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Full MOX ABWR neutron characterization with void increase: the FUBILA Program

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ABSTRACT

CEA and JNES have undertaken a first-of-a-kind full MOX core physics experiments, FUBILA, in the EOLE critical facility of the CEA Cadarache centre. The experiments have been designed to obtain core physics data of high burn up 9x9 and 10x10 BWR MOX assemblies operating conditions. The experimental program, consisting in eight different core configurations, started in January 2005 and finished on September 1st, 2006. The analysis of some part of the experimental data has been carried out using the French TRIPOLI-4.3 continuous energy Monte Carlo calculation Code with its JEF2.2 and JEFF-3.0 nuclear data libraries, used for all Design and Safety calculations. The average C/E discrepancies obtained enable to estimate all the integral and local parameters with uncertainties largely within the target uncertainties, demonstrating the capability of the code to treat complex geometries with a high degree of accuracy.

Key Words: ABWR, MOX, high burn up, void reactivity effect, power distributions, TRIPOLI, JEF

1. THE FUBILA PROGRAM

1.1 Aim of the Program

On January 6th 2005, the first divergence of the FUBILA [1] reference configuration in the EOLE facility concretised two years of intense preparative work between the Japanese Incorporated Administrative Agency JNES (Japan Nuclear Energy Safety Organisation) and the French Consortium CEA/AREVA-NC, following and complementing two previous successful R&D programs aimed at studying MOX recycling in Advanced LWRs (the MISTRAL [2] and BASALA [3] experiments performed between 1995 and 2002 in the same facility). The last coutings on September 1st, 2006, with the measurement of integral boron worth in the reference configuration, concluded the experimental phase of this first-of-a-kind program.

The FUBILA experimental programme is aimed at investigating the physical phenomena occurring in Japanese Advanced High Burn-Up Boiling Water Reactors fully loaded with MOX fuel. Their characteristics is a strong geometrical heterogeneity within the assembly, with different Pu contents and void fractions, pushing the limits of the available calculation tools and nuclear data libraries.

1.2 Core Configurations

Eight cores were implemented in the EOLE facility of the Cadarache Centre. These cores covered the entire range of specific situations that would be encountered in an HBU 100% MOX BWR core in hot conditions and increasing void fraction, from 0% void to 70% void.

Each configuration is based on a central experimental zone composed of four 9x9 MOX assemblies loaded with 4 types of BWR fuel pins with different amounts of Pu: 3wt%, 5wt%, 8.5wt% and 11.5wt%. The special 10x10 configuration, representing the high burn up loading pattern, is loaded with only 8.5wt% an 11.5wt%, for an average Pu amount of 10.5wt%. The channel box is simulated by a surrounding array of aluminium rods of properly selected diameters (figure 1).

The different configurations and the associated measurements performed therein are resumed in Table I. The criticality of each configuration is obtained by adjusting the number of MOX-7% fuel pins in the peripheral (buffer) zone.

Table I. ***Measurements performed in the different cores of the FUBILA Experimental Program***

Parameters	*Technique*	*REF*	*40%*	*70%*	*Axial*	*10x10*	*B_4C*	*UO_2*	*UGD*
Critical size	-	⊗	⊗	⊗	⊗	⊗	⊗	⊗	⊗
P(*r*)	γ-scanning	⊗	⊗	⊗	⊗	⊗	⊗		⊗
P(*z*)	γ-scanning	⊗	⊗	⊗	⊗	⊗	⊗		⊗
	^{235}U, ^{37}Np				⊗				
Power renormalization	Peak check	⊗				⊗			⊗
Spectral Indices	^{235}U		⊗	⊗					
	^{239}Pu		⊗	⊗					
	^{237}Np		⊗	⊗					
Conversion factor	Peak check		⊗	⊗					
B_4C CB worth	ASM		⊗						
UO_2-Gd_2O_3 worth	ASM							⊗	
UO_2 worth	ASM								⊗
Boron worth	ASM	⊗							
Void worth	ASM	⊗							

The increasing void fraction is simulated by inserting microrods between the fuel pins (as for the 40% "NORM" "UGD" and "B4C" cores – figure 2) or aluminium blocks (in the case of 70% void).

Figure 2 represents the particular mixed loading pattern of the UGD configuration, containing MOX, as UO_2 and UO_2-Gd_2O_3 fuel pins.

