Experimental Validation of the LWR Reactivity Loss with Burn up: Analysis of Spent Fuel Oscillation Experiments

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Abstract

The reactivity loss with fuel depletion is linked to a major parameter in PWR fuel management: the cycle length. This paper demonstrates that APOLLO2 lattice code and JEFF associated nuclear data are able to predict this main neutronics parameter. Spent fuel worth measurements of high burn-up UOx (up to 70GWd/t) and MOx pins (up to 60GWd/t) were performed in the MINERVE facility using an oscillation technique. The analysis shows slight discrepancies between calculated and experimental reactivity loss with depletion. These discrepancies are reduced using the latest European Nuclear Data File JEFF-3.1. Calculation-Experiment biases are summarized below:

	JEF-2.2 Results:	JEFF-3.1 Results:
UOx:	+2.3% ± 1.3%	+0.6% ± 1.3%(1σ)
MOx:	-1.0% ± 1.5%	+1.5% ± 1.5%(1σ)

KEYWORDS: LWR, Minerve, Reactivity Loss, Depletion, APOLLO2, JEFF

1. Introduction

A major industrial target in the near future is to reach fuel burnups up to 70GWd/t for UOx and up to 60GWd/t for MOx. In this industrial context, it must be demonstrate that fuel lattice codes and data are able to predict the main neutronics parameters within target accuracy. Hence, new experiments were carried out in order to extend the current experimental validation, particularly fuel depletion (Actinide-FP build-up and reactivity loss) to higher burnups in UOx and MOx fuels. The aim of this paper is to analyze the measurements of the reactivity loss of high burnup spent fuels performed in the MINERVE facility. The analysis is done using the APOLLO2 code [1,2] and its associated nuclear data libraries based on JEF-2.2 and JEFF-3.1 European files [3].

The first part of this paper will be devoted to a brief description of the oscillation technique achieved in the MINERVE reactor [4]. The second part will describe the methodology to interpret the oscillation experiment, such as the calculation scheme and the calibration factor between calculated and experimental results. The last part will show experimental validation results for UOx and MOx high burnup spent fuels.

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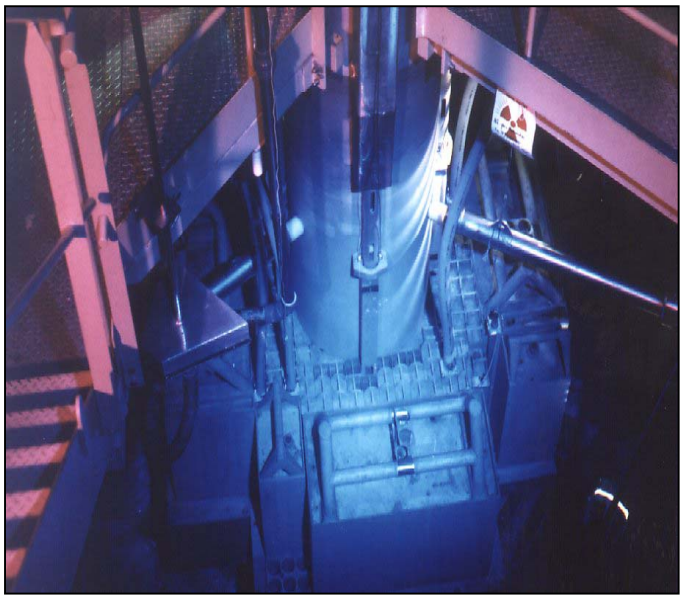
2. Experimental Settings

2.1 The MINERVE Facility



The experimental reactor MINERVE is devoted to neutronic studies of different reactor types. MINERVE achieved its first criticality in 1959 at the center of Fontenay-Aux-Roses and was transferred to CEA-Cadarache in 1977. MINERVE is a zero-power (<100W) pool reactor. The core is submerged under 3 meters of water and is used as a driver zone for the different experiments located in a central cavity. Several lattices corresponding to different neutron spectra can be loaded in this cavity, such as an over-moderated spectrum, two LWR spectra (UO_2 or MOX fuel), two intermediate spectra (MORGANE/S and R blocks for HCLWR studies) and a fast spectrum.

