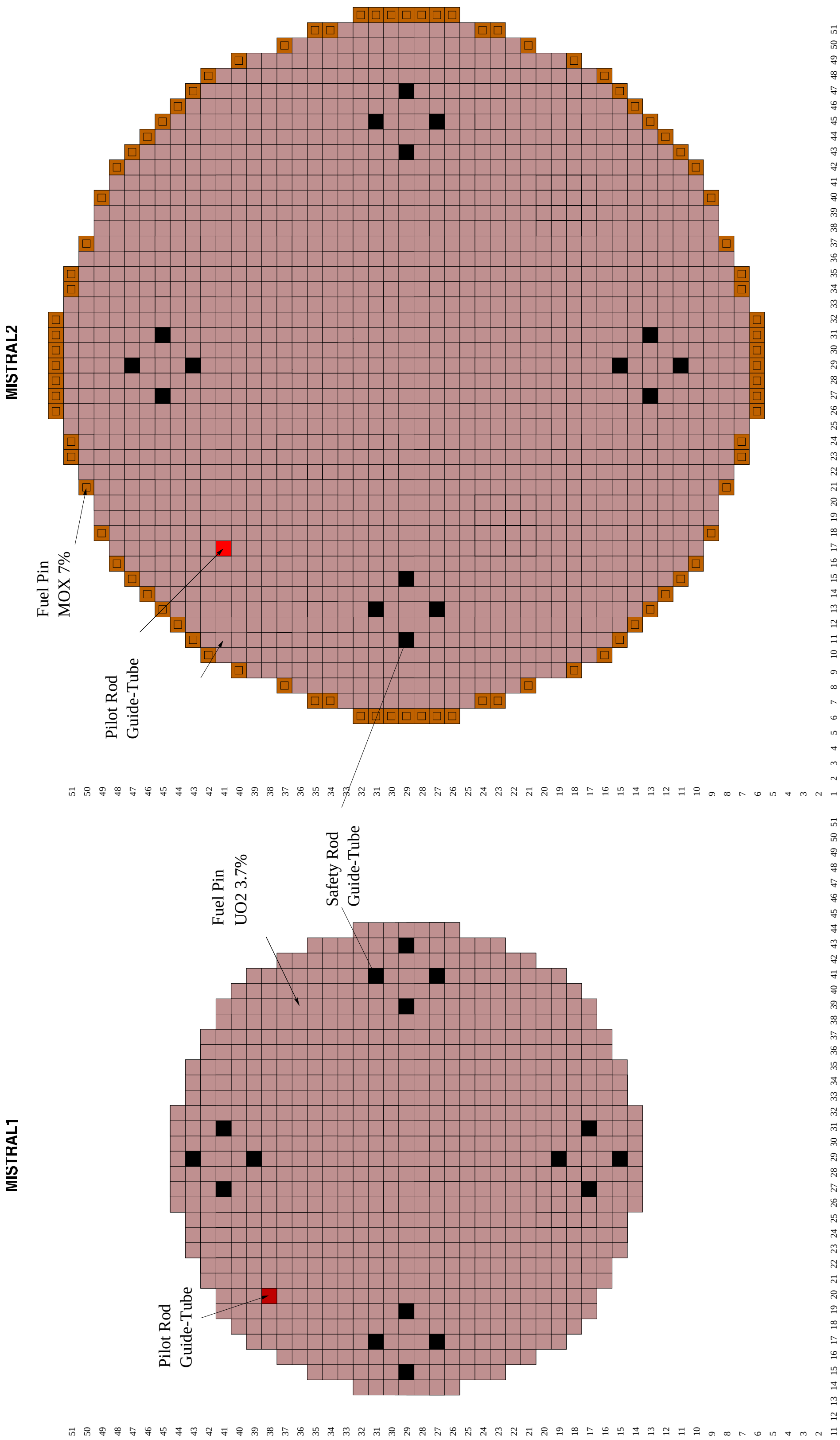

Figure 1: Radial cross section of Mistral1 and Mistral2 cores

2 DELAYED NEUTRON FRACTION MEASUREMENT

These measurements consist in the determination of the effective delayed neutron fraction by the noise analysis method coupled with the determination of the absolute total fission integral of the core. In a steady state reactor, reactivity fluctuations are produced by natural fluctuations of the fission and capture microscopic processes which lead to a variation of the counting rate observed by sensitive counters.