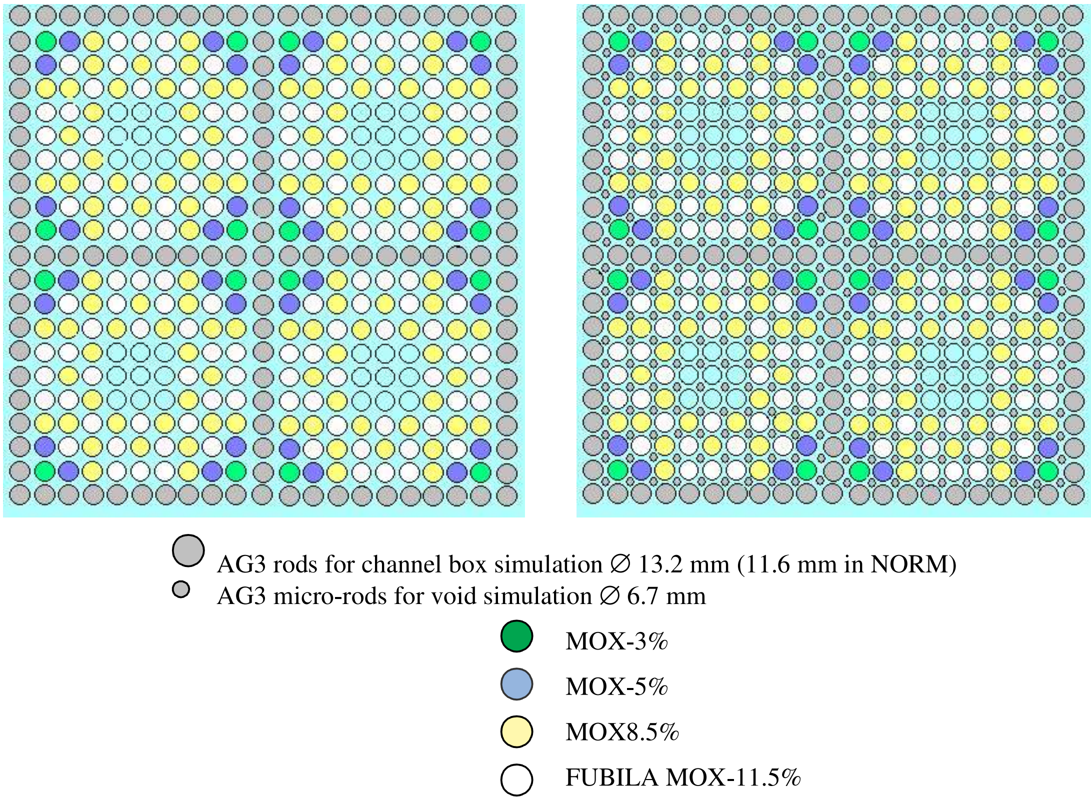

Figure 1. Central experimental zone of REF core (0% void – left) **and NORM core** (40% void – right)

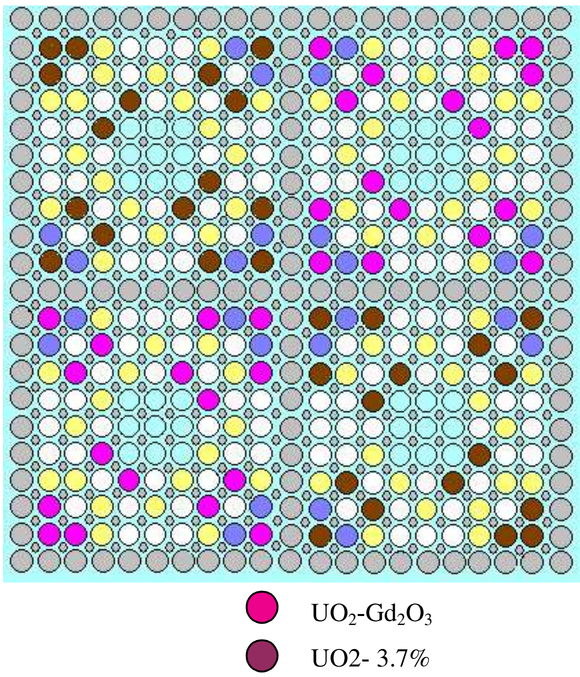

Figure 2. Central experimental zone of UGD core

2. THE EOLE FACILITY

The EOLE facility [4] is a critical facility operating to a maximum power of 100W. It is dedicated to the neutronic studies of moderated lattices like light water reactors (PWRs and BWRs). EOLE is an easily adaptable facility composed of a concrete structure offering biological shielding for flux levels up to 10^9 $n.cm^{-2}.s^{-1}$, a cylindrical vessel in AG3 (Aluminium alloy) with an over structure in stainless steel, able to contain various types of core structures, 4 control rods and a pilot rod, and a water circuit coupled with a thermoregulation station in order to analyze the temperature coefficients of the lattices ranging from 5°C to 90°C.

The EOLE flexibility enables one to analyse the safety parameters of new types of fuels and/or core/assembly designs by using special structures (void boxes, mini-grids for local pitch modification) or by adding various absorbers (single or clusters), as natural or enriched B_4C, Aluminium-Indium-Cadmium (AIC) or Hf. The main cross section of the core is reproduced on figure 3.

Critical (by doubling time measurements) and subcritical measurements (by amplified source method – ASM) are used for the reactivity effects, the reaction rates arising in the lattice being measured by both miniature fission chambers (for fission) and/or γ-scanning techniques (for fission and capture). The facility is being operating since 1965, through major experimental programs in support to the French and Foreign Industrial needs, as well as innovative phases for knowledge improvement of core physics. EOLE has demonstrated its suitability to answer key issues in integral parameters estimation in any kind of water moderated lattice.

