MINERVE Reactor Characterization in Support of the OSMOSE Program: Spectral Indices

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An ambitious program between the Commissariat à l'Energie Atomique (CEA) and the U.S. Department of Energy (DOE) has been launched with the aim of measuring the integral absorption rate parameters in the MINERVE experimental facility located at the CEA Cadarache Research Center. The OSMOSE Program (Oscillation in Minerve of isotopes in "Eupraxic" Spectra) includes a complete analytical program associated with the experimental measurement program and aims at understanding and resolving potential discrepancies between calculated and measured values.

The objective of the OSMOSE program is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs and to provide the experimental data for improving the basic nuclear data files. To gain the most information and insight from integral measurements, the essential reactor parameters should be as well characterized as possible. A description of the neutron spectra, or spectral indices, can be obtained via the use of fission chambers of different actinide isotopes.

As part of the OSMOSE program, extensive measurements have been performed to characterize the MINERVE reactor. The description and results of the spectral indices measurements of the R1-UO2 and R1-MOX reactor configurations are contained herein. Good agreement was obtained between the experimental measurements and calculations performed using the MCNP Monte Carlo code.

KEYWORDS: spectral indices, OSMOSE, MINERVE, actinides, fission chambers, MCNP, TRIPOLI4, integral reaction rates, neutron spectrum, gamma-spectroscopy, modified conversion ratio

I. Introduction

An ambitious program between the Commissariat à l'Energie Atomique (CEA) and the U.S. Department of Energy (DOE) has been launched with the aim of measuring the integral absorption rate parameters in the MINERVE experimental facility located at the CEA Cadarache Research Center. The OSMOSE Program (Oscillation in Minerve of isotopes in "Eupraxic" Spectra) includes a complete analytical program associated with the experimental measurement program and aims at understanding and resolving potential discrepancies between calculated and measured values.

The objective of the OSMOSE program is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs and to provide the experimental data for improving the basic nuclear data files. This program is very generic, in the sense that it will measure these reaction rates over a broad range of isotopes and spectra and will be used to provide guidance to all nuclear data programs in the world.

The OSMOSE program, objectives, and measurement approach is described in another paper in these proceedings [1]. The first step in obtaining better nuclear data consists of measuring accurate integral data and comparing it to integrated energy dependent data: this comparison provides a direct assessment of the effect of deficiencies in the differential data. The OSMOSE program aims at studying a majority of the separated heavy nuclides appearing during the reactor and the nuclear fuel cycle, i.e. ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm and ^{245}Cm. The measurements will be performed in a large range of neutron spectra corresponding to specific experimental lattices (thermal, epithermal, moderated/fast, and fast spectra). Seven different neutron spectra will be created in the MINERVE facility: overmoderated UO_2 (representative of a fuel processing plant or flooded storage cask), UO_2 matrix in water (representative of LWRs), mixed oxide fuel matrix (representative of cores containing MOX fuels), two thermal/epithermal spectra (representative of under-moderated reactors), moderated fast spectrum (representative of fast reactors which have some slowing down due to moderators such as lead-bismuth or sodium), and a very hard spectrum (representative of fast reactors with little moderation from reactor coolant). The different spectra are achieved by changing the lattice within the MINERVE reactor.

A critical aspect of the experimental and analytical program is the definition of the reactor state and characterization of the neutron flux for each configuration of the reactor. The characterization of the reactor for the OSMOSE program includes spectral indices measurements, axial and radial power distributions, and other safety parameters such as control rod worths, core excess reactivity, and control rod worth curves. This paper describes the spectral indices measurements that have been performed for the R1-UO2 and R1-MOX configurations. Another paper in these proceedings [2] describes characterization measurements and calculations performed for the other parameters.

Spectral indices provide a measure of the neutron spectrum at the given location where the measurement is performed relative to a reference neutron spectrum.

The MINERVE reactor is designed to allow a central experimental region that is well-behaved, in fundamental mode, and allows different spectra to be created in the sample location. The central position of the reactor includes a sample location. Samples are oscillated in this central region to conduct the measurements for the experimental programs. By removing the oscillator rod, fission chambers can be placed within the sample location to perform counting during reactor operation. These counting rates can be used to determine ratios of the fission rates for different isotopes, known as spectral indices. The utility and experimental technique of spectral indices measurements has been previously described [3]. Gamma-spectroscopy after irradiation is used to measure the modified conversion ratio of ^{238}U on irradiated fuel pins close to the oscillation channel [4]. It provides information on the ratio of the ^{238}U capture rate to the total fission rate inside the pin.

