

DETAILED BURNUP CALCULATIONS FOR TESTING NUCLEAR DATA

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A general method (MCQ) has been developed by introducing a microscopic burn up scheme which uses the Monte Carlo calculated fluxes and microscopic reaction rates of a complex system and a depletion code for burnup calculations, as a basis for solving nuclide material balance equations for each spatial region in which the system is divided.

Continuous Energy dependent cross section libraries and full 3D geometry of the system can be input for the calculations. The resulting predictions for the system at successive burn up time steps are thus based on a calculation route where both geometry and cross sections are accurately represented, without geometry simplifications and with continuous energy data, providing an independent approach for benchmarking other methods and nuclear data of actinides, fission products and other burnable absorbers.

The main advantage of this method over the classical deterministic methods currently used is that MCQ System is a direct 3D method without the limitations and errors introduced on the homogenization of geometry and condensation of energy of deterministic methods.

The Monte Carlo and burn up codes adopted until now are the widely used MCNP and ORIGEN codes, but other codes can be used also. For using this method, there is a need of a well known set of nuclear data for isotopes involved in burnup chains, including burnable poisons, fission products and actinides.

For fixing the data to be included on this set, a study of the present status of nuclear data is performed, as part of the development of MCQ method. This study begin with a review of the available cross section data of isotopes involved in burnup chains for power and research nuclear reactors. The main data needs for burnup calculations are neutron cross sections, decay constants, branching ratios, fission energy and yields.

The present work includes results of selected experimental benchmarks and conclusions about the sensitivity of different sets of cross section data for burnup calculations, using some of the main available evaluated nuclear data files (ENDF-B-VI-Rel.8, JEF-3, JENDL-3.3), on an isotope by isotope basis as much as possible.

The selected experimental burnup benchmarks are reference cases for LWR and HWR reactors, with analysis of isotopic composition as a function of burnup. For LWR's (H₂O-moderated uranium oxide lattices) four benchmarks are included: ATM-104 NEA Burnup credit criticality benchmark; YANKEE-ROWE CORE V; H.B.ROBINSON UNIT 2 and TURKEY POINT UNIT 3. For HWR's (D₂O-moderated uranium oxide cluster lattices), three benchmarks were selected: NPD-19-rod Fuel Clusters; PICKERING-28-rod Fuel Clusters and BRUCE-37-rod Fuel Clusters. The isotopes with experimental concentration data included on these benchmarks are: Se-79, Sr90, Tc99, Ru106, Sn126, Sb125, I129, Cs133-137, Nd143, 145, Sm149-150, 152, Eu153-155, U234-235, 238, Np237, Pu238-242, Am241-243 and Cm242-248.

Results and analysis of differences between calculated and measured absolute and/or relative concentrations of these isotopes for the 7 benchmarks are included on this work.