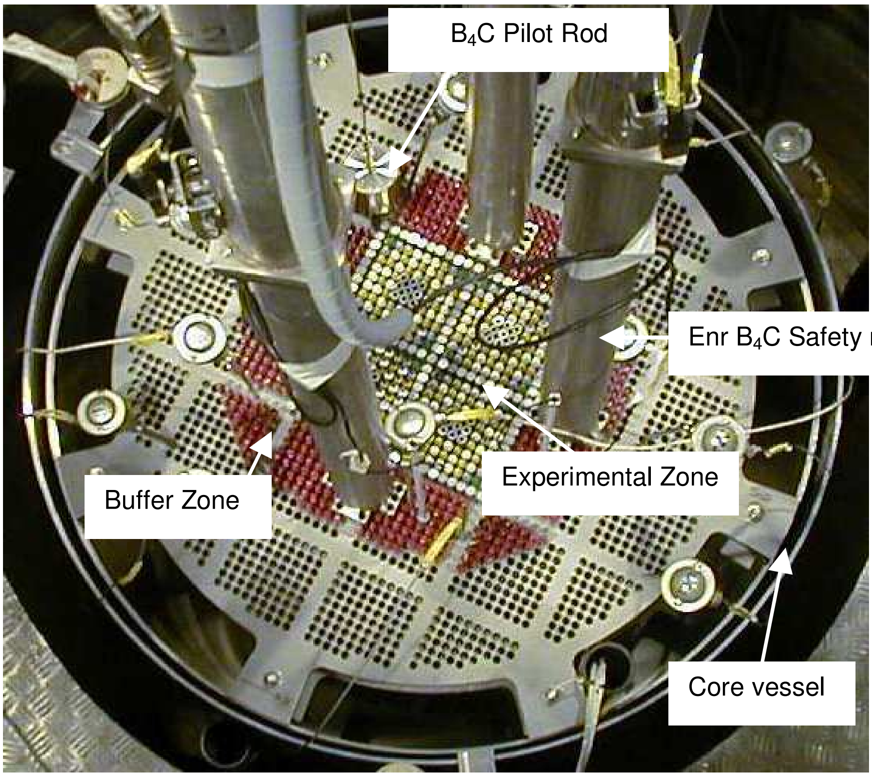

Figure 3. Top view of the EOLE cavity

3. CURRENT AND TARGET ACCURACIES

The current Japanese BWRs are loaded with UO_2 8×8-BWR assemblies whose ^{235}U enrichment is compatible with an average discharge burn-up of about 40 to 50 GWd/t. The goal of FUBILA program is to obtain experimental information allowing to increase this average discharge burn-up up to 65 GWd/t but in a 9×9 MOX fuel, leading to a modification of the assembly loading pattern and to an increase of the MOX initial enrichment. These design options lead to increase the current uncertainties on the main neutronic parameters in such reactors. The following table summarizes the estimated current uncertainties and the target ones [1] for the FUBILA program with CEA calculation tools.

Table II. Target uncertainties on full MOX ABWR lattices after completion of FUBILA program

Neutronic parameter	Current (estimated) uncertainty	Target uncertainty
Multiplication factor	± 1500 pcm (MOX cores)	± 500 pcm
Efficiency of control blades	± 15 %	< ± 7%
Void effect	Up to ± 1500 pcm	< ± 500 pcm
Radial power distribution	± 3 to 4 % (one type of fuel) ± 5 to 7 % UOX + MOX ± 8 to 10 % with Gd	< ± 2% < ± 5% < ± 5%
Axial power distribution	± 3 % in homogeneous axial zones ± 7% in heterogeneous axial zone	± 1 to 2% ± 2 to 3%

4. RESULTS

4.1 Calculation scheme

The calculation scheme is based on a TRIPOLI-4.3 [5,6] Continuous energy Monte Carlo code developed by CEA, associated with its European nuclear data libraries JEF2.2 and JEFF-3.0.
This code is used for Design and Safety calculations of the programs foreseen in the EOLE facility. Each core configuration is made by a full 3D model of the fuel pins and the associated core structures (grids and mini-grids, vessel, etc…). The core sizes are the experimental ones. They are resumed in Table 3. The results are based on unbiased calculations of the configurations on a massively parallel computer. The running time has been put to 24 hours, leading to 80 to 100 millions of neutron histories, depending on the core size modelled.

The statistical uncertainty on the calculated k_{eff} is about 10 pcm, as the uncertainty on the pin-by-pin fission rate is of the order of 1% in most of the situations.

4.2 Experimental Techniques

Various experimental techniques have been applied during the program , leading to major improvements of some methodologies, such as Modified Conversion Ratio measurements [7]. The critical sizes of each configuration were measured many times by doubling time method: the core period is followed on line by miniature fission chambers during the divergence. By using the Nordheim curve and associated nuclear data, the excess reactivity can be deduced with a precision of a few pcm..

