Contribution ID: **20** Type: **Invited Oral**

Experience with tritium retention and removal in JET-DTE2

Thursday 13 October 2022 13:40 (35 minutes)

JET is the largest tokamak in use and currently the only one capable of handling radioactive tritium (T). It operates since 2011 with the ITER-like wall (ILW), which consists of a tungsten (W) divertor and a beryllium (Be) main chamber. Following preparatory campaigns in deuterium (D), hydrogen (H) then T, JET has operated the second Deuterium Tritium Experimental campaign (DTE2, after DTE1 in 1997 with carbon based Plasma Facing Components, PFCs), aiming at answering urgent ITER needs [1].
About 1 kg Tritium has been supplied by the JET Active Gas Handling System (AGHS) during the T and DT campaigns, ran on cycles of three-four weeks of operation followed by one week of tritium reprocessing and accounting. A global gas balance was performed to assess the in-vessel T retention after one day of operation [2] by subtracting the amount of actively pumped neutrals from the amount of injected T through the gas injection systems. Long-term outgassing experiments completed the study, evidencing a faster decay of the T partial pressure compared to D. The so-determined long-term T retention will be compared with the global T accountancy by AGHS, still in progress.
After DTE2, a sequence of complementary fuel recovery methods was successfully operated to remove T from the PFCs. It consisted of baking the main chamber under vacuum at 320ºC, followed by isotopic exchange with Ion Cyclotron and Glow Discharges in D2 at this temperature, preferentially accessing T retained in the main chamber. 20 seconds long diverted plasmas with up to 16 MW NBI and ICRH power were then operated in different magnetic configurations with the main chamber at 200℃, targeting in particular the inner divertor baffle region, where the majority of the retained fuel is known to reside in thick Be deposited layers [3].
At the end of the sequence, the plasma isotopic ratio T/[H+D+T] inferred from neutron spectroscopy, was found to be ~10-4 in H-mode D plasmas, well below the 1% target set by the allocated 5·1019 14 MeV fusion neutrons budget for the following D campaigns [4]. In total, about 4·1023 T atoms were removed from JET PFCs, among which ~45% by baking and 50% by ICWC and GDC, the remaining 5% being removed afterwards from PFCs already depleted from T by limiter and diverted plasmas. Similar results had been obtained in a qualification experiment prior to the T campaign, where D was removed from PFCs using baking, as well as ICWC, GDC and plasma operated in H [5]. Though removal by baking seems to be less effective for T than for D, ICWC and GDC promote extra removal in both cases. Access to T buried in co-deposited layers at the upper part of the inner divertor was clearly evidenced from the increased neutron rate and elevated surface temperatures above 1200℃ in plasmas that had the inner strike point raised onto this area. Still, the likely re-deposition of released material has to be assessed. For this high fidelity numerical simulations are on-going using the Monte-Carlo erosion/migration code ERO2.0 [6].

Speaker's Affiliation

Institute for Magnetic Fusion Research, CEA Centre de Cadarache

Member State or IGO

France

Primary authors: Dr DOUAI, David (CEA, IRFM, Association Euratom-CEA, 13108 St Paul lez Durance, France); MATVEEV, Dmitry (Forschungszentrum Juelich); JEPU, Ionut (UK Atomic Energy Authority); BREZINSEK, Sebastijan (Forschungszentrum Jülich); Dr WAUTERS, Tom (ITER Organization)

Presenter: Dr DOUAI, David (CEA, IRFM, Association Euratom-CEA, 13108 St Paul lez Durance, France)

Session Classification: Tritium Behaviors and DEMO Fuel Cycle

Track Classification: Interface btw Plasma Physics & Fuel Cycle Technology