Neutronic Analysis of Moroccan TRIGA MARK-II Research Reactor using the DRAGON.v5 and TRIVAC.v5 codes

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Abstract

The two-dimensional core neutronic analysis of the Moroccan TRIGA MARK-II research reactor using the DRAGON.v5 code allows us to obtain all databases that will be used in calculations of full TRIGA core in using TRIVAC.v5 code. In this study, the burnup calculations of two-dimensional fuel element and two-dimensional TRIGA core were performed using DRAGON.v5 code, and effective multiplication factor was calculated for full core via TRIVAC.v5 code. The obtained results indicate that the DRAGON.v5 and TRIVAC.v5 codes are capable of neutronic analysis of TRIGA MARK-II research reactor, provided that appropriate treatments are applied in both fuel element and core levels.

Keywords: Triga Mark-II core, Neutronic analysis, DRAGON/TRIVAC versions 5 codes

1. Introduction

The TRIGA MARK-II research reactor is only the nuclear research reactor in Morocco that achieved initial criticality on May 2, 2007. This reactor has been constructed for manpower training, production of isotopes, neutron activation analysis and studies of various fields of nuclear research.

In Morocco, The several studies of neutronic parameters on the TRIGA Mark-II reactor have been performed via the Monte Carlo code or combination of codes like internally developed burnup code called BUCAL1 [1], while uses of the deterministic codes for analysis of these parameters are almost neglected. To this end, we carried out a preliminary study on our research reactor using the DRAGON.v5 code for exploiting the obtained databases in calculations of these parameters in using the DONGON/TRIVAC codes. The DRAGON/DONJON codes have already used for the burnup-dependent calculations of both assembly level and the full core level of the JRR-3M plate-type research reactor [2]. In this paper, the burnup calculations of 2D standard fuel element and 2D TRIGA core using DRAGON.v5 code and the keff calculation of TRIVAC.v5 are based on hexagonal configuration.

The goal of this research is to simulate the core of Triga Mark II research reactor using DRAGON.v5/TRIVAC.v5 codes in order to calculate the neutronic parameters. Moreover, the second objective is to examine also the applicability of DRAGON.v5/TRIVAC.v5 for the analysis of Moroccan Triga Mark-II research reactor with the characteristics of hexagonal geometry and highly heterogeneity.

In this paper, DRAGON.v5 simulation of two-dimensional fuel element and twodimensional TRIGA code is given in section 2. The obtained results are introduced in sections 3 and 4, and the conclusions are presented in section 5.

2. DRAGON.v5 simulation of the Moroccan Triga Mark-II research reactor

The DRAGON lattice code is widely used to generate few-group macroscopic properties for reactor core calculations [3]. DRAGON is divided into many calculation modules lined together by the GAN generalized driver [4]. In the first part, we present the 2D geometries simulation of TRIGA MARK-II research reactor. Then we will introduce the calculation options used in DRAGON.v5.

2.1. Geometries simulation

TRIGA MARK-II is a light water moderated and cooled, graphite reflected research reactor with nominal power of 2 MW [5]. Fig.1 shows the results given by DRAGON.v5 simulation of 2D Triga Mark-II core configuration with the control rods inserted and removed. In all cases, the averaged fuel temperature was set to 600 °K and 300 °K for the rest of structures (cladding, graphite elements, moderator, etc.).

The TRIGA core is composed of 96 standard fuel elements, 5 fuel follower control rods, 17 pieces of graphite reflector, 1 central thimble and 1 pneumatic transfer system irradiation terminus. The water-filled aluminum tank is installed around the core. This 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. The H/Zr atom ratio is approximately 1.65. The physical properties of standard fuel and fuel follower elements are given in Table 1 [6]. The 2D standard fuel element geometry of the TRIGA Mark-II reactor is represented in Fig.2.





Fig.1. 2D core configuration of the Moroccan TRIGA reactor via DRAGON.v5 code



Fig.2. DRAGON.v5 simulation of 2D the standard fuel element geometry

| Table 1. Geometr | y and material | composition | data of sta | ndard fuel | and fuel | follower elements |
|------------------|----------------|-------------|-------------|------------|----------|-------------------|
|------------------|----------------|-------------|-------------|------------|----------|-------------------|

| | Fuel element (cm) | Fuel follower (cm) |
|--------------------------------|-------------------|--------------------|
| Outer diameter | 3.76 | 3.44 |
| Fuel diameter | 3.65 | 3.33 |
| Cladding thickness | 0.0508 | 0.0508 |
| Diameter of zirconium rod | 0.6350 | 0.6350 |
| Amount of uranium U-ZrH (wt %) | | 8.5 |

2.2. Calculation options

The calculations here are performed in DRAGON Version 5 using the ENDFB-VII.0 U.S. Evaluated Nuclear Data Library, issued 2006. 172 energy groups are chosen for all calculations. The self-shielding calculations were performed using the SHI module based on the generalized Stammler method with Livolant and Jeanpierre (LJ) option. The NXT module which is based on the collision probability method was used to solve the neutron transport equation. The EVO module was used to perform the burnup calculations. Moreover, the other modules (LIB, GEO, ASM, FLU, and EDI) were also used. Finally, the PSP module to generate PostScript images for 2D geometries was used. In both cases, the reflective boundary conditions are used.

