Calculation of Kinetic Parameters of the Moroccan TRIGA Mark-II Reactor Using the Monte Carlo Code MCNP

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Abstract

In reactor physics calculations, the measurement of kinetic parameters has an important influence on control and safety analysis of a nuclear reactor. The most important kinetic parameters of nuclear reactors are the effective delayed neutron fraction ($\beta_{\text{eff}}$), the prompt removal lifetime ($\tau_r$) and the mean neutron generation time ($\Lambda$). In this study, the kinetic parameters of the Moroccan TRIGA Mark-II reactor were calculated using the 3-D continuous-energy Monte Carlo code MCNP5. The geometrical structure of the Moroccan reactor core and the fuel configuration were designed. The MCNP Results of these parameters are analyzed and compared with references data.

Keywords: Research Reactor, TRIGA, MCNP code, Kinetic parameters, Safety

1. Introduction

Kinetic parameters measurement in nuclear reactors is the most important factor with respect to the control and safety of reactor. The principal kinetic parameter of nuclear reactors is the effective delayed neutron fraction ($\beta_{\text{eff}}$) which is defined as the ratio between the adjoint and the spectrum weighted number of fissions induced by delayed neutrons and number of fissions induced by all neutrons [4]. Generally, in nuclear reactor, over 99% of neutrons are emitted promptly at fissions and a small but important fraction (~1%) of the neutrons emitted
in fission events are delayed neutrons. The delayed neutrons are emitted by decay of precursors at times later than prompt fission neutrons [6]. Other kinetic parameters are the prompt removal lifetime (τₚ) and the mean neutron generation time (Λ), which will be defined in this paper. The kinetic parameters strongly depend on the number of fuel elements in the core, the enrichment of ²³⁵U, the core configuration (size) and the nuclear data library used in calculations. These parameters are usually provided by the manufacturer in the design phase for the research reactors and are not calculated for various core conditions [7].

In Morocco, a preliminary analysis of the TRIGA Mark II reactor was done with a simplified model. A three-dimensional model of the TRIGA reactor was elaborated using the Monte Carlo code MCNP5. Some practical calculations of the neutronic safety parameters of the Moroccan TRIGA such as core excess reactivity, total and integral control rods worth and power peaking analysis were performed on the one hand, and the burnup simulation was established by using an internally developed burnup code called BUCAL1 on the other hand [1], [3]. Also, the safety analysis and optimization of the core fuel when the reactor is working with a power around 2 MW was established [2].

In this study, the most important kinetic safety parameters such as effective delayed neutron fraction (βeff), prompt removal lifetime (τₚ) and the mean neutron generation time (Λ) were calculated for the Moroccan TRIGA Mark-II research reactor using the MCNP5 code. The simulation results concerning these parameters were compared with the values of reference data for the TRIGA Mark-II reactor type.

2. Simulation of the Moroccan TRIGA reactor

The Moroccan research reactor is a TRIGA Mark-II type. It is only the nuclear research reactor in Morocco that achieved initial criticality on May 2, 2007 of 2MW nominal power. The reactor has been used for manpower training, radioisotope production, neutron activation analysis and in the various fields of the nuclear research. This reactor is moderated and cooled by light water, equipped with a graphite reflector, four beam ports and a thermal column (See Fig.3).

The TRIGA core consists of 96 fuel elements, 5 fuel follower control rods, 17 graphite elements, 1 central thimble and 1 pneumatic transfer system irradiation terminus. Fig.2 shows the MCNP simulation of the geometrical configuration of core, the fuel elements, graphite dummy elements and the control rods loaded in the core. The reactor core is located near the bottom of water-filled aluminium tank. The reactor tank has a diameter of 2.44m and a depth of 8.84m, which is surrounded by a concrete biological shield structure. The reactor fuel is a solid homogeneous mixture of hydride of uranium-zirconium (U-ZrH), enriched about 20% of ²³⁵U, encapsulated in a stainless steel cladding [5]. The H/Zr atom ratio is approximately 1.65. The fuel element structure of the TRIGA Mark-II reactor is described in Fig.1, and the Physical properties of fuel and fuel follower elements are presented in Table.1 [1].
Calculation of kinetic parameters

