MCNP5 CALCULATIONS COMPARED TO EXPERIMENTAL MEASUREMENTS IN CEA-MINERVE REACTOR

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The paper describes the calculation performed by INR-Pitesti of a benchmark problem proposed by CEA Cadarache consisting in experimental measurements in MINERVE reactor. The purpose of the problem concerned the qualification of Valmont fuel pin (U-Mo-Al) designed for radio-pharmaceutical sources production. First, an MCNP model for the MINERVE facility was created. The model was then used to obtain results to compare with safety parameters and experimental physics measurements such as: control rods worth, Valmont pin reactivity worth, axial fission rate distribution in some of the experimental zone pins including Valmont, spectral indices and U-238 conversion ratio. A description of the MCNP model is provided in the paper together with the results obtained against the experimental results.

Keywords: Benchmark, MCNP, MINERVE.

1. Introduction

The present work was done as a contribution to an IAEA CRP for benchmarking against experimental data of the thermalhydraulic and neutronic computer codes for research reactor analysis. For 2010-2011, INR engaged in neutronic calculation of the MINERVE reactor.

MCNP5 [1] is a general-purpose Monte Carlo N–Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport, including the capability to calculate eigenvalues for critical systems. Using MCNP5, in the frame of the above mentioned CRP, the main author of this paper has previously done calculations for SPERT-III reactor, consisting in steady-state neutronic calculations with respect to hot and cold core reactivity excess and control rods worth, and also prompt neutron life and reactivity feed-back coefficients [4]. This time, a pool facility dedicated to neutronics measurements (MINERVE) is concerned. This activity constitutes an opportunity for verification of our skills in using MCNP for both safety parameters calculation at research reactors and experimental physics measurements simulation.

2. MINERVE facility description

According to [2], the reactor consists of a driver core and an experimental lattice. The central cavity which hosts the experimental lattice has a variable size

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adjustable by slicing the grids of the driver core. The reactor is surrounded with a graphite wall. A radial cut of the facility is given Fig.1, where the different zones composing the facility are indicated.

The experimental zone consists of a UO₂ fuel pins lattice positioned in the central cavity.

The experimental zone is surrounded by an aluminum buffer region. On the external edges of the buffer region there are driver zone fuel elements which are plate type (MTR), 90-93% enriched. Around the driver zone there are graphite reflector blocks, with different dimensions. For neutronics calculations the core is assumed to be surrounded by light water beyond graphite reflector.

3. **MCNP model for MINERVE**

The model was created using the information offered by CEA-Cadarache in [2]. An overall radial and axial view of the facility model produced with the MCNP visual editor is given in Fig.2 and Fig.3 respectively.

There are two essential zones in the model:
1) the experimental zone – located in the central area and composed of a fuel pins lattice, 776 pins with UO₂, 3% enriched, cladded in zircaloy and 24 aluminum pins in the corners of the fuel lattice as shown in the detail of the experimental zone (see Fig.4). The experimental zone is surrounded by a buffer region composed of aluminum blocks.

The cell of a fuel pin inside the experimental zone is shown in Fig.5. From center to periphery there are: the UO₂ pellet, the zircaloy cladding (homogenized with gap), a water gap between cladding and overclad, the aluminum overclad and
inter-pins water. The UO\(_2\) material in a “normal” pin is replaced by U-Mo-Al 19.75%.

Axially, an UO\(_2\) pin has different regions with their respective heights as depicted in Fig.6, which are represented in the model by the different materials seen in Fig.7.

2) the driver zone – consists of MTR fuel plate assemblies and reflector assemblies. There are several types of assemblies: with 18, 12 or 9 plates. The fuel plates are made of uranium-aluminum alloy with 20% of uranium which mass enrichment is equal to 90% or 93% \(^{235}\)U.

Concerning the modelling, Fig.9 through Fig.10 presents a zoom made in the x-y directions of the MCNP model on one of the 18 plates and 9 plates assemblies respectively. The active height of the plates is 60 cm and the middle height of these corresponds to the middle height of the UO\(_2\) pellets in the experimental zone.
**Control rods** - Control rods are standard 18 fuel plates elements with mass enrichment 93% where plates have been replaced by:

- Plates 3, 5 and 14, 16: inert aluminum plates
- Plates 4 and 15: hafnium element

Hafnium element consists of 2 absorber plates with a stainless steel extender. There are four such control assemblies. Fig. 11 presents a control assembly from the MCNP model at the axial level of the absorbant (cladded in stainless steel). The length of the absorbant which is not inside the core, is placed in the top reflector (Fig. 11).

![Fig.6. Fuel pin- Experimental zone (2)](image1)

![Fig.7. MCNP model - axial model of the fuel pin](image2)

![Fig.8. Loading of the driver zone (2)](image3)
4. Calculations and results against experimental data

The series of calculations with the model described was composed of: control rods worth; Valmont pin reactivity worth; axial fission rate distribution in some of the experimental zone pins; spectral indices; U-238 conversion ratio.

