EFDA–JET–CP(13)01/09

S. Koivuranta, J. Likonen, A. Hakola, J.P. Coad, A. Widdowson
D.E. Hole, M. Rubel and JET EFDA contributors

Post-Mortem Measurements of Fuel Retention at JET

Post-Mortem Measurements of Fuel Retention at JET

S. Koivuranta[1], J. Likonen[1], A. Hakola[1], J. P. Coad[1], A. Widdowson[2], D.E. Hole[3], M. Rubel[4] and JET EFDA contributors*

JET-EFDA, Culham Science Centre, OX14 3DB, Abingdon, UK

[1]Association EURATOM-TEKES, VTT, P.O. Box 1000, 02044 VTT, Espoo, Finland
[2]EURATOM/CCFE Fusion Association, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK
[3]Departmentof Engineering and Design, School of Science and Technology, University of Sussex, Brighton, BN1 9QH, East Sussex, UK
[4]Alfvén Laboratory, Royal Institute of Technology, Association EURATOM-VR, 100 44 Stockholm, Sweden
** See annex of F. Romanelli et al, "Overview of JET Results", (24th IAEA Fusion Energy Conference, San Diego, USA (2012)).*

Preprint of Paper to be submitted for publication in Proceedings of the
14th International Conference on Plasma-Facing Materials and Components for Fusion Applications,
Jülich, Germany.
13th May 2013 - 17th May 2013

ABSTRACT
The deuterium inventory at Joint European Torus (JET) in co-deposits and bulk material in inner, outer and upper wall components of the vacuum vessel exposed in 2007-2009 has been determined using Nuclear Reaction Analysis (NRA) and Secondary Ion Mass Spectrometry (SIMS) and optical microscopy. Approximately 1.8g of the deuterium is retained in these areas which correspond ~0.07% of the retained deuterium.

1. INTRODUCTION

The erosion of plasma facing materials in magnetically confined fusion devices is a key issue with respect to: Component lifetime by erosion owing to the edge plasma and plasma contamination by eroded material. In addition, safe management and accounting of tritium in ITER and future fusion power reactors will be crucial for the reliable operation of the reactors. ITER will be operated with mixtures of deuterium and tritium and therefore tritium retention in plasma facing components must be as low as possible as it may lead to an unacceptable inventory of radioactive tritium.

Carbon or carbon fibre composite (CFC) has been a plasma-facing material in many fusion devices, for example in Joint European Torus (JET), because of its excellent power-handling capabilities and small radiation losses due to carbon impurities. CFC has the ability to implant large amounts of hydrogen isotopes from plasma. This is especially dangerous in the case of deuterium-tritium operation as it may lead to an unacceptable inventory of radioactive tritium. Determination of deuterium retention in plasma facing components by post-mortem analysis is crucial for the assessment of the overall fuel inventory.

The disadvantages of CFC could be compensated by using high-Z materials such as tungsten (W). The CFC tiles can be coated with a thin layer of W. The erosion yield of W is orders of magnitude below those of low-Z materials like e.g. CFC or graphite with an erosion yield of a few percent [1]. Tritium retention is a genuine problem for CFC, whereas W does not show such a strong effect to tritium retention.

In the period 2007-2009 JET operated with the MkII-HD divertor. Cross-section of the JET vessel is presented in Figure 1. CFC tiles from the shadowed area of the divertor and inner, outer and upper wall components of the JET vessel was analysed using Nuclear Reaction Analysis (NRA), Secondary Ion Mass Spectrometry (SIMS) and optical microscopy after the campaign for erosion and deposition studies. NRA analyses give quantitative deuterium/carbon ratio near the surface region (down a depth ~7μm) whereas SIMS provides information on the deuterium levels in principle all the way down to the substrate. Thickness of the co-deposited layers was determined both with SIMS and optical microscopy.

2. EXPERIMENTAL

During the 2007-2009 campaign JET was operated with plasma facing components (PFCs) made of CFC (Concept I manufactured by Dunlop Ltd). New CFC tiles were installed prior to 2007 and after the campaign, the tiles were removed for surface analysis with NRA and SIMS. The tiles were sent to VTT Technical Research Centre of Finland (VTT) for sample preparation. Core samples were taken from plasma facing surfaces of the tiles. Sampling took place in a glove box using a drill