**Fig.1.** MCNP model of the fuel element in TRIGA core

**Fig.2.** Core configuration of the Moroccan TRIGA reactor via MCNP code

<table>
<thead>
<tr>
<th></th>
<th>Fuel element (cm)</th>
<th>Fuel follower (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Outer diameter</td>
<td>3.76</td>
<td>3.44</td>
</tr>
<tr>
<td>Fuel diameter</td>
<td>3.65</td>
<td>3.33</td>
</tr>
<tr>
<td>Fuel height</td>
<td>38.1</td>
<td>38.1</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.0508</td>
<td>0.0508</td>
</tr>
<tr>
<td>Diameter of zirconium rod</td>
<td>0.6350</td>
<td>0.6350</td>
</tr>
<tr>
<td>Amount of uranium U-ZrH (wt %)</td>
<td>8.5</td>
<td></td>
</tr>
</tbody>
</table>

Table 1. Geometrical data and material composition of fuel and fuel follower element

3. Calculation results and discussion

3.1. Calculation of $\beta_{\text{eff}}$

The effective delayed neutron fraction ($\beta_{\text{eff}}$) is one of principal kinetic parameters of nuclear reactor. The delayed neutrons have the ability to induce fission and consequently release power in the fission process. In order to calculate the $\beta_{\text{eff}}$, the TOTNU card to produce delayed neutrons as well as prompt neutrons (KCODE default) has been used in calculation via MCNP code. To calculate the effective delayed neutron fraction, the following equation [a] has been used [8], [9]:

...
Where $K_{\text{eff, tot}}$ and $K_{\text{eff, prompt}}$ are total and prompt effective multiplication factor respectively, $N_d$ is the number of fissions induced by delayed neutrons and $N_{\text{tot}}$ is the total number of fissions induced by total neutrons.

To evaluate the value of the effective delayed neutron fraction of the Moroccan TRIGA reactor, the fuel rod was modelled exactly, meaning that Zr rod, Molybdenum supporting disc, stainless steel cladding and air gaps were modelled explicitly. The graphite reflector, supporting grid and irradiation channel in the core were also explicitly modelled (see Fig.3). In order to accurately model the neutron interactions at energies below a few eV, $S(\alpha, \beta)$ thermal neutron treatment for hydrogen, graphite and zirconium as well as the continuous-energy neutron interaction data based on the ENDF library were used.

Two different simulations runs with and without the TOTNU card was performed. In the first run, the effective multiplication factor $K_{\text{eff, tot}}$ is measured by taking into account of both prompt and delayed neutrons, in the second run, $K_{\text{eff, prompt}}$ is calculated, but just considering the prompt neutron contribution. In this study, the effective multiplication factor was calculated in critical operating condition of the Moroccan TRIGA reactor, such as ENDF is the nuclear data library used in our work.

![Fig.3](image_url) The axial view of the Moroccan TRIGA reactor model via MCNP Code

\[ \beta_{\text{eff}} = \frac{N_d}{N_{\text{tot}}} = \frac{K_{\text{eff, tot}} - K_{\text{eff, prompt}}}{K_{\text{eff, tot}}} = \frac{K_{\text{eff, delayed}}}{K_{\text{eff, tot}}} \]
The MCNP calculations were performed by the NJOY nuclear processing system, run with 10,000 cycles of iterations with a nominal number of source histories of 10,000 neutrons per cycle, for a total of 100 million simulated neutrons. The first 100 cycles was escaped to obtain a well-distributed neutron source. The multiplication factor had a relative error less than 10pcm. The equation [a] and the simulation results of the $K_{\text{eff,tot}}$ and $K_{\text{eff,prompt}}$ give the calculated $\beta_{\text{eff}}$ value, which is presented in Table 2.

<table>
<thead>
<tr>
<th>Moroccan TRIGA Fuel</th>
<th>$K_{\text{eff,tot}}$</th>
<th>$K_{\text{eff,prompt}}$</th>
<th>$\beta_{\text{eff}}$(calculated)</th>
<th>$\beta_{\text{eff}}$(reference)</th>
<th>Error (%) of $\beta_{\text{eff}}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>8.5% wt of U</td>
<td>1.00021 ± 0.00008</td>
<td>0.99278 ± 0.00008</td>
<td>0.00742</td>
<td>0.00730</td>
<td>1.65%</td>
</tr>
</tbody>
</table>

Table 2. The calculated values of $K_{\text{eff,tot}}$, $K_{\text{eff,prompt}}$ and $\beta_{\text{eff}}$ in the Moroccan TRIGA reactor.