$$\beta_{eff}^2 = \frac{2D}{F} \frac{VV'}{DSPI} \frac{1}{(1+\rho)^2} \tag{1}$$

where D is the Diven factor which represents the dispersion of the average prompt neutron emission during fission, F is the fission integral, ρ the reactivity in dollars, V and V' the average voltage of fission chamber amplifier and $DSPI$ the spectral density of interaction power. This value is decomposed in two parts : a first one which is measured (V, V', $DSPI$ and ρ) and a second one which is calculated (D and a part of F). The uncertainties of these different parameters are :
D : $\pm 2\%$,
F : $\pm 2\%$ (uncertainty on the fissile mass of the deposit in the fission chamber),
$DSPI$: $\pm 1\%$,
V, V' : $\pm 0.5\%$,
ρ : $negligible$.

3 CALCULATION ROUTE

The kinetic parameters of the two cores are obtained from direct and adjoint transport calculations. These calculations are performed with the french neutronic deterministic code APOLLO2.5 and the CEA93.V6 library [3] which is based on the european JEF2.2 nuclear data evaluation.
A first rough analysis is performed in a critical pincell pattern. In this case, the Boltzmann integral equation is solved using P_{ij} method and 172 energy groups (Xmas energy group structure). An $UP1$ interface current method is used : interface angular fluxes are assumed to be linearly anisotropic and leakage (P_{is}) and transmission (P_{ss}) probabilities are computed in the exact 2D geometry. An accurate space-dependent self-shielding formalism is considered to describe the resonance absorption during the neutron slowing-down.
A more accurate analysis is performed with a core calculation : the integro-differential form of the Boltzmann equation is solved using S_N-nodal calculations in a 20-energy group structure. This core calculation is performed in $XY-2D$ dimensions and the third dimension is patterned by an axial buckling. The use of an heterogeneous/homogeneous equivalence process allows the determination of homogenised cell cross-sections for the $S8/P1$ calculation, using linear-linear nodal resolution method with one mesh per cell.
In APOLLO2 code, delayed neutron fraction for each l precursor group are calculated from the formula 2 :

$$\beta_l = \frac{\int d\vec{r} \int dE \; \Phi^\dagger(\vec{r},E) \sum_j \chi_{l,j}^d(E) \int dE' \; \nu_{l,j}^d(E')\Sigma_{f,j}(E')\Phi(\vec{r},E')}{\int d\vec{r} \int dE \; \Phi^\dagger(\vec{r},E) \sum_j \chi_j(E) \int dE' \; \nu\Sigma_{f,j}(E')\Phi(\vec{r},E')} \tag{2}$$

where parameters have the following meaning:
l and j : precursor group and fissile isotope respectively.
$\Phi^\dagger(\vec{r},E)$ and $\Phi(\vec{r},E)$: adjoint and direct flux respectively.

$\chi^d_{l,j}(E)$: fractional delayed neutron emission spectrum.
$\nu^d_{l,j}(E)$: fractional delayed neutron yield.
$\chi_j(E)$: total fission spectrum.
$\nu\Sigma_{f,j}(E)$: total production cross-section.

The generation time is calculated in the following way :

$$\Lambda = \frac{\int d\vec{r} \int dE \ \Phi^\dagger(\vec{r},E) \frac{1}{v(E)} \Phi(\vec{r},E)}{\int d\vec{r} \int dE \ \Phi^\dagger(\vec{r},E) \sum_j \chi_j(E) \int dE' \ \nu\Sigma_{f,j}(E')\Phi(\vec{r},E')} \tag{3}$$

We have to calculate the direct and adjoint flux to be able to produce a β_{eff} value. Note that in JEF2, in the case of $U235$, the 6 time group decay constants of the precursors (λ_l) are not consistent with the fractional delayed neutron yields. λ_l are those from Keepin [4] and $\nu^d_{l,j}(E)$ comes from Brady and England [5] as mentioned in [6].

4 MISTRAL1 UOX CORE ANALYSIS

For both cell pattern (with critical leakage) and MISTRAL1 core calculations, effective delayed neutron fraction for each precursor group, total fraction (β_{eff}) and neutron generation time (Λ) are indicated in Table 1.