2.2 The Experimental Technique

The technique consists in oscillating spent fuel samples (PWR rod cuts) at the center of the experimental PWR-type lattice, in order to measure the associated reactivity worth with an accuracy better than 1% (1σ). The flux variation induced by the oscillation is detected in the driver zone by a boron chamber, which is servo-driven to a rotary automatic pilot rod (using the overlap of cadmium sectors). The experimental value corresponds to the angle of the rotor/stator reached by the pilot rod for the neutronics equilibrium and is linear versus the reactivity worth of the oscillation sample. The experimental unit is the Pilot Units (PU).

2.3 The Experimental Program

The so-called “High Burn up” experimental program is devoted to radiochemical assays of spent fuel rods cuts and to oscillations of the corresponding spent samples. This paper will describe the interpretation of the oscillation experiment of both UOx and MOx rod cuts respectively up to 70GWd/t (6 irradiation cycles) and 60GWd/t (5 irradiation cycles). Oscillated samples are made of a 10 cm PWR rod cut under a double clad. Rod cuts originates from various fuel pins (mid-height) of UOx and MOx 17x17 assemblies (in this paper, from CRUAS-II and DAMPIERRE-II 900Mwe respectively).

UOx samples are oscillated at the center of the PWR-type R1-UO_2 lattice (800 UO_2 3%^{235}U pins), while MOx samples are oscillated in the R1-MOX 4%Pu experimental lattice.

3. Methodology of Interpretation using the APOLLO2 code

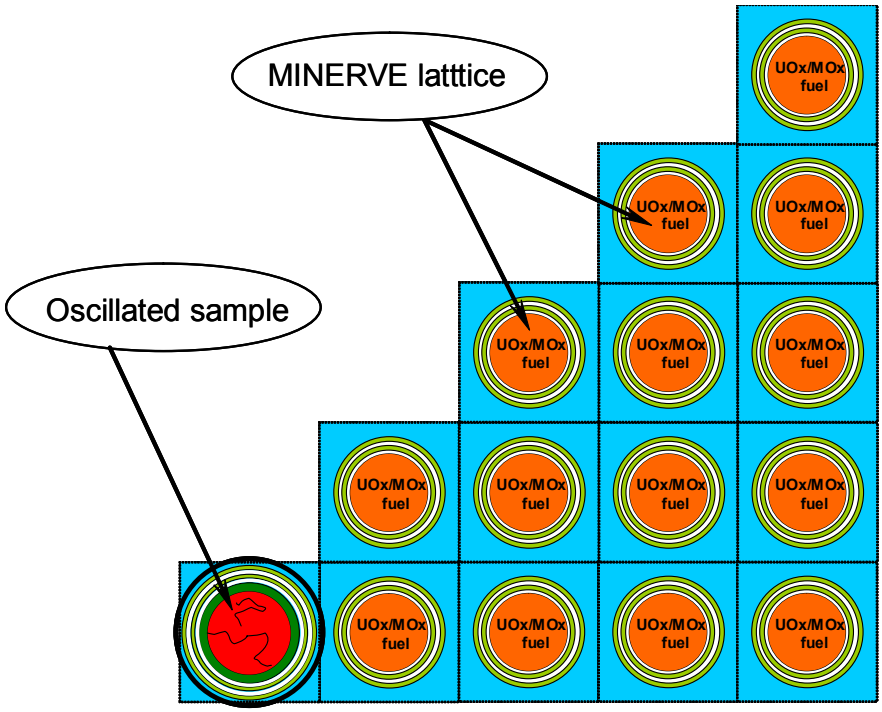
The interpretation is based on the French deterministic transport code APOLLO2.