For MCNP5 simulations, 10000 particles per generation and the same Nuclear Data Library as DRAGON.v5 are used in all cases.

3. Results and discussion

3.1. 2D standard fuel element depletion case

The configuration of 2D standard fuel element is presented in Fig.2.The kinf values given by MCNP5 and DRAGON are shown in Table 2. The results of MCNP5 and DRAGON.v5 are consistent with relative error which is in the order of -0.345 %. It should be

noticed that there is zero xenon and zero samarium poisoning in the initial fuel. The isotopic densities of 235 U, 236 U, 236 U, 239 Pu, 241 Pu, 240 Pu, 149 Sm and 135 Xe in fuel element are shown in Fig. 6. The macroscopic cross-sections variations as a function of group energies at the beginning (BOC) and at the end (EOC) of the burnup cycle are shown in Fig. 7. Figures 4 and 5 represent Neutron spectrum at the beginning and at the end of the burnup cycle and neutron flux distribution respectively.

After the 2D fuel element critical calculation, the depletion calculation is performed by DRAGON.v5. The kinf evolutions are shown in Fig. 3. At the beginning, there is the typical steep gradient on the kinf, and then the evolution is smoother. The burnup reach 107675.5 MWd/tU when the kinf is below 1.

| Code | kinf | | Relative error % |
|-----------|---------|---------------|-------------------------|
| MCNP5 | 1.39258 | ± 0.00022 | |
| DRAGON.v5 | 1.38778 | | -0.345 |





Fig. 3. 2D fuel element kinf evolutions of DRAGON.v5

Fig. 4. Neutron spectrum at the beginning and at the end of the burnup cycle







Fig. 5. Neutron flux distribution



Fig. 6. Isotopes atomic densities evolutions of DRAGON.v5



Fig. 7. Macroscopic cross-sections as a function of group energies at the beginning and at the end of the burnup cycle

3.2. 2D TRGA core burnup case

The configuration is shown as Fig. 1. In the case where all 5 fuel follower control rods are fully removed, the infinite multiplication factor for the 2D TRIGA core was calculated by DRAGON.v5 at 2 MW reactor power and was found to be 1.34553, in comparison to the MCNP5 calculated kinf of 1.32401 ± 0.00025 with relative error which is in the order of 1.625%. Moreover, the variation of the kinf as a function of number of control rods inserted in core was studied (see Fig. 9). The neutron spectrum at the beginning (BOC) and at the end (EOC) of the burnup cycle and neutron flux distribution are respectively represented in Figs. 10 and 11. To take account of the self-shielding of elements produced in reactor core, the 2D TRIGA core burnup is calculated as a function of kinf as shown in Fig. 8. The burnup reach 97187.47 MW*d/tU when the kinf is near 1.0.



Fig. 8. 2D TRIGA core kinf evolutions of DRAGON.v5



Fig. 9. kinf evolutions depending on the number of inserted control rods in TRIGA Core



Fig.10. Neutron spectrum at the beginning and at the end of the burnup cycle



Fig. 11. Neutron flux distribution

4. Simulation of full TRIGA MARK-II core of TRIVAC.v5

For the full core calculations, the TRIVAC Version 5 code [7] is used to compute effective multiplication factor. The void boundary condition in the z-axis and HBC COMPLETE REFL are used. The Simplified spherical harmonics transport theory (SPn) is adopted in this calculation with n=5. The tracking module TRIVAT is also used. The TRIVAA module is necessary to calculate the finite element system matrices. The OUT module is used to compute the reaction rates. Finally, the other modules (MAC, GEO, and FLUD) are used. Four group macroscopic cross sections and diffusion constants are read from the input data file using REDLEC. Four groups are selected to condense the four group macroscopic cross-sections. The energy of first group is from 1.9640E+7 eV to 9.0718E+5 eV, the energy of second group is from 9.0718E+5 eV to 9.9600E-1 eV, the energy of third group is from 9.9600E-1eV to 6.25E-1, and the energy of fourth group is from 6.25E-1 eV to 1.00E-5 eV. The obtained keff value of TRIVAC.v5 is 1.03363 in comparison to the Monte Carlo Code MCNP keff of 1.04975 [8] with relative error which is in the order of -1.536%.

5. Conclusions

The DRAGON.v5 code is applied to the neutronics calculation of the 2D TRIGA MARK-II core. The kinf values of DRAGON.v5 have been verified by Monte Carlo code MCNP5. In this paper, the obtained results show the capacity of neutronic analysis of DRAGON.v5 for the TRIGA MARK-II core. In addition, these databases will be exploited for calculating all neutronic parameters of full Moroccan TRIGA MARK-II research reactor core using deterministic codes in particular DONJON/TRIVAC codes. The kinf from DRAGON.v5 and keff from TRIVAC.v5 agree with the Monte Carlo Code MCNP simulations.

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