The spectral indices that have been calculated and measured at MINERVE include:

$$\frac{\overline{\sigma}_{c\,U8}}{\overline{\sigma}_{f,tot}}, \frac{\overline{\sigma}_{f\,P9}}{\overline{\sigma}_{f\,U5}}, \frac{\overline{\sigma}_{f\,P1}}{\overline{\sigma}_{f\,P9}}, \text{and } \frac{\overline{\sigma}_{f\,N7}}{\overline{\sigma}_{f\,P9}}$$

where index f means *fission* and c means *capture*, and where the average cross-sections are defined conventionally as

$$\overline{\sigma} = \frac{\int \sigma \ \Phi dE}{\int \Phi dE}$$

with the common shorthand notation used to identify the isotopes, i.e. U5 is uranium-235, N7 is neptunium-237, etc. Spectral indices are an indicator of the integrated neutron spectra as they are a ratio of the flux-averaged fission cross-section of two isotopes. So based on the level of confidence of the cross-sections or the neutron flux, these ratios allow one to determine the adequacy of the cross-section data and the neutron spectra. The utility of using more than one ratio is that it allows the neutron spectra and cross-section data to be assessed over smaller energy intervals. For example, the ratio of Np-237/U-235 is an indicator of the fast fission effect because Np-237 has a threshold for fission reactions. The ratio of Pu-239/U-235 is an indicator of thermal fission effects because both have high thermal fission cross-sections. All of the ratios provide additional information for a given neutron spectra.

II. Reactor Models

The analytic effort is being performed using separate suites of reactor analysis codes in the U.S. and in France. In this manner, a cross comparison can be performed on the results to identify potential errors in the cross-section evaluations in the numerical methods and assumptions used within the codes. This will allow the improvement of the codes. The initial modeling tasks include the development of a reactor model using MCNP for the R1-UO2 and R1-MOX core configurations, with both JEF2.2 and ENDF-BVI data libraries.

Another Monte Carlo model, based on the same input data, was also developed for R1-UO2 using TRIPOLI-4 French code with JEF2.2. This model will be further extended to the R1-MOX core loading.

Deterministic codes (vs. Monte Carlo codes) are better suited for certain types of reactor analyses and have traditionally been used for core physics calculations. They offer certain advantages over the Monte Carlo technique. In particular, the calculation of the reactivity-worth of small samples can be performed using deterministic codes because the uncertainties in the calculation are smaller. The Monte Carlo technique is not a good alternative because the size of the statistical uncertainties tends to be larger than the reactivity-worth of the samples themselves. However, the deterministic models use macroscopic cross-sections and homogenized regions (vs. discrete models). Therefore, the Monte

Carlo technique usually provides a more detailed solution for the neutron spectrum and spatial flux profile.

Detailed MCNP models of the R1-UO2 and R1-MOX core loadings for the MINERVE reactor were completed. Figures 1 and 2 show cross-sectional views of the core for the R1-MOX case. Figure 3 is a more detailed view of the experimental zone.

The regions are explicitly defined as the physical geometry of the core components. Highlights of the model include: 1. Individually defined elements in the experimental zone (i.e. fuel, cladding, gaps, and spacers), 2. The driver fuel elements include explicit regions for the fuel plates (including clad and spacing), 3. The assemblies are arrayed on a pitch that matches the grid plate, 4. Control rods are defined individually and can be translated in axial position to match the reactor critical configuration, 5. Homogenized regions are included for the table and support structure below the core region, 6. Regions are included above the core to address the withdrawal of control rods and other items that extend above the height of the fuel elements, and 7. The model extends to the size of the reactor tank in all three dimensions.

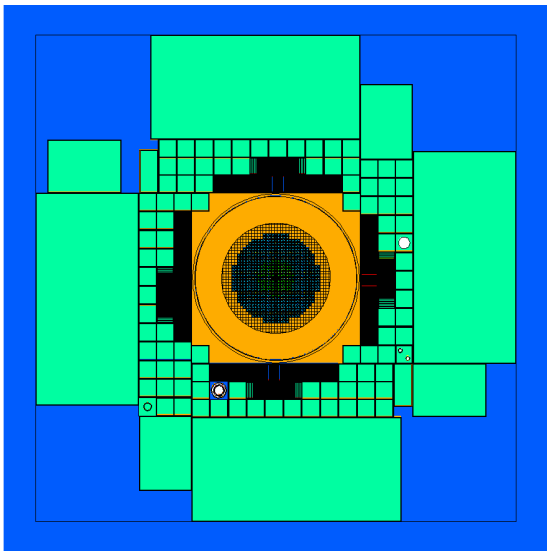

Figure 1: MCNP Mode1 of MINERVE Reactor with R1-MOX Loading (radial view)

