

Background

The incentive of creating Monte Carlo burn-up codes arises from its ability to provide the most accurate locally dependent spectra and flux values in realistic 3D geometries of any type. These capabilities linked with the ability to handle nuclear data not only in its most basic but also most complex form (namely continuous energy cross sections, detailed energy-angle correlations, multi-particle physics, etc.) could make Monte Carlo burn-up codes very powerful, especially for hybrid and advanced nuclear systems (like for instance Accelerator Driven Systems).

Still, such Monte Carlo burn-up codes have had limited success mainly due to the rather long CPU time required to carry out very detailed and accurate calculations, even with modern computer technology. To work around this issue, users often have to reduce the number of nuclides in the evolution chains or to consider either longer irradiation time steps and/or larger spatial burn-up cells, jeopardizing the accuracy of the calculation in all cases.

There should always be a balance between accuracy and what is (reasonably) achievable. So when the Monte Carlo simulation time is as low as possible and if calculating the cross sections and flux values required for the depletion calculation takes little or no extra time compared to this simulation time, then we can actually be as accurate as we want. That is the optimum situation for Monte Carlo burn-up calculations.

Objectives

The ultimate goal of this work is to provide the Monte Carlo community with an efficient, flexible and easy to use alternative for Monte Carlo burn-up and activation calculations, which is what we did with ALEPH. ALEPH is a Monte Carlo burn-up code that uses ORIGEN 2.2 as a depletion module and any version of MCNP or MCNPX as the transport module. For now, ALEPH has been limited to updating microscopic cross section data only.

By providing an easy to understand user interface, we also take away the burden from the user. For the user, it is as if he is running a simple MCNP(X) problem but with some “extra” options. The code has many features and capabilities, such as the possibility to simulate moving control elements and modelling the variation of boron concentration in the coolant during the irradiation.

Principal results

The efficiency is achieved in ALEPH by using the multi-group binning approach which allows the code to perform burn-up calculations in a matter of days or weeks where other Monte Carlo codes would need weeks or even years to do exactly the same thing without loss of accuracy.

By adopting the multi-group binning approach, it is as if we only have to calculate one average reaction cross section for the decay chains to know all of them. The time increase due to average reaction cross section calculation is therefore constant, regardless of the number of average cross sections. This allows us to achieve maximum accuracy for a minimum of CPU time because we can actually calculate as much reaction rates as we want *by default*.

ALEPH has already been applied to various applications including fuel cycle studies, single PWR fuel pins, fuel assemblies and even full 3D reactor calculations. The code can for instance be used to study the self-shielding effect in a nuclear reactor.

The self-shielding effect in a nuclear reactor is both energetic and spatial in nature. The energy self-shielding results from the resonances in the various cross sections of the different nuclides present in the fuel. The spatial self-shielding effect results from the heterogeneity of a reactor. Neutron absorption in the outside of a fuel pellet will effectively shield the inside of the pellet, resulting in a neutron flux dip in the centre of the fuel pin. This is strongly influenced by the neutron spectrum of the reactor in question. As a result of this effect, the burn-up at the edge of the fuel pellet will be larger than the burn-up in the centre. The distribution of fission products and other nuclides like Pu will also be dependent upon the radial position in the fuel pellet. At higher burn-up levels, this will have a significant effect on the pellet thermal properties, fission gas retention, etc. and therefore on the fuel performance itself.

The radial burn-up and Pu distribution of a high burn-up UO₂ PWR fuel pin (at burn-up levels of 95 and 102 MWd) is given as an example in figure 1. It shows a comparison of calculations by ALEPH (red curve), the TUBRNP module (blue curve) of the TRANSURANUS fuel performance code (ITU, Germany) and EPMA measurements (green dots).

