Nuclear Data Induced Uncertainty in the ALLEGRO-MOX Burnup Calculation

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Abstract. To license the advanced gas-cooled fast reactor design, sufficient data with associated uncertainties should be known about its safety and operation capability. The relevant core parameters intuitively depend on the actual fuel isotopic composition which develops during a reactor operation. In practice, the prediction accuracy of burnup calculations serves as the basis for the future precise estimation of a core lifetime and other safety-based core characteristics. The present study quantifies nuclear data induced uncertainties of nuclide concentrations and multiplication factors in GFR 2400 demonstrator ALLEGRO utilizing SCALE system and the TRITON sequence are used with the KENO stochastic solver. The super-sequence SAMPLER module that implements stochastic techniques is used to assess the uncertainty in computed results. The propagation of uncertainties in neutron cross-section and fission yields is studied through the depletion calculation of 2D heterogeneous ALLEGRO-MOX fuel, where totally 250 cases with uncertain parameters are computed and the results are evaluated by an auxiliary tool.