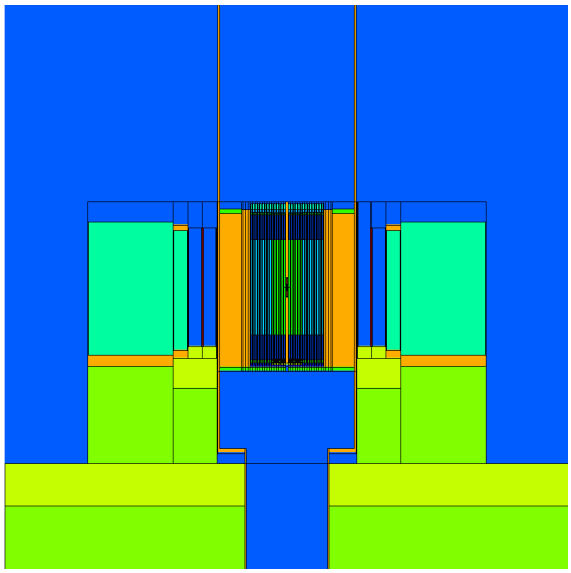

Figure 2: MCNP Model of MINERVE Reactor with R1-MOX Loading (axial view)

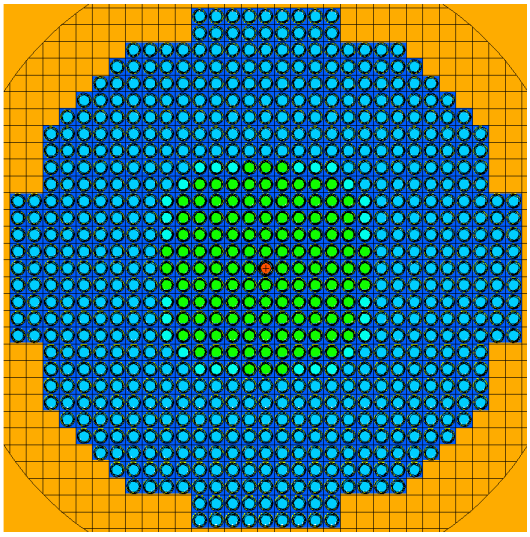

Figure 3: Expanded View of Figure 1 Showing the Experimental Zone for the R1-MOX Loading

III. Experimental Technique

III.1 Spectral Indices

The experimental technique [3] uses miniature 4mm diameter fission chambers that contain the studied isotopes - ^{235}U, ^{239}Pu, ^{241}Pu and ^{237}Np. These chambers are positioned axially and radially in the central part of the MINERVE experimental lattice using a waterproof stainless steel rod. The chambers were previously calibrated in a thermal column spectrum save for the ^{237}Np chamber that was calibrated in a ^{235}U fission spectrum. The fission chambers utilize a standard electronics circuit including a charge sensitive pre-amplifier, amplifier, high voltage power supply, and multi-channel analyzer. The chamber is operated in charge collection mode with the MCA set in a pulse height analyzer mode.

III.2 Conversion Ratio Measurements

Another parameter of interest is the conversion ratio of neutron capture in ^{238}U compared to the total fission rate. This ratio is obtained via gamma spectroscopy measurements of fuel pins in the experimental region. The principle of the γ-ray spectroscopy measurements is to determine the reaction rate of a nuclide by measuring its specific activity. The determination [4] is based on the measurement of the integral photopeak of a high-yield fission product (FP) relative to the total fission rate inside the fuel pin. The specific fission product gamma ray line that is used is the 293.27 keV line from ^{143}Ce. The 277.60 keV gamma ray line from ^{239}Np is used to measure the ^{238}U capture rate because it is related to the number of ^{238}U captures through subsequent beta decays:

$$^{238}U + n \longrightarrow {}^{239}U \xrightarrow{b^- 23.5\text{min}} {}^{239}Np \xrightarrow{b^- 2.355 days} {}^{239}Pu$$

To perform these measurements, a special device has been developed to detect the low energy γ-rays with high accuracy using a Low-Energy Germanium detector and adapted electronics. The