Table 1: Kinetic parameters - MISTRAL1

Group (i)	Cell β_i (pcm)	Core β_i (pcm)
1	26.5376	26.3011
2	143.010	141.581
3	136.168	134.596
4	311.741	308.507
5	143.150	141.281
6	55.5694	54.9117
β_{eff} (pcm)	816.1	807.2
Λ (μs)	22.4351	27.8683

Note that the β_{eff} value in a case of a cell infinite medium calculation (without leakage) was validated with MCNP4-C2 [7] and its new endfb6 library including the delayed neutron spectrum. This value ($\beta_{eff} = 715 \pm 20$ pcm) is in good agreement with the APOLLO2's value, under the same non-leakage conditions : $\beta_{eff} = 713$ pcm.

Table 2: Calculation - Experiment comparison on the β_{eff} parameter

Experiment	(C-E)/E (%) pincell calc.	(C-E)/E (%) core calc.
$MISTRAL1 - UOX$	$+3.5 \pm 1.6$	$+2.4 \pm 1.6$

The accurate core calculation is in good agreement with experiment. Calculation seems to slightly overpredict the β_{eff} value for UOX case as shown in Table 2.
The neutron spectrum obtained in Figure 2, corresponds to adjoint flux calculated with 172 energy-group structure by a P_{ij} method in the MISTRAL1 cell case (square pitch 1.32 cm and moderation ratio of about 1.7). This adjoint flux $\phi^{\dagger}$ represents the neutron importance in the cell. One can note hollows in the adjoint flux (which varies in first approximation as $\nu\Sigma_f/\Sigma_a$) corresponding to the U238 capture resonances.