4.2.1 Power distributions

The pin-by-pin power distribution is measured by using the so called gross (or integral) γ-scanning method. This technique is based on the measurement of the γ-ray activity of a fuel pin after irradiation. The idea is to rebuild the fission rate arising in the pin by pin distribution in the core. For the same type of fuel (MOX or UOx), the total gamma activity of a fuel pin is directly proportional to the fission rate that occurred in it, because the fission products produced are of the same order of magnitude. So, a measure of the γ activity between 550 keV (above the Compton front) and 2.5 - 3.0 MeV is realized with the acquisition system. Nevertheless, the activity behaviour of the fuel is strongly dependent of the FP's emission during the measurement, and this behaviour is far from being negligible because of the short period of the main fission products. A renormalization on a reference value is needed. This renormalization is based on the use of a monitor pin counted throughout the acquisition of each pin, whose activity variation with time will serve as a reference.

4.2.2 Reactivity effect by subcritical method

The reactivity effect of a perturbation, with this method called ASM (Amplified Source Method), is obtained by subcritical counting associated with the perturbation of fission chambers placed in the periphery of the core (therefore far from the perturbation). These counting are then compared to the counting performed in the reference core, and calibrated on the reactivity of the pilot rod. In a general way, the reactivity effect is calculated in the following manner :

$$\Delta\rho_{ASM} = \left(\frac{\varepsilon_{MSM}}{C_{Pert.}} - \frac{1}{C_{Ref.}}\right)\times RCaFa + \frac{\delta\rho}{\delta T}.\left(T^{Ref} - T^{Pert.}\right) + \frac{\delta\rho}{\delta t}.\left(t^{Ref} - t^{Pert.}\right) + \frac{\delta\rho}{\delta CB}.\left(CB^{Ref} - CB^{Pert.}\right) \tag{1}$$

where: $1/C$ represents the inverse of the counting rates

Ref. and Pert quote the Reference et perturbed conditions respectively

T is the average temperature

t represents the date of the measurement

$\delta\rho/\delta T$ is the differential reactivity temperature coefficient (if necessary)

$\delta\rho/\delta t$ is the Pu aging effect, due to the β decay of ^{241}Pu in ^{241}Am

ε_{MSM} is the MSM Factor (if necessary) between Ref ('0') and Pert ('1') conditions

$$\varepsilon_{MSM}(\vec{r}) \equiv \left(\frac{S_{eff,1}}{<\varphi_1^*, F\Phi_1>} \middle/ \frac{S_{eff,0}}{<\varphi_0^*, F\Phi_0>}\right)\times\frac{R_{c,1}(\vec{r})}{R_{c,0}(\vec{r})} \tag{2}$$

with $S_{eff} = <\varphi^*, S>$ is the effective source importance of the subcritical state

and $R_{c,i}(\vec{r}) = <\Sigma_d, \Phi_i>$ is the reaction rate (response function) of the fission chamber in state "*i*"

This factor, applied to the fission chamber count rate, enables the interpretation of high reactivity coefficient by taking into account the space and energy variation of the spectrum, through the response function of the detector. It has to be calculated *a posteriori*.

Formula (1) necessitates a calibration by introducing in the core an a priori known negative reactivity step (for example the pilot rod which has been previously measured). The *R*eactivity *Ca*libration *Fa*ctor (*RCaFa*) is obtained by using the reactivity value of the pilot rod measured by inverse kinetics (Rod-drop). It is given by relation

$$RCaFa = \Delta\rho^{PilotRod} \Big/ \left(\frac{\varepsilon}{C_{PRd.}} - \frac{1}{C_{PRUp.}} \right) \tag{3}$$

with :

$\Delta\rho_{PilotRod}$ is the Pilot Rod Efficiency
PRUp means *Pilot rod Up*
PRd means *Pilot Rod down*
ε is the MSM factor between the pilot rod up and down conditions, generally taken as unity

The uncertainties are calculated by using the well known propagation rule [8]. In general $\sigma(\Delta\rho_{RodDrop})$ ~ 5.1 % is the dominant term of the uncertainty formula. This methodology enables to reach precision of the order of 5 to 6%, by using specially designed high count rate fission chambers (∅8mm) with important fissile deposit.

4.3 C/E on Critical Masses

Table III shows, for each core configuration of the program, the detailed loading pattern of test and buffer regions, as the C/E on the k_{eff} deduced from doubling time, from the Nordheim curve associated to the core. The REF and 40% patterns are reproduced in figure 2, the UGD on figure 3.
As already mentioned, the increasing void effect is modelled by inserting, between the fuel pins, aluminium microrods to reduce the water content within the experimental zone. In the case of 70% void, the 4 central assemblies are inserted in Aluminium blocks whose height is 1,2 m. The particular case of the half void configuration consists in an upper half part identical to the REF situation (0%void), as the lower half part of the core is identical to the 70%void situation, by using 4 half aluminium blocks. This special configuration is specially devoted to the 0% - 70%void interface simulation by using 3D codes.