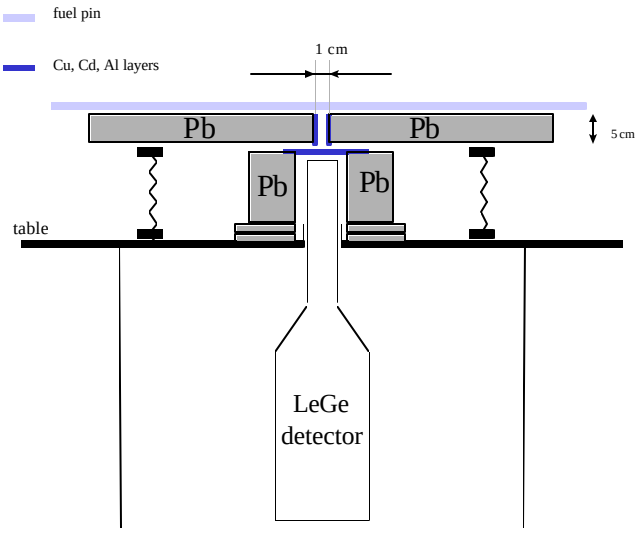

Figure 4: Gamma-ray Spectrometry Device

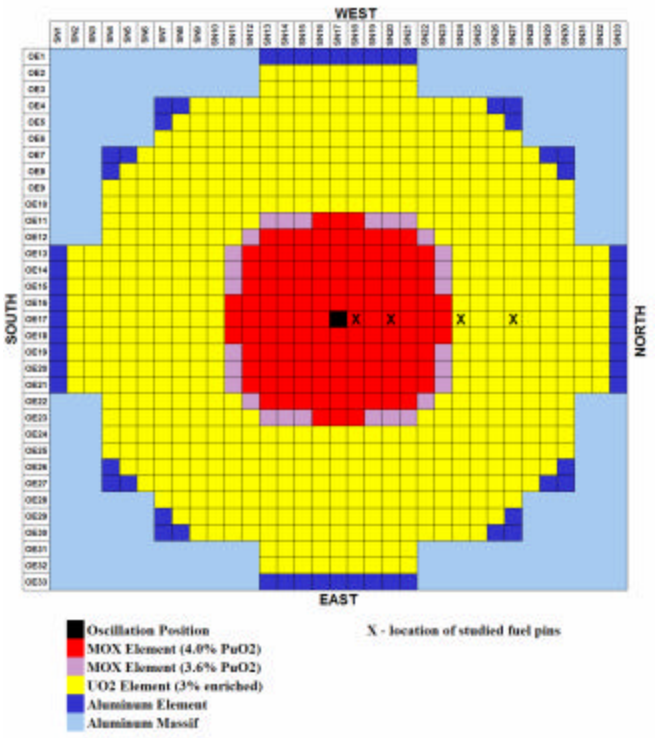

Figure 5: Locations of the Studied Fuel Pins

measurement device is shown schematically in figure 4. One mm thick Cu, Cd, and Al layers help to shield the 70-80 keV X-rays emitted by the lead collimation.

Figure 5 shows the different studied MOX and UOX fuel pins in MINERVE R1-MOX configuration, that are called “MOX1, MOX3, UOX7 and UOX10” (from the center to the outer part of the lattice).

IV. General Methodology

IV.1 Spectral Indices

One assumes that the studied spectral indices are relative to a reaction for which an isotopically pure chamber is available (^{235}U or ^{239}Pu). This assumption is appropriate as the purity of the deposit is more than 99% for the ^{235}U and ^{239}Pu fission chambers, and the impurities are non-fissile isotopes (^{238}U, ^{238}Pu and ^{240}Pu).

The count rates obtained from the reference chamber (denoted with subscript r) in the MINERVE lattice and in the calibration spectrum (denoted with superscript c) are:

$$C_r = N_r k_r \bar{s}_r \Phi$$
$$C_r^c = N_r k_r \bar{s}_r^c \Phi_c$$

where N is the number of atoms in the fission chamber and k is the efficiency of the chamber, with

$$\bar{s} = \frac{\int s\, f dE}{\int f dE} \quad and \quad \Phi = \int f dE ,$$

Further one denotes chambers that contain impurities with subscript 'a', and the impurities in that chamber with subscript 'i'. Thus, σ_a refers to the cross-section of the primary isotope of that chamber, and σ_i refers to the cross-section of the impurities in that chamber. So for chambers with impurities, the count rates are:

$$C_a = k_a(N_a \bar{s}_a + \sum_i N_i \bar{s}_i)\Phi$$
$$C_a^c = k_a(N_a^c \bar{s}_a^c + \sum_i N_i^c \bar{s}_i^c)\Phi_c$$

Note that σ^c represents the average cross-section of the isotope of interest in the calibration spectrum (thermal column in our case). Also note that it is assumed that the efficiencies of the chambers are equivalent in the two spectra.