saw to cut cylinders with a diameter of 17mm and a thickness of about 10 mm. After the sample preparation, core samples were sent to University of Sussex for NRA analysis. NRA measurements were carried out using a 2.5MeV ^{3}He beam produced by a Van de Graaff accelerator. The diameter of the ^{3}He beam was 1mm. The carbon (C), beryllium (Be) and deuterium (D) concentrations in the co-deposited layers were analysed using the NRA reactions $^{12}C(^{3}He,p)^{14}N$, $^{9}Be(^{3}He,p)^{11}B$ and $^{2}D(^{3}He,p)^{4}He$. The analysis depth in the NRA measurements for C is ~1μm, ~2μm for Be and ~7μm for D, respectively. SIMS analyses of the samples, for their part, were made with a double focusing magnetic sector instrument (VG Ionex IX-70S) at VTT. A 5keV O_2^+ primary ion beam with a current of 500nA was used and the ion beam was raster-scanned over an area of 300×430μm^2 [2]. In addition, the cross-sectional samples were prepared by cutting part of the core samples poloidally and placing them into cold mounting epoxy (Epofix by Struers). Grinding and polishing were made using a Struers Tegrasystem grinding and polishing device with a pre-programmed preparation method. The thicknesses of the co-deposits were assessed from optical microscopy images and the goal of optical microscopy is to produce information about the thicknesses of the deposited layers for comparison with SIMS results and to investigate the structure of the deposited layers.

3. RESULTS

A set of CFC tiles exposed in 2007-2009 at JET have been characterised using NRA and SIMS techniques allowing determination of D inventory under the assumption of toroidal symmetry. During the 2007-2009 campaign the total D input was ~2333g.

JET IWGL (Inner Wall Guard Limiter) tiles are installed in pairs: left and right (L and R). Position of the analysed IWGL tiles is presented in Figure 1. The general pattern of erosion/deposition on the IWGL tiles is the following: There is erosion on the left tile of the pair and deposition on the right tile at the top of the limiter, gradually changing to deposition on the left and erosion on the right tile of the pair at the bottom of the limiter.

Tile 3X11L shows four different areas: The reverse slope, which protects the edges of the tiles from the direct plasma impact, shows some deposition. Between the reverse slope and the dark deposition band there is mainly erosion. On the left side edge there is only little modifications by the plasma. Figure 2 shows the SIMS depth profile from sample 3X11L/1 which is located in the reverse slope. The depth profiles show clearly that there is a co-deposited layer which contains D, Be and Ni impurities. Tiles 3X11L and 3X11R after coring are presented in Figure 3.

There is very thin deposition on the reverse slope on Tile 3X11R. Figure 4 shows the SIMS depth profiles for sample 3X11R/1. There is a thin co-deposit with some D, but Be and Ni amounts are very small.

Tile 3X17L is completely in the deposition zone. Figure 5 shows SIMS depth profiles from sample 3X17L/1. Co-deposited layer is clearly thicker on sample 3X17L/1 than on sample 3X11L/1 in Figure 2. Total amount of D in analysed IWGL tiles is ~1.5g

There is some deposition on the OPL (Outer Poloidal Limiter) tiles near the top and centre of the limiters. The top part of the outer limiters interacts most strongly with the plasma and is likely to experience high power loads during ICRH (Ion Cyclotron Resonant Heating) heating experiments when the plasma comes closest to the outer limiters. NRA analysis showed that the D is deposited

at high levels but only at one end of the tile. The total D inventory in the analysed OPL tiles is only ~0.3g.

Upper wall tiles (OJ tiles) are in the erosion dominated area. Total amount of D in analysed OJ tiles is smaller than 0.1 g.

The total amount of retained D in measured tiles was obtained by assuming toroidal symmetry in deposition and multiplying the area of the tile segment with the thickness of the co-deposited layer obtained from SIMS and optical microscope measurements. The density of the deposited layers is assumed to be 1g/cm3 [3]. The D amounts are summarized in Table 1.

CONCLUSIONS

A set of inner, outer and upper wall tiles exposed in 2007-2009 at JET have been characterised using NRA and SIMS techniques allowing determination of D inventory under the assumption of toroidal symmetry.

The total D retention in inner, outer and upper wall tiles during 2007-2009 campaign is estimated to be ~1.8g which corresponds to retention of ~0.07%. Most of the D is co-deposited on inner wall tiles (~82%).

The total D retention in divertor tiles is estimated to be ~48g [4]. The total D retention in divertor, inner, outer and upper wall tiles during 2007-2009 campaign is estimated to be ~50g which corresponds to a retention of ~2.1 %. The majority of the retention is found on the divertor floor tiles 4 (~40 %) and 6 (~44%).

For long-term retention integrated over an experimental campaign, post-mortem analysis is a proven method to assess the D retention. In previous campaign 2001-2004 the long-term retention was ~3.7% [5]. The distribution of D in JET vessel in 2007-2009 campaign is similar to 2001-2004 campaign. The D inventory of ~2.1 % is comparable to other results obtained with post-mortem analysis. The long-term D retention fraction evaluated from integrated gas balance (~10-20 %) for various devices is larger than the D retention deduced from the post-mortem analysis of tiles. Post-mortem values are much reduced from gas balance because operational factors such as disruptions and glow discharge cleaning can reduce the D inventory whereas they are carefully avoided during gas balance measurements.