Reactivity worth is obtained using the exact perturbation theory : $\delta\rho = -\frac{<\phi^*, \delta H \phi'>}{<\phi^*, F'\phi'>}$

3.1 Calculation Route

The APOLLO2 code is a modular code, which solves both the Boltzmann integral equation and the integro-differential equation. The current APOLLO2.5 version uses several nuclear data libraries, which are processed from the JEF-2.2 and JEFF-3.1 files in the XMAS-172 or SHEM-281 group-structure. The APOLLO2 reference calculation "CEA-97" is defined by selecting the code options that yield known and acceptable errors [5] (consistent with target accuracy). The analysis of the OSMOSE experiment allows a limited multicell geometry: a 2D 11x11 cells pattern is used (Figure 1). The pattern is a regular UO_2 (or MOx) lattice for the central experimental device including the central oscillation tube and sample.

Figure 1: APOLLO2 Calculation Pattern for the Experiment Analysis



APOLLO2 calculation uses P_{ij}-UP1 model (interface current method with linear anisotropic angular fluxes). The space-dependent (4 concentric rings in every fuel pin description) self-shielding calculation for resonant isotopes is performed using the Probability Table method. Modeling errors were assessed [6] by comparison with the reference continuous-energy Monte-Carlo calculation TRIPOLI4 [7].

In order to compare the calculated value (in 10^{-5}=pcm) to the experimental value (Pilot Units) of a sample worth, we have to calibrate the calculated signal against the experimental one.

3.2 Calibration Factor between Calculation and Experimental Signals

The calibration factor α between a calculated reactivity worth (ρ_c) and a measured reactivity worth (ρ_m) is defined as following: $\rho_m = \alpha.\rho_c + \beta$. The determination of the calibration factor α is achieved through the reactivity worth measurement of ^{235}U and ^{10}B (which assumes that nuclear data are accurately known for these two isotopes).

The calibration factor is carried out from the measured and calculated reactivity worth of the following sample oscillations:

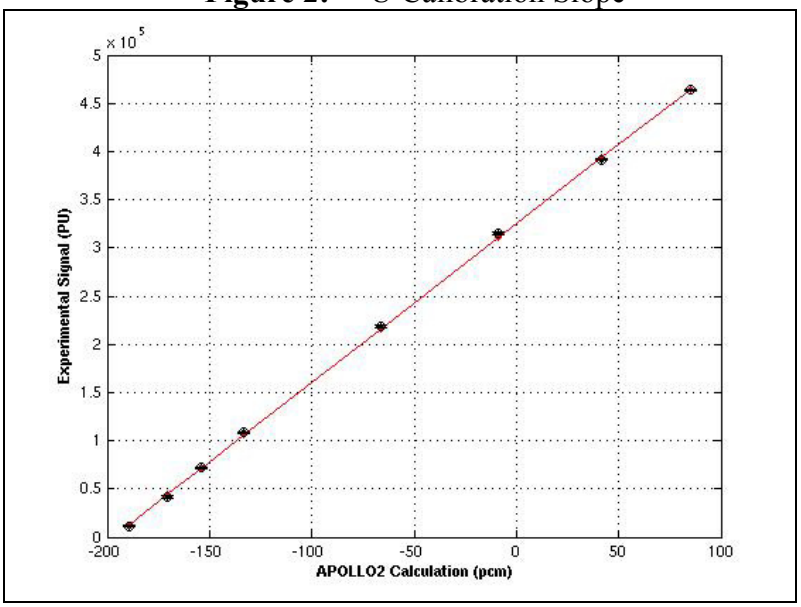
- 8 UO_2 samples with increasing enrichment (0.2%^{235}U up to 5%) which enable the signal calibration through ^{235}U cross-sections,
- 3 UO_2 samples (with a fixed ^{235}U enrichment of 0.25%) with increasing boron content (B/U from 0 up to $420x10^{-6}$), which allow signal calibration through ^{10}B cross-sections,
- 3 UO_2 samples (fixed ^{235}U enrichment of 0.53%) with increasing boron concentration (B/U: from 0 up to $1060x10^{-6}$), which allows α calibration through boron worth.