After treatment the spectral index obtained is:

$$\frac{\bar{s}_a}{\bar{s}_r} = \frac{C_a/C_r}{(C_a/C_r)_c}\left(\frac{N_a^c \bar{s}_a^c}{N_a \bar{s}_r^c} + \sum_{i\neq a}\frac{N_i^c \bar{s}_i^c}{N_a \bar{s}_r^c}\right) - \sum_{i\neq a}\frac{N_i \bar{s}_i}{N_a \bar{s}_r}$$

Using the following notations:

$$s_{a,c} = \frac{\bar{s}_a^c}{\bar{s}_r^c}, s_{i,c} = \frac{\bar{s}_i^c}{\bar{s}_r^c}, s_i = \frac{\bar{s}_i}{\bar{s}_r}, C = \frac{C_a/C_r}{(C_a/C_r)_c}, e_a = \frac{N_a^c}{N_a}, e_{i,c} = \frac{N_i^c}{N_a}, e_i = \frac{N_i}{N_a},$$

the spectral index can be written as

$$S = C(e_a s_{a,c} + e_{i,c} s_{i,c}) - e_i s_i$$

where it should be noted that the summation convention is used to represent the summation on the impurity index, i.

After some manipulation and considering all terms are independent, the fractional uncertainty in the spectral index can be written as:

$$\left(\frac{\Delta S}{S}\right)^2 = \left(\frac{S + e_i s_i}{S}\right)^2\left(\frac{\Delta C}{C}\right)^2 + \left(\frac{C e_a s_{a,c}}{S}\right)^2\left[\left(\frac{\Delta e_a}{e_a}\right)^2 + \left(\frac{\Delta s_{a,c}}{s_{a,c}}\right)^2\right] + \left(\frac{C e_{i,c} s_{i,c}}{S}\right)^2\left[f(e)\left(\frac{\Delta e_{i,c}}{e_{i,c}}\right)^2 + f(s)\left(\frac{\Delta s_{i,c}}{s_{i,c}}\right)^2\right]$$
$$+ \left(\frac{e_i s_i}{S}\right)^2\left[f(e)\left(\frac{\Delta e_i}{e_i}\right)^2 + f(s)\left(\frac{\Delta s_i}{s_i}\right)^2\right] + f(e)\left(\frac{S - C e_a s_{a,c}}{S}\right)^2\left(\frac{\Delta e_i}{e_i}\right)^2$$

where $f(\varepsilon)=0$ if $\varepsilon_{i,c}=\varepsilon_i$, and is unity otherwise, and $f(\sigma)=0$ when i=r, and is unity otherwise.

For the special case of the ^{237}Np fission chamber, it is not possible to calibrate in a thermal column due to the fact that the fission cross-section of ^{237}Np is a threshold reaction, and therefore zero in a thermal column. Alternatively, an effective mass from calibration in a ^{235}U fission spectrum can be used as a reference because ^{237}Np is isotopically pure. Using a similar approach, the spectral index is written:

$$\frac{\bar{s}_a}{\bar{s}_r} = \frac{M_r}{M_a}\frac{A_a}{A_r}\frac{C_a}{C_r},$$

where M is the effective mass of the chamber and A is the atomic weight. Using the notations:

$$m = \frac{M_r}{M_a}\frac{A_a}{A_r}, \quad C = \frac{C_a}{C_r},$$

the spectral index becomes $S = mC$ and the fractional uncertainty can be written as

$$\left(\frac{\Delta S}{S}\right)^2 = \left(\frac{\Delta m}{m}\right)^2 + \left(\frac{\Delta C}{C}\right)^2$$

IV.2 Conversion Ratio

Considering the β^- decay constant of ^{239}U as negligible in comparison with that of ^{239}Np, the resolution of the equations of evolution of ^{143}Ce and ^{239}Np during the different phases of the measurement leads to the conversion ratio:

$$\frac{C8}{F} = \frac{\lambda_{U9} - \lambda_{Np}}{\lambda_{FP}} \times \frac{\lambda_{Np}}{\lambda_{U9}} \times \frac{N_{Np}}{N_{FP}} \times Y_{FP} \times \frac{g_{FP}}{g_{Np}} \times \frac{h_{FP}}{h_{Np}} \times \frac{f_{FP}}{f_{Np}} \times \frac{\left(1 - e^{-\lambda_{FP} \times t_e}\right) \times e^{-\lambda_{FP} \times t_c} \times \left(1 - e^{-\lambda_{FP} \times tm}\right)}{\left(1 - e^{-\lambda_{Np} \times t_e}\right) \times e^{-\lambda_{Np} \times t_c} \times \left(1 - e^{-\lambda Np \times tm}\right)}$$

where C8 is the ^{238}U capture rate, F is the total fission rate, λ_i is the decay constant of nuclide *i*, N_i is the integral counts of photopeak for nuclide *i*, η_i is the detection efficiency for gamma ray of nuclide *i*, g_i is the gamma-ray emission probability for nuclide *i*, Y_{FP} is the effective fission yield of fission product *FP*, f_i is the gamma-ray shielding factor of nuclide *i* (or the probability for the gamma rays of nuclide *i* to escape from the fuel to the detector without any interaction), and t_e, t_c, t_m are the irradiation, cooling, and measurement times.