3.2. Calculation of $\tau_r$ and $\Lambda$ in the Moroccan reactor

The average time from the emission of a prompt neutron in fission to the removal of the neutron by some physical process such as fission, capture, or escape, is called neutron prompt removal lifetime $\tau_r$. The difference between the mean neutron generation time $\Lambda$ and neutron prompt removal lifetime $\tau_r$ is that the $\Lambda$ taking into account only the neutron absorptions inducing fission. In order to evaluate $\tau_r$ using the MCNP code, the TOTNU card to produce delayed neutrons as well as prompt neutrons was used. The following equation [b] defines mathematically the kinetic parameter ($\Lambda$).

$$\Lambda = \frac{\tau_r}{k_{\text{eff}}}$$  \hspace{1cm} [b]

Such as $K_{\text{eff}}$ and $\tau_r$ are the effective multiplication factor and the prompt removal lifetime respectively [12]. The MCNP results concerning the prompt removal lifetime ($\tau_r$) and the mean neutron generation time ($\Lambda$) values, for the Moroccan TRIGA Mark-II reactor are presented in Table 3, with $K_{\text{eff}} = 1.00021$

<table>
<thead>
<tr>
<th>Moroccan TRIGA Fuel</th>
<th>Enriched in $^{235}$U</th>
<th>MCNP5 value of $\tau_r$(µs)</th>
<th>MCNP5 value of $\Lambda$ (µs)</th>
</tr>
</thead>
<tbody>
<tr>
<td>8.5% Wt of U</td>
<td>20%</td>
<td>43.91 ± 0.15</td>
<td>43.90 ± 0.14</td>
</tr>
</tbody>
</table>

Table 3. Calculated Prompt removal lifetime ($\tau_r$) and mean neutron generation time ($\Lambda$) values using MCNP5 code, in TRIGA reactor of Morocco.
3.3. Discussion

According to the references, several calculation methods were performed to estimate the effective delayed neutron fraction for TRIGA reactors type. In 1965, Keepin calculated the $\beta_{\text{eff}}$ value, which is equal to 0.0065. Furthermore, a study has been performed for four core of TRIGA reactor type, such as every TRIGA core contains a different amount of fuel, 8.5 w/o, 12 w/o, 20 w/o and 30 w/o of uranium[8]. The corresponding obtained values of the $\beta_{\text{eff}}$ were 0.00740, 0.00751, 0.00730 and 0.00758 respectively. Other value of the $\beta_{\text{eff}}$ in the reference data was reported ~0.007 for TRIGA core type [10], [11].

In this work, the MCNP calculated value of the effective delayed neutron fraction ($\beta_{\text{eff}}$) for TRIGA Mark-II reactor of Morocco is about of 0.00742, as shown in Table 2. In addition, the $\beta_{\text{eff}}$ value in the principal design data for 2 MW TRIGA MARK-II reactor of Morocco equals 0.00730 [2]. There is a good agreement with the calculated value of $\beta_{\text{eff}}$ and the reference value which was mentioned above (0.00730). The relative difference between the obtained value of $\beta_{\text{eff}}$ in the Moroccan reactor and the reference values of $\beta_{\text{eff}}$ in the other TRIGA reactors is mainly due to various components of the Moroccan TRIGA core and the nuclear data library used in our calculations.

From Table 3, the MCNP calculated values of the Prompt removal lifetime $\tau_r$ and the mean neutron generation times $\Lambda$ for TRIGA of Morocco (8.5 wt. %) equal 43.91$\mu$s and 43.9$\mu$s respectively. In the reference data, the mean neutron generation time value is varied from a reactor to the other one. This value varying from 28$\mu$s to 48$\mu$s for 30 wt. % or 8.5 wt.% uranium content, respectively [8]. At that point, it can be observed that the obtained values of the $\Lambda$ and $\tau_r$ for the Moroccan TRIGA reactor have a good agreement with the reference values. The difference observed between the $\tau_r$, $\Lambda$ values of the Moroccan TRIGA and the values of the other reactors is mainly due to the enrichment of uranium 235, the number of fuel pins in the core and the nuclear data used in calculation.

4. Conclusion

The aim of this work was to calculate the kinetic safety parameters of the Moroccan TRIGA reactor based on MCNP5 code. In this study, the 3-D model of the Moroccan TRIGA MARK II research reactor was simulated using the MCNP5, the reactor geometry and material composition were taken with great precision. The kinetic parameters such as the effective delayed neutron fraction ($\beta_{\text{eff}}$), the Prompt removal lifetime $\tau_r$ and the mean neutron generation time $\Lambda$ were calculated and compared with the reference values. The simulation results of the $\beta_{\text{eff}}$, $\tau_r$ and $\Lambda$ show a good agreement with the reference data measured during the TRIGA reactor. From this study it was observed that the calculated values of these parameters vary from a reactor to the other one, with a relative difference. Consequently, the knowledge of the reactor geometry and its components are recommended for every calculation of kinetic parameters in TRIGA nuclear reactor, in order to ensure the safety and efficient working of the reactor.
References


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