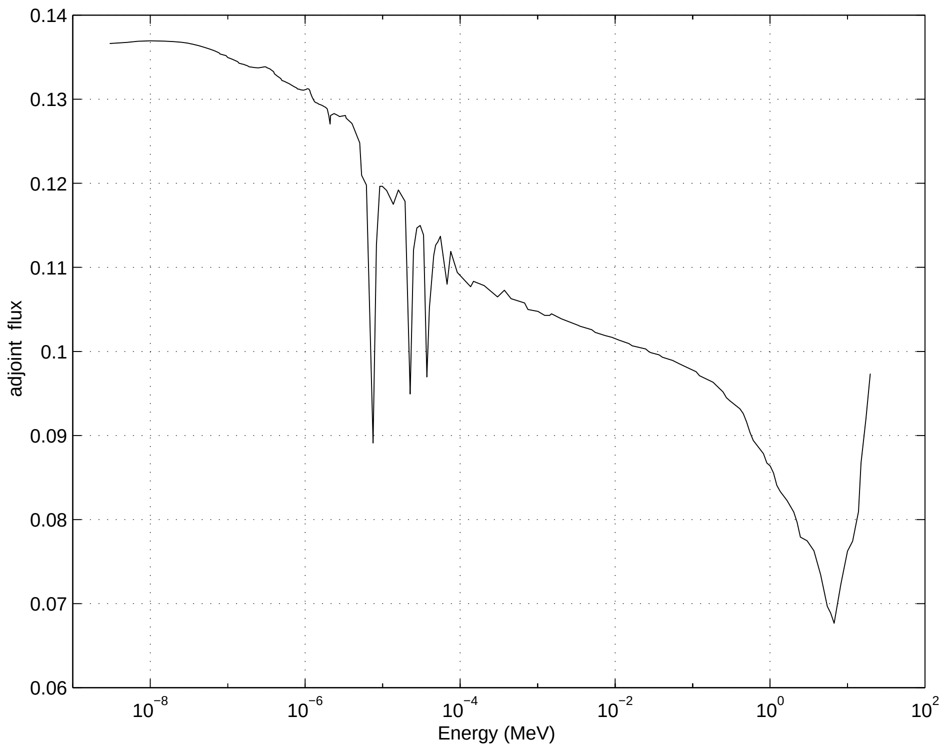

Figure 2: Adjoint flux in UOX 3.7% cell with critical buckling

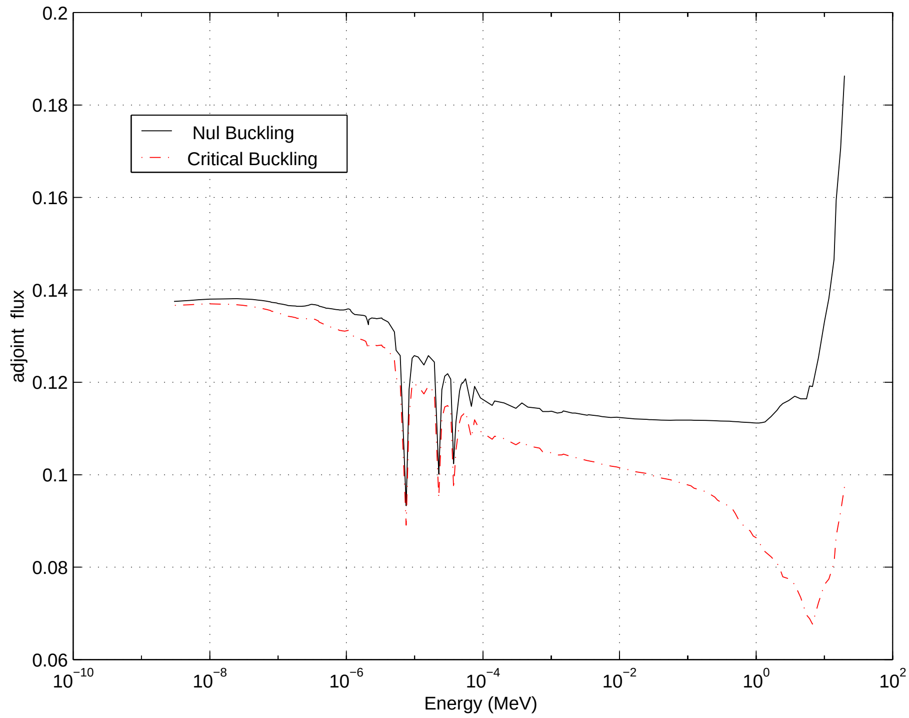

Figure 3: Adjoint flux in UOX 3.7% cell with critical and nul leakage

Figure 3 shows adjoint flux calculated with critical buckling (MISTRAL1 core) and nul buckling (infinite lattice).
The difference between the average energy of delayed neutrons $\overline{E}^d$ ($\simeq$ 500 keV) and prompt neutrons $\overline{E}^p$ ($\simeq$ 2 MeV) plays a particularly important role in finite media (critical buckling) ; delayed neutrons mean free path (λ^d) is lesser than prompt neutrons one (λ^p) and so the leakage probability is lesser in the case of delayed neutrons. Consequently, their efficiency is better (the effective delayed neutron fraction is greater than their real fraction) : $\beta_{eff} = \gamma\beta > \beta$), as one can see on Figure 3. Between average emission energy of prompt and delayed neutrons, the importance increases of about 20% faced with leakage.
With or without leakage, at high energy ($E > 5\ MeV$) one can see an increase of importance function due to, first the increase of U238 fission cross-section, second capture cross-section becomes negligible and third U235 ν factor highly increase ($\Delta\nu/\Delta E \simeq 0.1/MeV$).
Compared to the infinite lattice (nul buckling) where the importance function is almost constant, between 100 eV and 1 MeV (in this situation, efficiency is the same for delayed or prompt neutrons), the calculation with critical leakage shows the two opposed phenomenon, cited before: the importance function decrease for neutrons until 2-3 MeV and the increase after 3 MeV.

5 MISTRAL2 MOX CORE ANALYSIS

Concerning the cell model (with critical leakage) and the MISTRAL2 core calculation, effective delayed neutron fraction for each precursor group, total fraction (β_{eff}) and neutron generation time (Λ) are indicated in Table 3.

Table 3: Delayed neutrons kinetic parameter - MISTRAL2

Group (i)	Cell β_i (pcm)	Core β_i (pcm)
1	11.2954	11.2472
2	79.9436	79.4191
3	67.1794	66.6762
4	133.302	131.622
5	65.0452	63.6834
6	18.4087	18.0269
β_{eff} (pcm)	375.2	370.7
Λ (μs)	9.16615	13.7933

Table 4: Calculation - Experiment comparison on the β_{eff} parameter

Experiment	C-E)/E (%) pincell calc.	(C-E)/E (%) core calc.
$MISTRAL2 - MOX$	$+1.3 \pm 1.6$	$+0.1 \pm 1.6$

The accurate core calculation is in good agreement with experiment. Calculation seems to slightly overpredict the β_{eff} value for MOX case as shown in Table 4.

