Overview of Long-term Fuel Inventory and Co-deposition in Castellated Beryllium Limiters at JET

M. Rubel\textsuperscript{a}, J.P. Coad\textsuperscript{b} and D.E. Hole\textsuperscript{c}

\textsuperscript{a}Alfvén Laboratory, Royal Institute of Technology, Teknikringen, 33, S-10044 Stockholm, Sweden

\textsuperscript{b}Culham Science Centre, EURATOM-UKAEA Fusion Association, Abingdon, OX14 3DB Oxon, United Kingdom

\textsuperscript{c}University of Sussex, Science and Technology, Pevensey Building, BN1 9QH Brighton, United Kingdom
rubel@kth.se

All plasma-facing components (PFC) in ITER will be castellated, i.e. composed of small blocks separated by narrow grooves (less than 1 mm) in order to reduce thermally-induced stress. The contact of plasma with several different elements (including carbon and beryllium) on the first wall, as foreseen for ITER, will also lead to the co-deposition of eroded material together with fuel species in the castellation. The main question is whether fuel retention in such places may significantly contribute or even be decisive for the overall tritium inventory in ITER where the total number of castellated grooves will exceed 1 million. This calls for detailed studies of castellated structures and gaps between PFC tiles from present-day tokamaks.

Until now, large castellated structures have been used at JET (divertor and limiters) and recently in Tore Supra (pump limiter). This contribution provides an account of the detailed examination of several castellated beryllium tiles from the belt limiter (of which approx. 2000 pieces were in total) exposed to the JET plasma for 56000 s. The major aim of the investigation carried out by means of ion beam analysis methods (nuclear reaction analysis [NRA] and enhanced proton scattering [EPS]) was to determine the fuel retention and material mixing on the tiles. Analyses have been performed on both sides of castellated grooves, on plasma-facing and side surfaces of the tiles. The essential results are summarised by the following:

- deuterium retention in the grooves is associated with co-deposition of carbon;
- decay length of deposition in the castellation is short: $\lambda \sim 1.5$ mm;
- the maximum deuterium content in the groove does not exceed $8 \times 10^{17}$ cm$^{-2}$; this is measured at the distance of approximately 2 mm from the entrance to the gap;
- no deuterium (above the detection limit of NRA technique) is detected in bulk Be;
- bridging of gaps by molten beryllium is observed, but gaps are not filled with Be;
- fuel content on plasma-facing surfaces is fairly low (in the range $1-8 \times 10^{17}$ cm$^{-2}$); it is mainly associated with carbon co-deposition
- on side surfaces of the tiles the formation of BeO layer is detected at a distance of 20 mm and more from the plasma-facing surface.

The consequences for a long-term operation of a reactor-class device with several different plasma-facing materials will be discussed.

Number of words in abstract: 376

Keywords:
Technical area: 51. Materials for plasma-facing components Armor materials
Special session: Not specified
Presentation: No preference
Special equipment: No special equipment