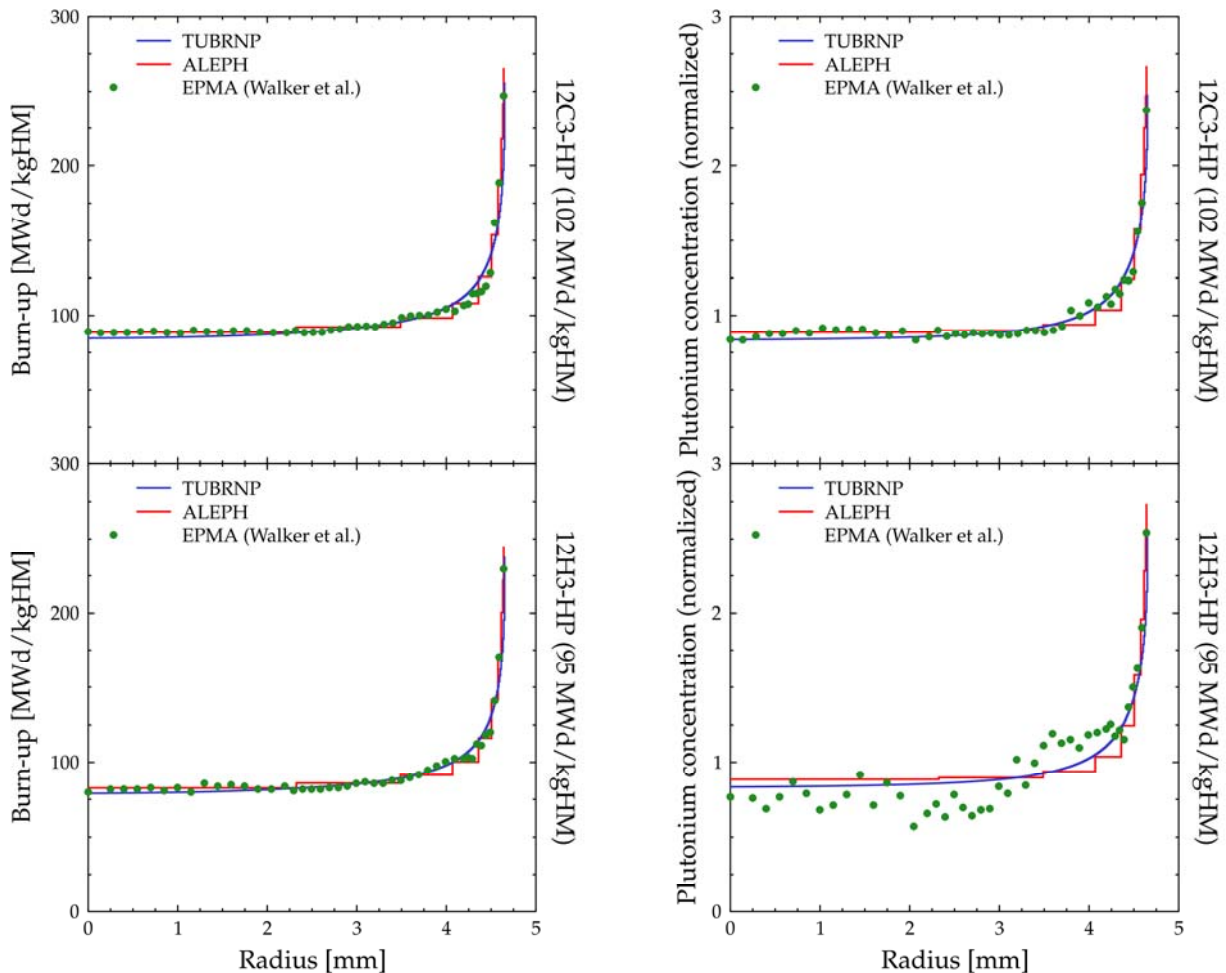


Figure 1: Radial burn-up (left) and Pu distribution in a high burn-up PWR UO_2 fuel pin.

There is good agreement of the calculated profiles with the EPMA measurements, except for the Pu profiles of the 12H3 sample which exhibit large fluctuations. These fluctuations do not appear in the Pu profiles of the 12C3 sample, although some minor fluctuations can still be discerned.

Despite assumptions on the irradiation history and irradiation conditions that had to be made for the ALEPH computations, the radial profiles are well predicted even with the simple geometrical model used. A sensitivity study of the irradiation history has been carried out with ALEPH and revealed no significant differences in our results. Slight variations in the irradiation history or even simplifications (by using an overall average power level for the entire irradiation history) resulted in very similar radial profiles.

Future developments

ALEPH is already quite capable of performing highly accurate depletion calculations in an acceptable amount of time, but there is still some room for improvement. For now, ALEPH has been limited to updating microscopic cross section data only. Other data like isomer production branching ratios, fission yields, etc. have been left unchanged in ORIGEN 2.2.

Another possible upgrade to ALEPH would be to replace ORIGEN 2.2 by a built-in depletion module to address some of the limitations of ORIGEN 2.2. This includes the use of more reaction channels (like (n,d), (n,t), etc.), an extension of the number of actinides with direct fission yields and an extension of the decay channels. Such an upgrade of the depletion module would also allow the implementation of a better treatment for the total recoverable energy per fission.

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Main reference

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