Development of Burn-up Calculation System for Fusion-Fission Hybrid Reactor

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A fusion-fission hybrid reactor which is a fusion reactor with a blanket region containing nuclear fuel. It is expected to be realized earlier than genuine fusion reactor because even for a relatively lower plasma condition, neutrons can be well multiplied by fission in the nuclear fuel and tritium is thus bred so as to attain its self-sufficiency. And then, enough energy multiplication is expected and moreover transmutation of nuclear waste is possible. In our group, a calculation system for analysis of fusion-fission hybrid reactor has been developed and various transport and burnup calculations were carried out for hybrid systems with three-dimensional ITER model.

The burnup calculation system consists of a general-purpose Monte Carlo code MCNP-4B, a point burnup code ORIGEN2 and some other postprocessing codes developed by us, which process the calculated results. The calculation procedure of one burnup cycle consists of criticality calculation, neutron transport calculation, making of collapsed cross section and burnup calculation. In order to evaluate collapsed cross section for burnup as precise as possible, making of collapsed cross section process was postprocessed using not tally function of MCNP but neutron tracklength data of the MCNP calculation directly. We are using a modified version of MCNP-4B so as to output all neutron tracklength data in the blanket cell. JENDL-3.3 pointwise data and JENDL Activation Cross Section File 96 were used as base cross section libraries.

The details of the calculation system for analysis of fusion-fission hybrid reactor are to be presented in the symposium especially concentrating on making of collapsed cross section for ORIGEN2 together with some burnup calculation results of Japanese possible fusion fission hybrid reactor.