Values of N_i and η_i are obtained from measurement using the photopeaks of ^{239}Np (277.60 keV) and ^{143}Ce (293.27 keV). After calibration of the counting system in energy and in efficiency (with an accuracy better than 1.3 %), counting is performed between 2 and 4 days after irradiation at 240 W/h. Counting times are established so that accuracies greater than 1% (i.e. >10000 counts) are obtained. An example of the counting is shown in figure 6.

Values for Y_{FP} and f_i are obtained from calculations. The gamma-ray shielding factors and the effective fission yields are computed using MCNP. Calculations of the reaction rates inside the fuel pin are necessary to determine the gamma-ray shielding factors and the effective fission yield. The fuel pellet is divided into 20 equal volumetric and concentric regions, in which both total fission rate and ^{238}U capture rate are calculated. The total thermal and fast fission rates and the ^{238}U capture rate are also calculated for the entire fuel pin.

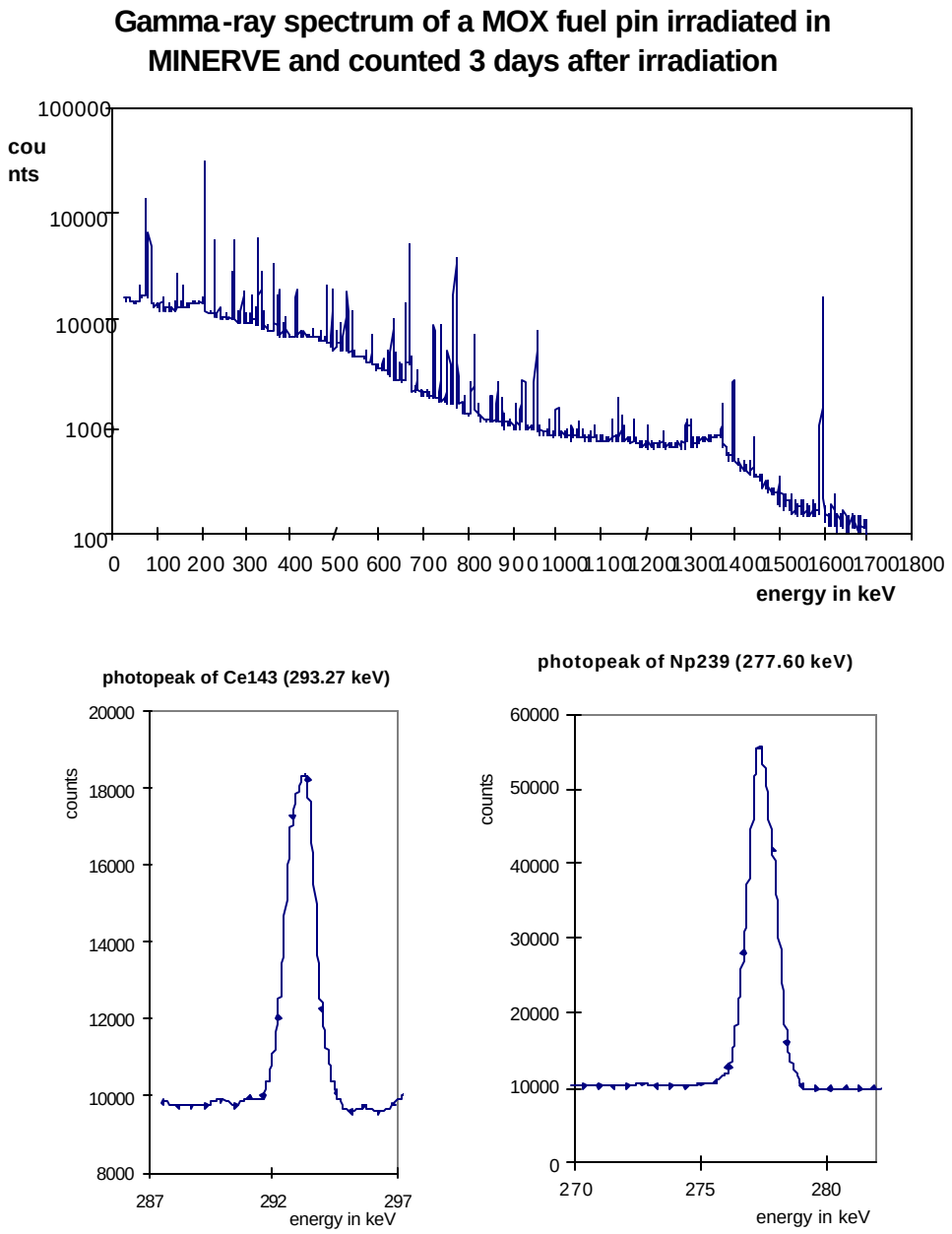

Figure 6: Gamma-Ray Spectrum and Photopeaks

The gamma-ray shielding factor can be determined as follows (see results in table 1):