Table III. Cores critical conditions

Config	REF	40% void	70% void	Axial void	10x10	B_4C	UGD
(Test region)					-		
MOX 3%	16(4x4)	16	16	16	-	16	-
MOX 5%	32(4x8)	32	32	32	112(4x28)	32	24
MOX 8.5%	112(4x28)	112	112	112	256(4x64)	112	72
MOX 11.5%	128(4x32)	128	128	128	32(4x8)	128	128
UO_2 3.7%	-	-	-	-	-	-	32
$UO_2.Gd_2O_3$	-	-	-	-	-	-	32
Guide tube	36(4x9)	36	36	36	36	36	36
AG3 µrod	-	400(4x100)	-	-	129	400	400
AG3 block	-	-	4	4 (half)	-	-	-
AG3 rod	117	117	117	117	617	117	117
						870	870
(Driver region)							
MOX 7%	598	894	1130	878	618	1474	1226
Reactivity (pcm)	55	57	79	55	40	75	67
Calc. JEF-2.2	1.00109	1.00188	1.00347	1.00259	1.00318	1.00280	1.00259
Calc. JEFF-3.0	1.00022	1.00119	1.00259	1.00181	1.00324	1.00247	1.00152
C/E JEF2.2	54	131	267	203	277	204	191
C/E JEFF-3.0	-33	62	179	126	283	171	85

On an average, the calculations predicts with a very good accuracy the critical masses of the various configurations. The TRIPOLI-4.3 and its libraries gives an average

$$\text{C/E (JEF-2.2)} = (190 \pm 77)\ \text{pcm}$$
$$\text{C/E (JEFF-3.0)} = (125 \pm 101)\ \text{pcm}$$

These values remain largely within uncertainty targets detailed in Table II.

4.4 Reactivity Worth

The reactivity worth of increasing void fractions, from 0% to 70% have been measured by subcritical method in the REF core. The void fraction is simulated as explained in §I.2. The core size is maintained constant in all void situations, the reactivity effect is obtained by combining the count rates of 3 fission chambers positioned in the core and in the moderator.
Complimentary measurements of the natural B_4C control blade in the NORM core have been performed by ASM.
The reactivity effect of the 2x16 poisoned pins and 2x16 UO2 fuel pins in the case of UGD core have been measured by subcritical method.

The C/E results on the reactivity worth, obtained with JEF-2.2 library, are resumed in table IV here below:

Table IV. C/E on ASM reactivity worth

Reactivity worth	C/E	Uncert (2σ)
40% VOID in REF(*)	1.06	14.7%
70% VOID in REF(*)	1.18	14.7%
Axial Void in REF(*)	1.12	15.0%
B_4C in NORM	1.07	15.6%
2x16 UO2-Gd2O3 in UO2	1.00	15.0%
2x16 MOX in UGD	1.02	13.1%

(*): corrected from calculated peripheral pin worth

An over prediction of the void effect appears in the calculation. This effect can be associated to the void behaviour (the cross sections of Aluminium are suspected) because the other reactivity effects are very well calculated. The special analysis of the 0%/70% interface in the case of Half void configuration, that has been axially measured with miniature ^{235}U and ^{237}Np miniature fission chambers will be of great interest to identify the possible cause of the discrepancy.

4.5 Power distributions

Axial and radial power distribution, by using gross γ-scanning of particular peak check method [9] directly on the fuel pins have been used to determine precisely the fission rate distribution in the core mid-plane. The calculated fission rate distributions on the various cores analyzed up to now (REF, NORM, 70%VOID, 10x10 and UGD) are reproduced on figures 2 to 9. The C/E together with its C/E uncertainty (experiment & calculation) is given in tables V to IX here below. We give only the results linked to the voided cores.

Table V. Pin-by-pin C/E for REF Core

1.03	1.01	0.98	0.99	0.99	1.00	0.99	0.99	1.02
1.02	0.99	1.02	1.00	0.99	1.00	1.02	0.98	1.00
1.00	0.99	1.01	1.02	0.99	1.01	0.95	0.98	0.97
1.00	1.01	1.03				0.98	0.99	1.00
1.01	0.99	0.98				0.99	0.99	0.99
1.02	1.00	0.99				1.02	1.00	1.01
1.03	1.03	1.02	1.00	1.00	1.01	1.01	1.00	1.00
1.04	1.03	1.03	1.03	1.01	1.00	1.01	0.97	1.01
1.02	1.02	1.02	1.01	1.00	0.97	1.01	1.00	1.01

Average C/E = 1.00 ± 1.7% Av. uncertainty = 1.7% (1σ)

Table VI. Pin-by-pin C/E for NORM Core

1.00	1.03	1.00	0.99	0.99	1.01	0.98	0.99	0.99
1.01	0.98	0.98	1.00	1.00	0.98	1.01	0.98	1.00
0.99	0.99	0.98	1.01	1.02	1.00	0.97	0.97	0.96
1.01	0.98	0.98				1.01	0.98	0.98
1.01	1.01	0.98				1.00	1.00	0.99
1.01	1.02	1.01				0.98	0.98	1.00
0.97	0.99	1.03	1.02	1.01	1.02	1.02	0.99	0.96
1.06	1.01	1.03	1.02	1.02	1.02	0.97	0.98	1.00
1.03	1.03	1.04	0.99	0.98	1.04	0.99	1.04	1.01