The β_{eff} value in a case of a cell calculation in infinite medium (without leakage) was validated with MCNP4-C2 and its new ENDF/B6 library including the delayed neutron spectrum (as for UOX core). This value ($\beta_{eff} = 354 \pm 20$ pcm) is consistent within the statistical margin with the APOLLO2's value, under the same non-leakage conditions : $\beta_{eff} = 341$ pcm.

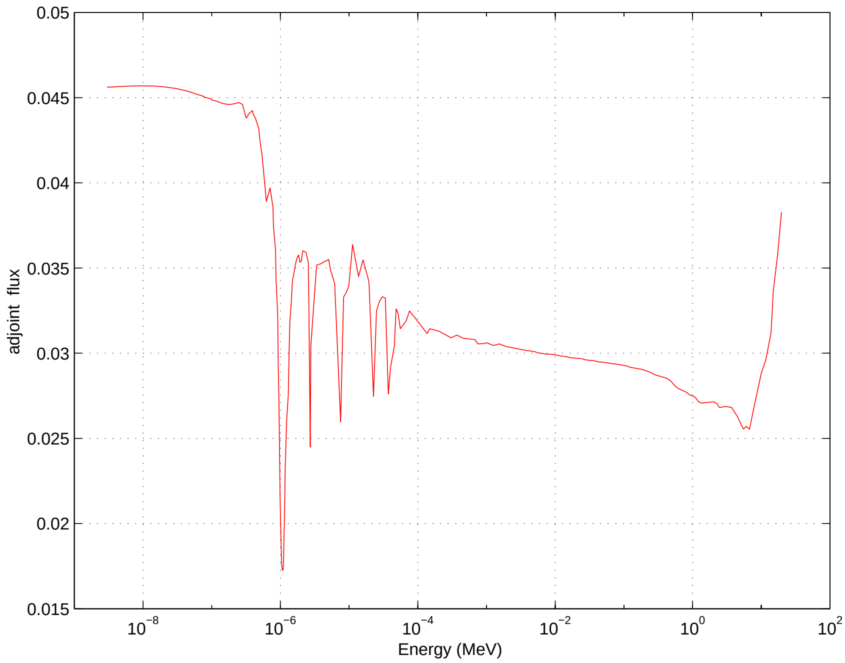

Figure 4: Adjoint flux in a MOX 7% cell with critical buckling

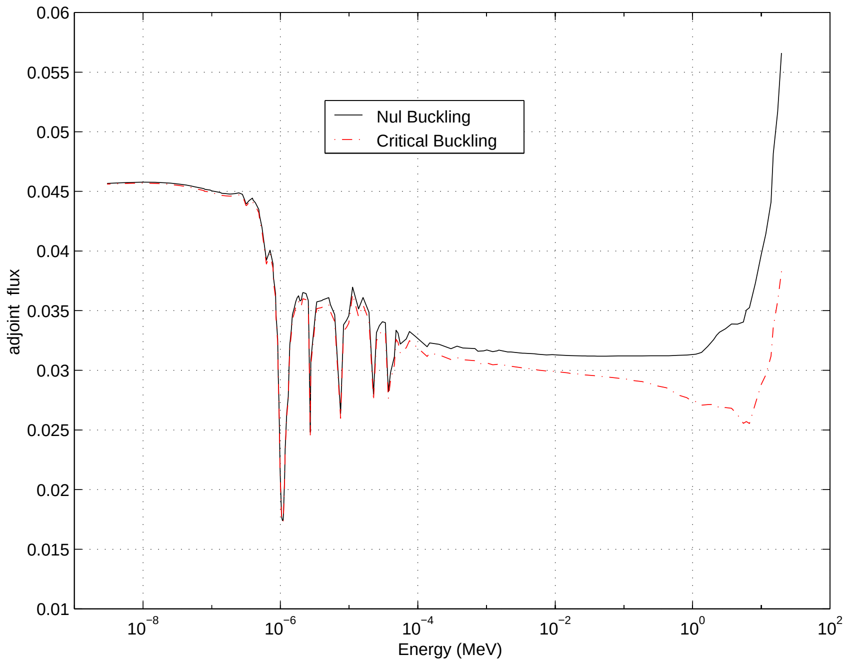

Figure 5: Comparison between adjoint flux in a MOX 7% cell with critical and nul leakage