ACKNOWLEDGMENTS

This work, supported by the European Communities under the contract of Associations, was carried out within the framework of the European Fusion Development Agreement. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

REFERENCES

[1]. J. Roth, Review and Status of Physical Sputtering and Chemical Erosion of Plasma Facing Materials, in: Robert E.H. Clark and Detlev H. Reiter (Eds.), Nuclear Fusion Research: Understanding Plasma Surface Interactions, 2005, pp. 203-224.

[2]. J.P. Coad, J. Likonen, M. Rubel, et al., Nuclear Fusion **46** (2006) 350.

[3]. S. Krat, J.P. Coad, Yu. Gasparyan, et al., “Erosion and Deposition on JET Divertor and Limiter

Tiles During the Discharge Campaigns 2004-2009", Journal of Nuclear Materials (2013), http://dx.doi.org/10.1016/j.jnucmat.2013.01.158.

[4]. S. Koivuranta, J. Likonen, A. Hakola, et al., "Post-mortem measurements on fuel retention at JET in 2007-2009 experimental campaign", J. Nucl. Mater. (2013), http://dx.doi.org/10.1016/j.jnucmat.2013.01.156.

[5]. J. Likonen, J.P. Coad, D.E. Hole, et al., Journal of Nuclear Materials **390-391** (2009) 631-634

Location	Amount of D [g]
Outer wall tile OPL 8D21T	< 0.1
Outer wall tile OPL 8D13T	0.1
Outer wall tile OPL 8D9B	< 0.1
Outer wall tile OPL 8D1B	0.2
Upper wall tile OJ11	< 0.1
Upper wall tile OJ1	< 0.1
Upper wall tile OJ14	< 0.1
Upper wall tile OJ06	< 0.1
Upper wall tile OJ09	< 0.1
Inner wall tile IWGL 3X11L	0.1
Inner wall tile IWGL 3X11R	0.3
Inner wall tile IWGL 3X17L	0.8
Inner wall tile IWGL 3X17R	0.2

Table 1: Amounts of D co-deposited in different areas of JET

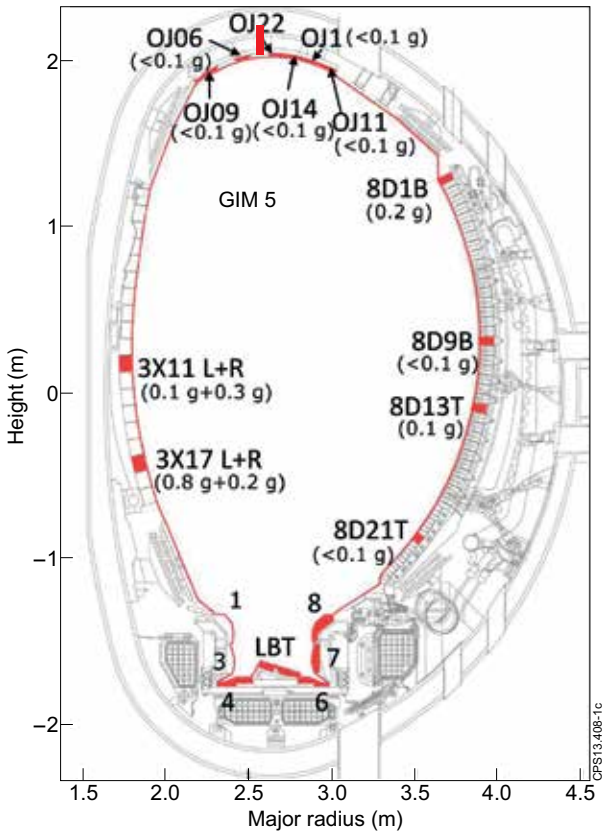

Figure 1: Cross-section of the JET vessel and the D amounts in the inner, outer and upper wall tiles.