$$f = \frac{\sum_i t_i \times v_i}{\sum_i t_i \times c_i}$$

where τ_i is the fission or capture rate in the region i, ?$_i$ is the probability for the gamma ray to escape from region i to the detector in the case of a void pin (i.e. pin without fuel), and c_i is the probability for the gamma ray to escape from region i to the detector for a fuel pin.

Table 1 Gamma-Ray Shielding Factors for MINERVE R1-MOX Fuel Pins		
	Ce-143	Np-239
MOX1	0.349	0.330
MOX3	0.348	0.330
UOX7	0.346	0.345
UOX10	0.346	0.344

Table 2 Effective Fission Yields of Ce-143 in MINERVE R1-MOX Fuel Pins		
	Y_{Ce}	s.d. (%)
MOX1	4.614	1.03 %
MOX3	4.613	1.03 %
UOX7	5.864	1.29 %
UOX10	5.881	1.31 %

The associated uncertainty can be written as follows:

$$\left(\frac{\Delta f}{f}\right)^2 = \frac{\sum_i \left(t_i^2 \times (\Delta(n_i))^2\right)}{\left(\sum_i t_i \times n_i\right)^2} + \frac{\sum_i \left(t_i^2 \times (\Delta(c_i))^2\right)}{\left(\sum_i t_i \times c_i\right)^2} + \frac{\sum_i \left((\Delta(t_i))^2 \times (n_i)^2\right)}{\left(\sum_i t_i \times n_i\right)^2} \frac{\sum_i \left((\Delta(t_i))^2 \times (c_i)^2\right)}{\left(\sum_i t_i \times c_i\right)^2}$$

where $\Delta\tau_i$, Δn_i and Δc_i are the standard deviations given by Monte Carlo calculations. The same uncertainty of 1.1 % was found on each gamma-ray shielding factor presented in table 1.
The effective fission yield can be written as follows:

$$Y_{FP} = \sum \left(Y_{iT} \times t_{iT} + Y_{iF} \times t_{iF}\right)$$

where Y_{iT} and Y_{iF} are the thermal and fast fission yield of nuclide i (given by ENDF-BVI data library), τ_{iT} is the ratio of the thermal fission rate to the total fission rate for nuclide i, and τ_{iF} is the ratio of the fast fission rate to the total fission rate for nuclide i. Results for the effective fission yields are shown in table 2 with statistical error from MCNP and ENDF-BVI data library.

The associated uncertainty can be written as follows :

$$\left(\frac{\Delta Y_B}{Y_B}\right)^2 = \frac{\sum_{i,j} \left[Y_{ij}^2 \times (\Delta t_{ij})^2 + (\Delta Y_{ij})^2 \times t_{ij}^2\right]}{Y_B^2}$$

where $\Delta\tau_{ij}$ is the standard deviation relative to the convergence of Monte Carlo calculation for isotope i in the energy range j, and ΔY_{ij} is the uncertainty on ENDF-BVI fission yield for isotope i in the energy range j.

Note that for both gamma-ray shielding factors and effective fission yields, the bias on τ_i due to nuclear data or the calculation scheme is not taken into account as it will bias all the results in the same way and as only the ratios of shielding factors and effective fission yields appear in the expression of normalization factors.

Table 3 Experimental Results of Spectral Indices				
	R1-UO2		R1-MOX	
Ratio	Spectral Index	s.d. (%)	Spectral Index	s.d. (%)
Pu9/U5	1.924	2.5 %	1.952	2.5 %
Pu1/Pu9	1.134	2.7 %	1.123	2.7 %
Np7/Pu9	0.00378	3.2 %	0.00729	3.2 %

Table 4 Experimental Results of Modified Conversion Ratio for R1-MOX lattice		
Position	C8/F	s.d. (%)
MOX1	0.499	2.8 %
MOX3	0.499	2.8 %
UOX7	0.569	2.8 %
UOX10	0.488	2.8 %

V. Experimental Results

The experimental results of spectral indices measurements in R1-UO2 and R1-MOX are shown in Table 3. The experimental uncertainties are from 2% to 3 % in R1-UO2 and in R1-MOX.