Average C/E = 1.00 ± 2.1% Av. uncertainty = 2.1%(1σ)

Table VII. Pin-by-pin C/E for 70% VOID Core

1.02	1.00	1.03	1.04	1.03	1.00	0.99	1.02	1.02
0.98	1.00	1.02	1.01	1.00	1.02	0.99	0.97	1.02
0.99	1.00	0.99	1.03	1.00	1.02	0.97	0.99	1.02
1.06	0.98	1.00				1.00	1.00	0.99
1.02	1.05	1.01				1.00	0.98	1.03
1.01	1.00	0.99				1.00	1.00	1.00
1.01	1.00	1.05	0.96	1.01	1.02	1.00	0.99	1.04
1.03	1.01	1.01	0.98	0.99	0.99	1.02	1.02	0.98
1.03	1.01	1.00	1.04	0.99	1.02	1.02	1.01	0.99

Average C/E = 1.01 ± 1.9% Av. uncertainty = 2.1%(1σ)

Table VIII. Pin-by-pin C/E for 10x10 core

0.99	1.00	0.97	0.97	0.98	1.00	1.00	0.94	0.99	0.98
0.99	1.01	1.00	1.00	0.99	1.00	0.97	0.97	0.99	0.98
0.94	0.98	0.97	0.98	0.97	1.01	1.00	0.98	0.96	0.99
0.99	0.97	0.96			0.98	0.97	0.99	0.96	1.00
0.97	0.97	0.98			0.97	0.95	0.96	0.97	1.00
1.00	0.98	0.96	0.97	0.99			0.97	0.98	0.98
0.98	0.99	0.96	0.96	1.01			1.00	1.00	0.99
0.99	0.97	0.97	1.00	0.97	0.98	0.94	0.94	0.96	1.01
1.00	0.98	1.00	0.98	0.99	0.98	0.97	0.97	0.98	0.97
1.01	1.00	0.97	1.02	1.00	0.98	0.98	0.98	1.02	0.98

Average C/E = 0.98 ± 1.7% Av. uncertainty =2.0%(1σ)

Table IX. Pin-by-pin C/E for UGD Core

1.01	1.01							0.99									0.97
						1.00										1.01	0.98
							0.98			1.01					1.02		
						1.00											
								0.98			0.99						
		0.99					0.98			0.98					0.99		
								1.00	0.99		0.99						1.02
1.03	0.99	1.03	1.02	1.01	1.02	1.02	1.00	1.01	0.98	1.01	0.99	1.01	1.01	1.00	1.01	1.00	0.97
1.03	0.98	1.02	1.02	0.99	1.00	1.02	0.99	1.00	1.02	1.02	1.00	1.02	0.97	0.99	1.06	1.01	1.01
1.00	1.01	1.02	1.03	1.01	1.02	1.03	0.97	1.02	1.00	1.00	1.02	0.96	1.01	0.99	1.00	1.02	1.00
1.00	1.05	1.04				1.00	1.02	1.01	1.01	1.01	0.98				0.97	1.00	1.02
1.02	0.99	1.02				1.01	0.98	0.97	0.96	1.01	1.01				1.02	0.99	1.01
1.00	1.01	1.03				1.01	0.98	1.01	1.01	1.01	1.00				1.03	1.01	0.98
0.99	1.01	0.98	1.01	1.01	1.02	1.02	1.01	1.02	1.03	0.98	0.99	1.03	0.99	0.97	1.01	0.99	0.99
1.00	1.04	1.04	1.00	1.01	1.01	0.99	0.99	1.04	1.01	0.99	1.03	0.99	0.97	1.01	1.02	1.00	0.98
0.99	1.00	1.03	1.01	0.98	0.96	0.97	1.01	1.00	0.96	0.99	1.00	0.99	1.00	1.00	1.01	0.97	1.04

Average C/E = 1.00 ± 1.9% Av. uncertainty = 1.7%(1σ)

Average MOX = 0.99 ± 1.7%

Average UGD 1.00 ± 1.6%

On an average, the TRIPOLI-4.3 code and its JEF-2.2 library is able to reproduce the high heterogeneous pin-by-pin distribution within 1 to 2% on an average, with a dispersion of about 2.0%, including both experimental uncertainty on the fission rate and the statistical uncertainty associated to the Monte Carlo calculation.

In any case, these values cope with the target uncertainties given in Table II.

