Assessment of the MCNP+ACAB code system for burnup credit analyses.

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An automated tool for depletion and sensitivity/uncertainty analysis in BUC calculations is presented. This system combines the Monte Carlo neutron transport code MCNP and the inventory code ACAB as a suitable tool for high burnup calculations.

The potential impact of nuclear data uncertainties on some response parameters (decay heat, neutron emission, radiotoxicity, keff, …) may be large, but only very few codes can treat this effect. The uncertainty analysis methodology implemented in the ACAB code, including both the sensitivity-uncertainty method and the uncertainty analysis by the Monte Carlo technique, enables to assess the impact of neutron cross section uncertainties on the inventory and other inventory-related responses in high burnup applications.

A well referenced high burnup pin-cell benchmark exercise is used to test the MCNP–ACAB performance in inventory predictions. In addition, the potential of our system, including the uncertainty capability, is demonstrated. It is proved that the inclusion of ACAB in the system allows obtaining results that are at least as reliable as those obtained using other inventory codes. We estimate the errors due to activation cross section uncertainties in the prediction of the isotopic content up to the high-burnup spent fuel regime. The most relevant uncertainties are highlighted, and some of the most contributing cross sections to those uncertainties are identified.