Then, three calibration factors ($\alpha_i \pm \delta\alpha_i$) are obtained using least-square linear fitting, one for ^{235}U (fissile positive signal) and two for ^{10}B (negative absorbent signal). The overall uncertainty of each calibration factor is the quadratic combination of:

- the experimental uncertainty (cf. [4] [8] for details),
- the least-square linear regression uncertainty,
- the uncertainty on the material balance of the calibration samples
- the uncertainty of the ^{235}U and ^{10}B nuclear data (assessed from the difference between JEFF-3.1/JEF-2.2 evaluations).

As shown in Figure 2, the calibration slope using ^{235}U samples (and the XMAS group library based on JEF-2.2) gives in the UO_2 lattice: $\alpha_{235U} = 1720 \pm 28(1\sigma)$ PU/pcm.

Figure 2: ^{235}U Calibration Slope



3.3 Length Correction for Boron calibration samples

The APOLLO2 bi-dimensional calculation model does not take into account different axial geometries of the samples. The external length of all the samples is the same (aluminum container H = 103.5 mm), however the fissile length L_s is sometimes higher (boron calibration samples) than a reference length value L_R=9.4cm which corresponds to the length of PWR fuel samples. Then, a correction called Length Correction (C_L) is applied to the experimental value :

$$\rho'_m = \rho_m \times C_L$$

with

$$C_L = \frac{\int_{V_s} \Phi(\vec{r},z)\Phi^+(\vec{r},z)d\tau}{\int_{V_R} \Phi(\vec{r},z)\Phi^+(\vec{r},z)d\tau}$$

$$and \qquad \int_V \Phi(\vec{r},z)\Phi^+(\vec{r},z)d\tau \quad \propto \quad \int_{-L/2}^{+L/2} cos^2(B_z.z)dz$$

$$then \qquad C_L \quad = \quad \frac{\left[cos(B_z.L_s)sin(B_z.L_s) + B_z L_s\right]}{\left[cos(B_z.L_R)sin(B_z.L_R) + B_z L_R\right]}$$

using ϕ,ϕ^+ respectively the nominal direct and nominal adjoint fluxes and B_z^2=0.193x10-2cm^{-2}. For boron calibration samples, C_L=1.06. The uncertainty due to this length correction is negligible.

Finally, the calibration factor averaged on ^{235}U and Boron samples is:
$<\alpha>$ = 1703 ± 22 PU/pcm.

3.3 Material Balance of High burn-up Samples

The fuel inventory of spent fuel samples is obtained from JEF-2.2 and JEFF-3.1 libraries, using the APOLLO2 CEA-97 scheme : self-shielding calculations at successive depletion steps, actual irradiation cycle follow-up (fuel pin displacements, time cooling, assembly specific power modifications, temperature modifications of fuel-clad-moderator, boron concentration,…). The depletion chains involve 77 fission products for the library based on JEF-2.2 and 93 FPs for the JEFF-3.1 library. The neodymium concentration determines the actual burn-up of the fuel rod cut. Moreover, the APOLLO2 fuel inventory is validated versus radiochemical assays within few percents accuracy, particularly for main actinides and fission product contents [9].

The Fission Gas Release in such spent fuel has to be taken into account in the calculated material balance of oscillated samples. It involves almost 4% of volatile isotopes (Xenon, Krypton, Iodine and Cesium) at 60GWd/t and roughly 6% at 70GWd/t for UOx fuels and almost 6% at 50GWd/t and 8% at 60GWd/t for MOx fuels [10].

4 Qualification Results on Burnup Reactivity Loss

The difference between the PWR spent fuel sample signal and the fresh sample signal (UO_2 sample or MOx sample) matches the burnup reactivity loss in corresponding spectra (UO_2 or MOx assembly).

