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Neutronic Calculation of CEFR Core using Different Nuclear Data Libraries

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The uncertainities of evaluated nuclear data represent one of the most important sources of uncertainity in the reactor physics simulation. The improvement of these data used is requiered for the development, safety assesment and licensing process of a reactor. Is generally recognised the need for further investigation (experimental included) regarding the uncertainities on some main cross-section (e. g. 238U, 242Pu, minor actinides etc.).

The paper deals with the investigation of keff discrepancies induced by the differences among the cross-sections from ENDF/B-VIII.0, JEFF-3.3 and JENDL-4.0 libraries. For this study, a benchmark neutronic calculations for the first criticality of China Experimental Fast Reactor core configuration have been performed using the Continuous-energy Monte Carlo Reactor Physics Burn-up Calculation Code - SERPENT 2, version 2.1.31. The reactor reached the first criticality for a load of 72 fuel subassemblies at cold state (250℃±5℃) with only one regulating rod inserted at a certain position; all other control rods have been withdrawn out-of-core position.

The results of SERPENT 2 code show a relatively large variation in the keff values obtained with different libraries, as following: ENDF/B-VIII.0 library yields excess reactivity of 98 pcm while JEFF3.3 and JENDL-4.0 yield excess reactivity of 243 pcm and 627 pcm, respectively.

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