

Study on the Method of Correction of Fast Reactor Power Distribution by MCNP

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The tally cards F6 and F7 in MCNP program allow users to calculate reactor power. After a time of operation, the fission products increased, which caused the delayed energy in the reactor. Thus, the power directly calculated by F6 and F7 would not correspond with the real value, and for the fast reactor, the energy distribution of fuel and other structural materials will also deviate from the actual value. In this paper, to obtain more accuracy of core power distribution by the MCNP power tally cards, the first core of China Experimental Fast Reactor (CEFR) is taken as an example, the distribution of the neutron energy, prompt γ energy, delayed γ energy and delayed β energy are calculated. The delayed γ energy and delayed β energy which cannot be calculated directly have been corrected. The delayed β is regarded as deposited in the fuel area, while delayed γ would transport in the whole reactor range. A source of body type of delayed γ is described, the heat release distribution is evaluated with the effect of delayed γ energy.

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