Helium and tritium in neutron-irradiated beryllium

Abstract

Beryllium is considered as a potential neutron multiplier in a Helium Cooled Pebble Bed tritium breeding blanket for future fusion power reactors. Under neutron irradiation, helium and tritium are produced in beryllium. The formation of helium bubbles induces swelling; tritium retention is a safety and waste handling issue. In-pile gas release should be sufficiently high to avoid the evacuation of the plant site in case of a serious accident leading to the abrupt release of all accumulated tritium.

A reliable prediction of the behavior of helium and tritium in beryllium, in-pile and during out-of-pile fast temperature transients, is necessary in order to prove the attractiveness of the blanket concept and to optimize design and materials. The lack of experimental data for beryllium pebbles, in the range of neutron fluence and temperature typical of the blanket module operation, imposes an extrapolation of models outside their validation range. A more sophisticated gas kinetics model and a more detailed validation of its single parts are necessary for beryllium in a fusion reactor blanket, than for uranium oxide in fission reactors. Since 1992 the code ANFIBE has been developed to predict the behavior of helium and tritium in neutron-irradiated beryllium. The aim of the present work is to improve the code for both theoretical modelling and experimental validation, in order to increase confidence in its extrapolation to fusion reactor conditions. This requires to produce a more detailed, comprehensive and relevant experimental database than the one which was available during the early development phase of the code.

The following milestones have been reached: (1) experimental characterization of all helium and tritium diffusion and release stages in neutron-irradiated beryllium, also from a microscopic point of view; (2) assessment of helium and tritium thermal diffusion coefficients; (3) improvement of the model for gas precipitation into bubbles on the basis of the experimental study; (4) definition and application of an integrated validation procedure for the analytical model, based on the changes in the material microstructure related to different gas release stages. The final result of this study is a new version of the ANFIBE code, which can better describe gas atomic diffusion and precipitation into bubbles and the corresponding gas release. The code has then been applied to approximately assess tritium retention in beryllium at the End-Of-Life of a blanket module in a fusion reactor of 1.5 GW electric power. On the basis of such assessment, tritium retention in beryllium appears to be a much less critical issue than it was believed in the past.