4.1 Flux spectrum in the Minerve PWR-type lattice spectrum

Preliminary calculations of spectral indexes in the various lattices are performed and compared to experimental values in order to ensure the flux calculation. The comparison of the ^{238}U conversion ratio in the homogeneous lattice between APOLLO2/JEF-2.2 calculation and experiment (γ-spectrometry) shows a good agreement by about [C/E-1]±[δE/E]=+1.8%±1.4% (1σ) in the UO_2 lattice. This discrepancy is in agreement with previous qualification results, particularly in the MISTRAL program [11].

4.2 Reactivity Loss Qualification Results using the JEF-2.2 library

Table 1 shows the Calculation-Experiment biases on the reactivity loss versus burn-up for various UOx depleted fuel rod cuts. Similar results in a MOX assembly are reported in Table 2.

Table 1: UOx fuel results using the JEF-2.2 Library

UOx irradiated samples		JEF-2.2 Results	
Fuel rod location in assembly	^{235}U initial enrichement (w/o)	Burn-up	[C/E-1] ± [δE/E]
D02	4.5%	57GWd/t	+ 1.5% ± 1.3%
P02	4.5%	69GWd/t	+ 2.9% ± 1.3%
B16	4.5%	69GWd/t	+ 1.9% ± 1.3%
		Mean Value for UOx fuels:	+ 2.3% ± 1.3%

Table 2: MOx fuel results using the JEF-2.2 Library

Mox irradiated samples		JEF-2.2 Results	
Fuel rod location in assembly	Pu initial content (w/o)	Burn-up	[C/E-1] ± [δE/E]
P16	5.3%	52GWd/t	- 1.8% ± 1.6%
E09	6.7%	53GWd/t	- 1.0% ± 1.8%
C15	5.3%	60GWd/t	- 1.5% ± 1.9%
H10	6.7%	58GWd/t	+ 1.9% ± 2.0%
		Mean Value for MOx fuels:	- 1.0% ± 1.5%

Such discrepancies fit with an underestimation (UOx) or an over-estimation (MOx) of the fuel reactivity at the end of cycle by about -700pcm and +300pcm respectively.

The (δE/E) uncertainty takes into account:

- the statistical experimental uncertainty of the difference between the irradiated sample signal and the fresh fuel sample signal,
- the statistical experimental uncertainties of the sample material balance (~1%),
- the systematic uncertainty due to experimental worth signal calibration (1.3%).

4.3 Reactivity Loss Qualification Results using the JEFF-3.1 library

The material balance of irradiated samples is computed using the JEFF-3.1 library (taking into account the irradiation cycle follow-up).

The mean values of the C/E biases are summarized in the Table 3:

Table 3: UOx and MOx fuel results using the JEFF-3.1 Library

	[C/E-1] ± [δE/E]
Mean Value for UOx fuels:	+ 0.6% ± 1.3%
Mean Value for MOx fuels:	+ 1.5% ± 1.5%

Clear improvements are observed on the prediction of the reactivity loss of UOx fuels using JEFF-3.1 files, thanks to the cross-sections, fission yields, and decay data modifications. The prediction of the reactivity loss of UOx and MOx fuels turns into a slight overestimation, but is still in the experimental accuracy (in one standard deviation).

We can notice that this overestimation of the reactivity loss is cancelled using the new SHEM fine multigroup mesh [12], due to the correct handling of the resonance self-shielding of poisoning FPs and minor actinides.

5. Conclusion

This paper has shown that APOLLO2 lattice code and JEFF associated nuclear data is able to predict the reactivity loss with burn-up. Measurements of fuel cycle reactivity loss for high burn-up UOx (up to 70GWd/t) and MOx (up to 60GWd/t) were performed in the MINERVE facility using an oscillation technique. The analysis has pointed out minor biases between calculated and experimental reactivity worth of PWR spent fuel samples. These discrepancies are reduced using the latest European nuclear data File JEFF-3.1.

[C/E-1]±[δE/E] values are summarized as follows:

	JEF-2.2 Results:	JEFF-3.1 Results:
UOx:	+2.3%± 1.3%	+0.6% ± 1.3%
MOx:	-1.0% ± 1.5%	+1.5% ± 1.5%

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