4.1 Control rods worth

For the work dedicated to this point, information from [3] was used. First, the excess reactivity of the reactor was determined with all rods fully extracted. Then, in different computations, each rod was fully inserted while the others remained extracted. Finally, a case was run with all four rods fully inserted. Reactivities were converted in dollars ($) using a delayed neutron fraction of 716 pcm given both in [2] and [3]. The calculated core reactivity excess is 3.96 $ with a standard deviation of 0.05 $.

The comparison of our calculated values with the experimental data is presented in Tab.1. As can be noticed in the table, calculated mean values are within the experimental error interval (or very close to it). The calculated standard
deviation was reached in these calculations after 950 active cycles with 5000 histories each.

Table 1

<table>
<thead>
<tr>
<th>Rod inserted</th>
<th>Experimental worth ($)</th>
<th>Calculations with MCNP5 ($)</th>
</tr>
</thead>
<tbody>
<tr>
<td>B1</td>
<td>1.37 ± 0.06</td>
<td>1.44 ± 0.07</td>
</tr>
<tr>
<td>B2</td>
<td>1.25 ± 0.06</td>
<td>1.30 ± 0.07</td>
</tr>
<tr>
<td>B3</td>
<td>1.52 ± 0.07</td>
<td>1.44 ± 0.07</td>
</tr>
<tr>
<td>B4</td>
<td>1.55 ± 0.08</td>
<td>1.47 ± 0.07</td>
</tr>
<tr>
<td>All rods</td>
<td>7.98 ± 0.43</td>
<td>7.54 ± 0.07</td>
</tr>
</tbody>
</table>

4.2 Valmont pin reactivity worth

Criticality was sought changing the absorber elements position (the same for all four elements), with UO$_2$ in the centre. Then, the pin in the centre of the experimental zone was replaced with a Valmont pin. The computations ran 10000 cycles with 5000 particles histories each. The results are given in Tab.2. Although the mean statistical value is relatively good (11% lower than the experimental), the propagated uncertainty is large compared to the small change in reactivity.

Table 2

<table>
<thead>
<tr>
<th>keff UO$_2$</th>
<th>keff Valmont</th>
<th>Calculated Valmont weight ($)</th>
<th>Experimental Valmont weight ($)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.00065 ± 11pcm</td>
<td>1.00094 ± 11pcm</td>
<td>0.040 ± 0.022</td>
<td>0.0453 ± 0.001</td>
</tr>
</tbody>
</table>

4.3 Axial fission rate distribution

The fission rate axial distribution used 20 intervals along the fuel pins from the experimental zone. Normalization to the maximum local/average value was performed in order to be consistent with the reported experimental curves. Fig.12 presents the calculated distributions and Fig.13 the experimental ones.

Fig.12. Calculated axial fission rate distribution, in Valmont pin and UOX04 fuel pins

Fig.13. Axial fission rates distribution for Valmont, UOX01 and UOX04 fuel pins of the R1-UO2 lattice [2]
The requested tallies were produced by a 10000 active cycles run with a relative standard deviation of about 2%.

4.4 Spectral indices

As described in [2], the spectral indices are measured by using \(^{235}\text{U}\), \(^{241}\text{Pu}\), \(^{239}\text{Pu}\) and \(^{237}\text{Np}\) miniature fission chambers located in the central channel at fuel mid-plan. In the calculations, these isotopes and also \(^{238}\text{U}\) are introduced for tallies. The central pin interior was replaced by void in order to model in a simple manner the measurement configuration. Tab.5 summarizes the results and highlights the comparison between calculation and measurements.

<table>
<thead>
<tr>
<th>Fission ratio</th>
<th>Value</th>
<th>Standard deviation (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(^{239}\text{Pu})/(^{235}\text{U})</td>
<td>Calc. 1.954</td>
<td>Measured 1.913</td>
</tr>
<tr>
<td>(^{241}\text{Pu})/(^{239}\text{Pu})</td>
<td>Calc. 1.121</td>
<td>Measured 1.133</td>
</tr>
<tr>
<td>(^{237}\text{Np})/(^{239}\text{Pu})</td>
<td>Calc. 0.00358</td>
<td>Measured 0.00383</td>
</tr>
</tbody>
</table>

4.5 Conversion ratio

Reference [2] defines the modified conversion ratio C8/F as the ratio between \(^{238}\text{U}\) capture rate to the total fission rate inside the fuel pin. It is measured by using single peak gamma-scanning method on irradiated fuel pin. Tab.6 gives the results of the calculation compared to measurements.

<table>
<thead>
<tr>
<th>Conversion ratio results</th>
</tr>
</thead>
<tbody>
<tr>
<td>C8/F calculated</td>
</tr>
<tr>
<td>UOX4 Valmont</td>
</tr>
<tr>
<td>0.491</td>
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</tbody>
</table>

5. Conclusions

An MCNP model for MINERVE facility was created and was used to calculate safety parameters and neutron physics measurements. Comparison with measured data proved good agreement and demonstrates once again the capability of MCNP5 for various purposes at nuclear reactors, ranging from safety analysis to research in test facilities.
Concerning the benchmark exercise proposed by CEA-Cadarache, it is complete and data are clear, while additional information is publicly available over the Internet. In general, we consider that the quality of the benchmarks should be thoroughly scrutinized before being proposed as tools for validating computer codes and methodologies for calculation, since not every “benchmarking” problem reach this standard.

REFERENCES


