BURNUP CALCULATIONS BY COMBINED MONTE CARLO AND LINEAR CHAINS METHODS

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The objective of burnup calculations is to follow the time development of core parameters of a nuclear reactor. These include nuclide inventory, effective multiplication factor, neutron flux, power distribution and reactivity coefficients, among others. Such calculations are needed all the way from very front end analyses of new reactor concepts to adjusting and optimization of power reactors.

Burnup calculations require solving the time development of both neutronics and material compositions. Neutron energy spectrum and flux in the reactor determine transmutation rates of materials, and materials determine the neutron spectrum. Material composition changes slowly, however, and therefore the coupled problem can be solved by updating the spectrum only at certain points in time, and considering it constant between them.

A test program, DeTraBurns, was developed in this work. DeTraBurns performs burnup calculations by linking the pre-existing programs MCNP [1] and DeTra [2]. MCNP solves neutronics using the Monte Carlo method. Neutrons are simulated using random numbers and their statistical behavior is used to calculate desired values. DeTra solves the Bateman equations governing nuclide decay and transmutation chains using the linear chains method, where the complex web of possible decay and transmutation trajectories is decomposed to a number of linear ones, which are solved analytically. DeTraBurns handles the information flow between the two programs. It also handles the parts which the other programs can’t, for example fission yields and the total flux.

The main objective of this work was to get insight to and experience of such calculations. DeTra had previously not been used in reactor conditions. It was modified to be able to handle burnup calculations and the modifications were included as a new feature to DeTra. Results from DeTraBurns are within a few percent of those calculated with monteburns [3], but take multiple times longer to obtain because of inefficient use of MCNP. The reasons and solutions to inaccuracy and poor performance were identified but the development was aborted, as enough had been learned and there already exist working programs using the same method.

[1] X-5 Monte Carlo Team, LANL, USA.
MCNP-A General Monte Carlo N-Particle Transport Code - Version 5