Adjoint flux, calculated with a critical buckling in the MOX cell is drawn on Figure 4 ; compared to the UOX case, between average emission energy of prompt and delayed neutrons, the importance function increase of about only 5% due to the big size of MISTRAL2-MOX core.
An other hollow appears at 1 eV corresponding to the Pu240 capture cross-section but nothing appears at 0.3 eV for the Pu239 great resonance because the fission process is predominant compared to the capture process.
Contrary to the UOX case (Figure 3), one can not see on Figure 5 (MOX case) a strong difference in the level of the adjoint flux in the intermediate energy range between critical and nul leakage calculations.
One can easily verify that the non-leakage probability in the thermal range $\mathcal{P}^{th}$ of the MISTRAL2 core is greater than in the UOX MISTRAL1 core. If we express this probability in the following manner $1/(1+L^2B^2)$ with $L2$ the diffusion area in thermal range (two-group theory), we have $\mathcal{P}^{th}_{UOX} < \mathcal{P}^{th}_{MOX}$ because $B^2_{UOX} \sim 2B^2_{MOX}$ and $L^2_{UOX} \sim 2L^2_{MOX}$.

6 SENSITIVITIES

The β_{eff} sensitivity to delayed neutron yield of the isotope i is calculated with a 5 energy groups structure (Table 5) commonly used in the previous studies performed on fast reactors [8].
Sensitivities per isotope are reproduced in the Table 6.
In MISTRAL 1, the sensitivity of β_{eff} to U235 is about 0.87 (specially in group 1 from 0 to 11 keV because of the thermal nature of the spectrum) and about 0.13 to U238 (specially for the groups 3 and 4 from 0.5 MeV to 6.7 MeV).
In MISTRAL 2, the β_{eff} sensitivity to the delayed neutron yield of U238 appears in groups 3 and 4 (fast fission threshold at 1 MeV). The β_{eff} sensitivity to U235 is weak, even in group 1 (5%), compared to UOX core (85%). Group 1 almost represents the total of the β_{eff} sensitivity to delayed neutron yield of Pu239.

Table 5: 5 energy groups structure

Group	5	4	3	2	1
Energy (MeV)	19.64	6.7	3.7	0.5	0.011

Concerning the UOX case which was already calculated with an other system code (ERANOS) the results are the same [8], that is to say the sensitivity of β_{eff} to ^{235}U is 0.87 from 0. to 11 keV because of the thermal nature of the spectrum. Same kind of calculations is performed in the MOX case with an enhanced contribution of the $1-6$ MeV range.

Table 6: Sensitivity S_{β_{eff}/ν_i^r}

		1	2	3	4	5	Total
MISTRAL1	$U235$	0.856	0.006	0.007	0.0007	0.0002	0.87
	$U238$	$4E-5$	$4E-5$	0.1	0.03	$8E-3$	0.13
MISTRAL2	$U235$	0.04	0.0008	0.0008	0.0001	$2E-5$	0.04
	$U238$	$1E-4$	$1E-4$	0.22	0.06	0.01	0.29
	$Pu239$	0.47	$6E-3$	0.01	0.001	$1.E-4$	0.49
	$Pu240$	$5E-4$	$2E-4$	0.004	$6.E-4$	$8E-5$	0.01
	$Pu241$	0.168	$2E-3$	$2E-3$	$2E-4$	$2E-5$	0.17

7 CONCLUSION

Even if the six group decay constants incorporated in the JEF2.2 evaluation are not consistent with the relative abundances [6] the results are meeting the required target accuracy of ±3%. The low value of the experimental uncertainty is due to the fact that it is β_{eff}^2 which is determined (with an uncertainty of about ±3%) by the method of frequencies.
The analysis of the MISTRAL integral experiments point out that ^{235}U and ^{239}Pu delayed neutron data are quite satisfactory: the main kinetic parameter β_{eff} is predicted within ±3% in UOX and MOX cores. However the evaluation is now improved in JEFF3 file by a better consistency between weights and periods of the precursor groups.

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