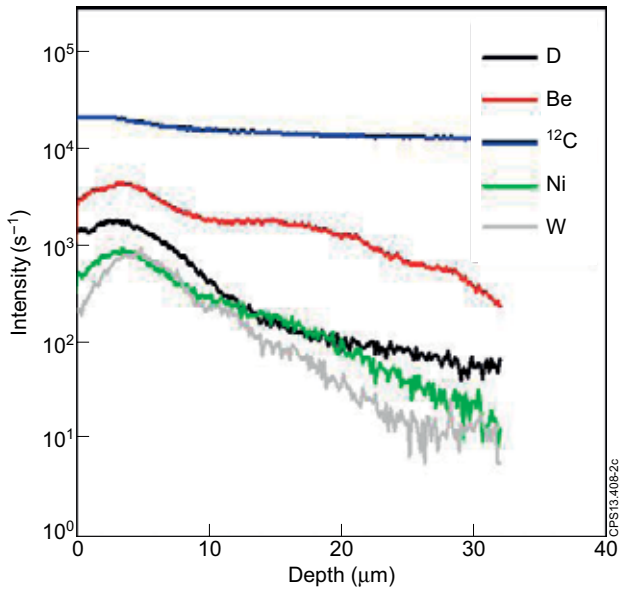

Figure 2: SIMS depth profiles from sample 3X11L/1.

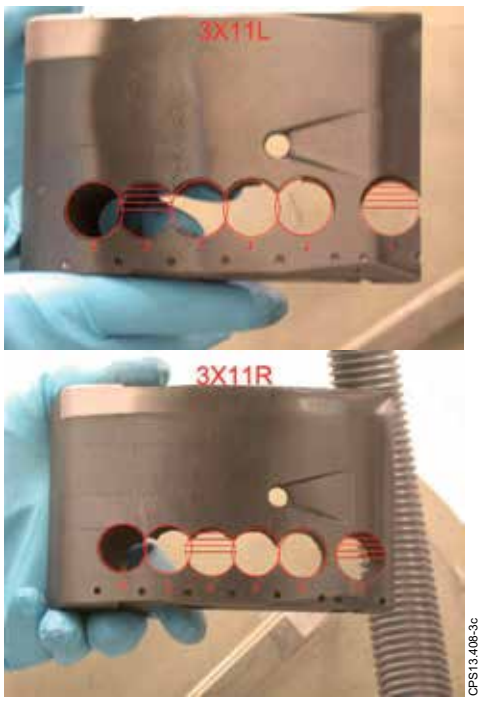

Figure 3. Tiles 3X11L and 3X11R after coring.

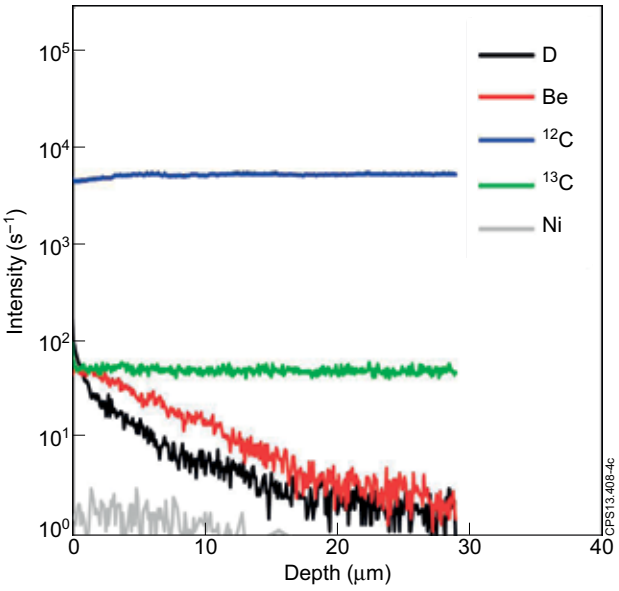

Figure 4: SIMS depth profiles from sample 3X11R/1.

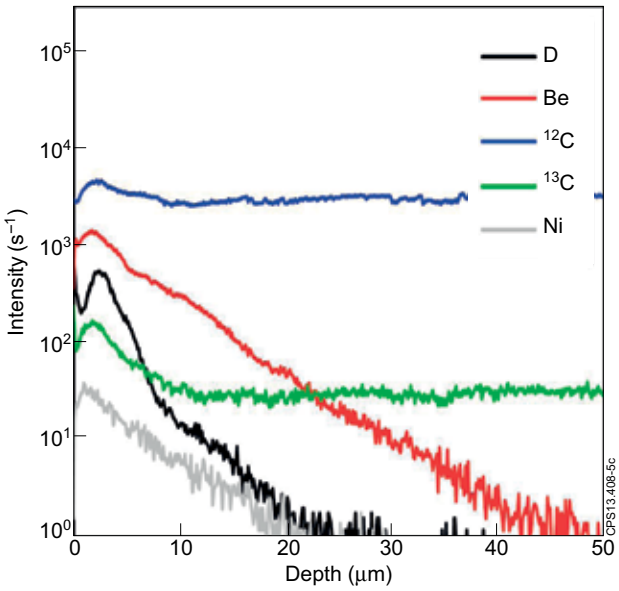

Figure 5: SIMS depth profiles from sample 3X17L/1.