The Evolution of the Neutronics Parameters versus Burnup for the Moroccan TRIGA Research Reactor

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This work presents the results of the burn up calculation of the Moroccan TRIGA Mark II research reactor at Centre d'Etudes Nucléaire de la Maâmora (CENM). The fuel cycle length and the changes in several core parameters such as core reactivity, flux, control rods position, depletion of ^{235}U and production of ^{239}Pu and other parameters are estimated. Burn up calculations were done using BUCAL1 computer code based on MCNP5 code. For this purpose, we have used the ENDF/B-VII evaluated neutron reaction data recently released from the Brookhaven National Laboratory (BNL). The processing of the ENDF/B-VII evaluation into library suitable for use with the MCNP code was done using the modular system NJOY99. Besides, the study gives valuable insight into the behavior of the reactor and will ensure better utilization and operation of the reactor, also it will allow planning of strategies for fuel reshuffling and/or reloading schemes and its safe implementation.