The experimental results for the modified conversion ratio are shown in table 4. The experimental uncertainty is obtained by combining the uncertainties for all parameters in the equation for C8/F in section IV.2. The conversion ratio is much higher at the interface of the MOX and UOX regions (pin UOX7) because of a much harder neutron spectrum at this position.

VI. Calculational Results

The comparison of experiments to calculations is given in tables 5 and 6 for spectral indices in R1-UO2 and R1-MOX, and in table 7 for the modified conversion ratio. Note that all standard deviations presented in those tables are obtained combining quadratically the experimental uncertainties and the Monte Carlo convergence standard deviations.

In R1-UO2, the C/E ratios are consistent within better than 2.4 % for ratios of fissile isotopes, i.e. Pu9/U5 and Pu1/Pu9. As for Np7/Pu9 index, it shows an underestimation of the calculation of about 6% with MCNP-4C2 (that is however in agreement with the experimental index within 2 experimental standard deviations) and a very good consistency with TRIPOLI4.

Table 5 Comparison of Experiments and Calculations for R1-UO2 Lattice				
MCNP-4C2 calculation				
	ENDF-BVI		JEF2.2	
Ratio	C/E	s.d. (%)	C/E	s.d. (%)
Pu9/U5	1.008	3.5 %	1.012	3.5 %
Pu1/Pu9	1.004	3.6 %	1.003	3.6 %
Np7/Pu9	0.938	3.7 %	0.953	3.7 %
TRIPOLI4 calculation				
Pu9/U5	/	/	1.031	2.8 %
Pu1/Pu9	/	/	0.993	3.0 %
Np7/Pu9	/	/	0.989	3.3 %

Table 6 Comparison of Experiments and MCNP-4C2 Calculations for R1-MOX lattice				
	ENDF-BVI		JEF2.2	
Ratio	C/E	s.d. (%)	C/E	s.d. (%)
Pu9/U5	0.990	3.9 %	1.008	4.0 %
Pu1/Pu9	1.014	4.0 %	1.010	4.1 %
Np7/Pu9	1.044	4.1 %	1.002	4.0 %

Table 7 Comparison of Experiments and Calculations for R1-MOX lattice				
MCNP-4C2 calculation				
C8/F	ENDF-BVI		JEF2.2	
position	C/E	s.d. (%)	C/E	s.d. (%)
MOX1	1.067	3.3 %	1.054	3.56%
MOX3	1.044	3.3 %	1.071	3.48%
UOX7	1.065	3.2 %	0.988	3.40%
UOX10	1.069	3.2 %	1.025	3.36%

For the R1-MOX configuration, the C/E ratios are again consistent within better than 1.5 % for ratios of fissile isotopes, i.e. Pu9/U5 and Pu1/Pu9. The Np7/Pu9 index shows a good agreement between experiment and calculation for both ENDF-BVI and JEF2.2.

For the modified conversion ratio, comparison between experiment and calculation shows an overestimation of the calculation between 2% and 7% (safe for pin *UOX7* with JEF2.2 for which the experimental result needs further investigation), most likely a result of the capture cross section of ^{238}U, depending on the nature of the fuel pin, and on its position (i.e. also the neutron spectrum), and also eventually a result of a slight error in the material balance of MOX fuel pins. Also note that a bias on the gamma-ray shielding factors on pin *UOX7* is possible, as this pin is placed at the MOX-UOx interface and so as division of the fuel pellet in 20 equal volumetric and concentric regions (see § IV.2) may be not perfectly adapted to the strong gradient of flux in this place.

VII. Conclusions

This paper has described the experimental results of spectral indices measurements in R1-UO2 and R1-MOX configurations of the MINERVE facility, and of the modified conversion ratio of ^{238}U in R1-MOX. It has also compared the results to calculations performed with the MCNP code with JEF2.2 and ENDF-BVI data libraries, and with the TRIPOLI4 code with JEF2.2 data library for spectral indices in R1-UO2.

The main conclusions are as follows:

1. There is excellent agreement between calculation and experiments for Pu9/U5 and Pu1/Pu9 spectral indices, within 2.4%, for both R1-MOX and R1-UO2 cores.
2. There is an underestimation of the Np7/Pu9 index in R1-UO2 of about 5% with MCNP (however within 2 standard deviations) and a good agreement with TRIPOLI4 + JEF2.2, and good agreement in the case of R1MOX with MCNP.
3. There was an overestimation of the calculation for the modified conversion ratio of ^{238}U from 2% to 7%. This is most likely due to the capture cross section of ^{238}U, and to a potential bias on the material balance of MOX fuel pins.

These initial results show that the MCNP and TRIPOLI4 models developed for the R1-UO2 and R1-MOX configurations of the MINERVE reactor give consistent results with the experimental results for most of the physical parameters that have been measured.

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