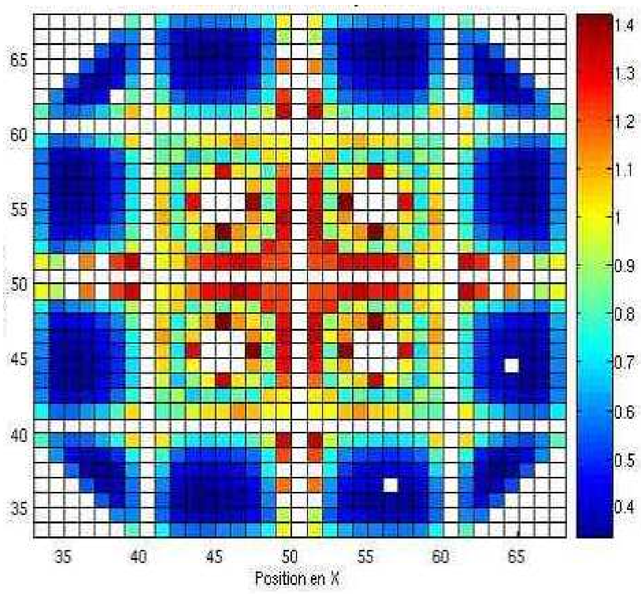

Figure 4. FUBILA/REF radial power distribution

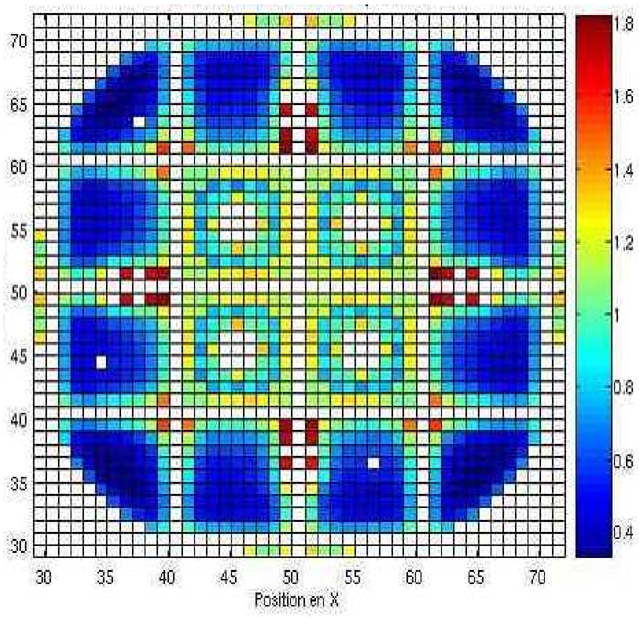

Figure 5. FUBILA/NORM radial power distribution

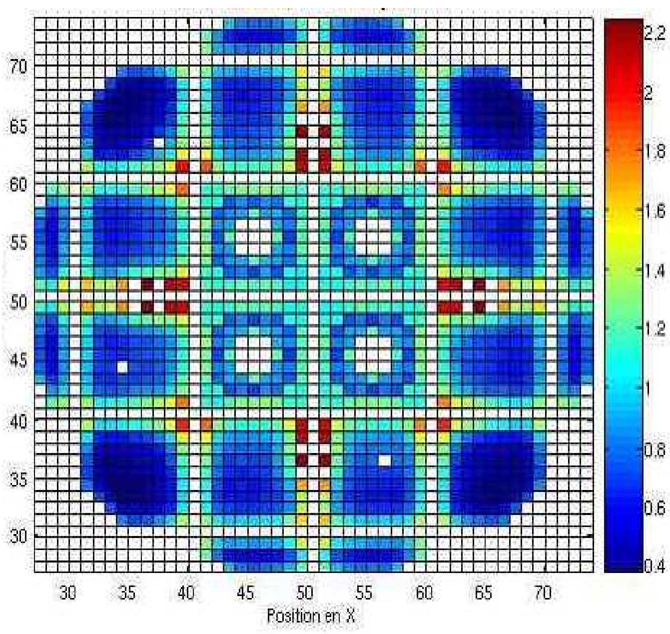

Figure 6. FUBILA/70%VOID radial power distribution

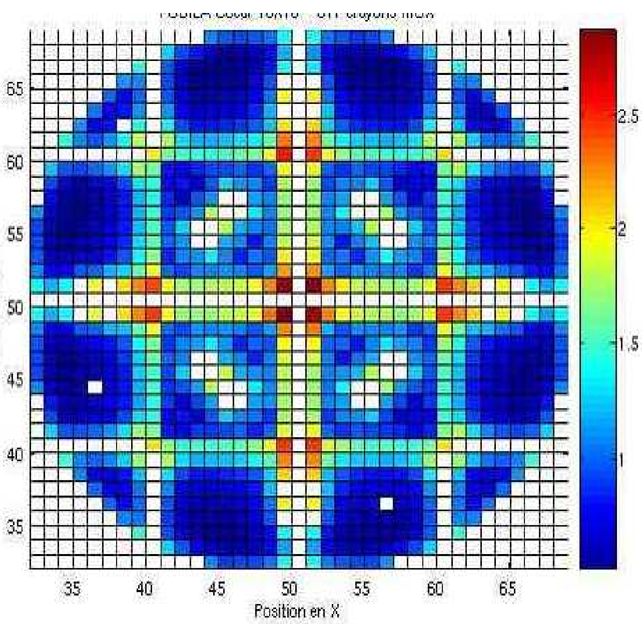

Figure 7. FUBILA/10x10 radial power distribution

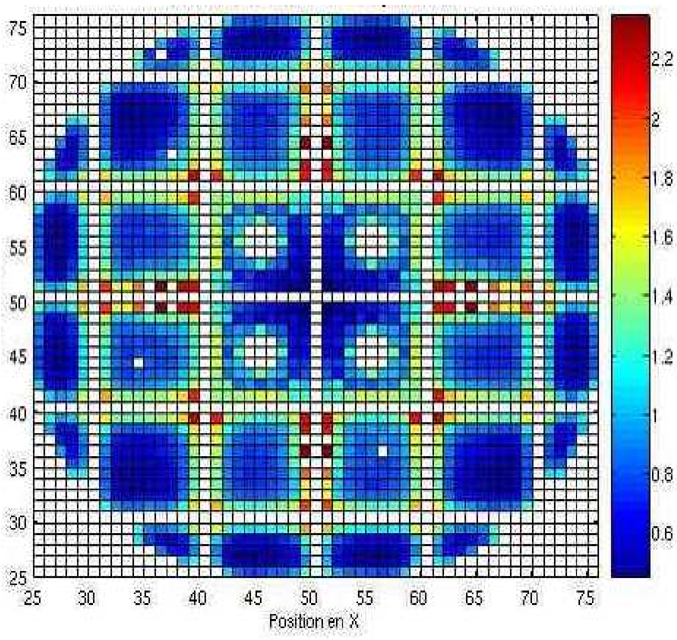

Figure 8. FUBILA/B4C radial power distribution

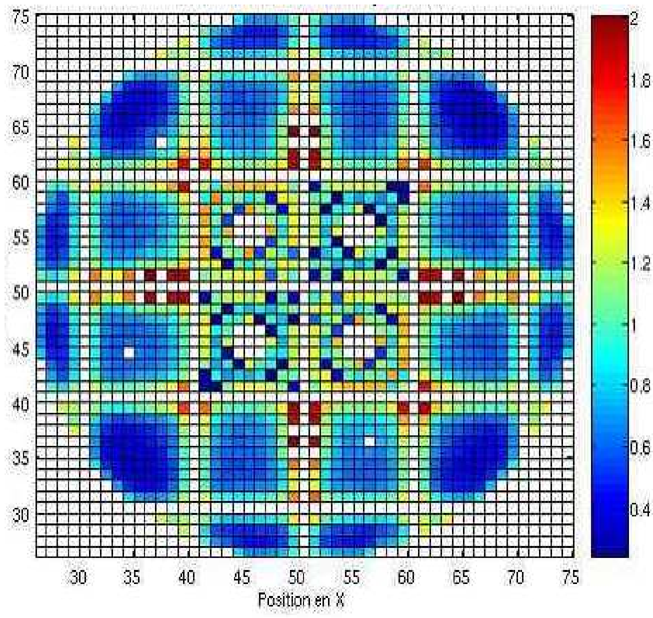

Figure 9. FUBILA/UGD radial power distribution

5. CONCLUSIONS

FUBILA Program launched in close collaboration between the French Commissariat à l'Energie Atomique (CEA) and the Japan Nuclear Energy Safety Organization (JNES) in the French EOLE critical Facility is devoted to the analysis of neutron behaviour and characteristics of advanced boiling water reactor concepts, in 9x9 or 10x10 geometries, of high burn up 100% MOX, with void fraction increase (from 0% to 70%).
The detailed analysis of an important part of the experimental results has been presented in the paper. It consists in the C/E comparison on the critical masses, as reactivity effects of increasing void fraction in the reference 0% void configuration, as radial power distribution within the ABWR assemblies. Calculations have been performed by using the continuous energy Monte Carlo TRIPOLI-4.3 and its associated JEF2.2 and JEFF-3.0 libraries.

For critical masses, the average reactivity C/E obtained for the cores is less than 200 pcm.

For reactivity worths, the reactivity effects of absorbers are very well predicted, the C/E of a few percents. Nevertheless, a discrepancy remains for the increasing void fraction effect, for which the C/E is situated outside the 2σ in the case of 70% void. The analysis of the axial fission chamber traverses in the axial void core will bring additional information on this discrepancy.
The pin-by-pin power distributions are calculated within 2% on an average in these heterogeneous lattices, coping with the target accuracies.

In any case, the C/E discrepancies obtained with TRIPOLI-4.3 and JEF-2.2 enable to estimate all the integral and local parameters with uncertainties largely within the target uncertainties, demonstrating the capability of the code to treat complex geometries with a high degree of accuracy.

The future TRIPOLI-4 interpretation work will concern the detailed analysis of the axial void core and the integral boron worth in the reference core, by using the so called MSM factors defined by eqn (2). The previous calculations will be resumed with the JEFF-3.1 release of the TRIPOLI Library.

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