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ABSTRACT.
The Joint European Torus (JET) fusion machine is the only device capable of operation with tritium and Be, what makes it best suited to study ITER relevant issues. A large variety of activities are performed within the JET Fusion Technology Task Force. In this paper, some topics such as erosion/deposition & material transport, flakes characterization and detritiation techniques are highlighted. Recent results obtained using a pumping cryopanel and on plasma facing component characterisation are given. Finally, issues that will be addressed in the forthcoming JET work-programme are presented, such as a beryllium main wall for JET and in-situ laser detritiation.

INTRODUCTION

Since 2000, the JET fusion machine is operated in the frame of European Fusion Development Agreement. It is the only world fusion device licensed for operating with tritium fuelling and beryllium facing the plasma. A Task Force on Fusion Technology (TF-FT) was created aiming at answering ITER relevant technological issues, making use of JET facilities and of the related operating experience.

Trapping of hydrogen isotopes in Plasma Facing Components (PFC), erosion/deposition, flakes formation and characterisation and particle transport have been addressed in successive TF-FT workprogrammes. Moreover, promising results have been obtained in the investigations of in-situ flash lamp and laser detritiation techniques.

On the other side, JET facilities have been used for some years to test new ITER relevant hardware. As an example, the prototype of pumping cryopanels was in operation during the last Trace Tritium Experiment.

In the following, obtained results will be summarised, together with the issues remaining open for future workprogramme.

1. EROSION/DEPOSITION AND MATERIAL TRANSPORT

1.1. BERYLLIUM ACCUMULATION AT THE INNER DIVERTOR AT JET

The MKIIGB divertor tiles exposed in JET for the 1998-2001 campaigns have been used to assess the amount of beryllium and carbon deposited at the inner divertor walls. A set of samples each of 17mm diameter was cut from the divertor tiles using a coring technique [1]. The poloïdal positions of the samples are shown in figure 1. SIMS depth profiling has been made from a number of samples on inner divertor tiles 1, 3 and 4.

On tiles 1 and 3, the deposit forms two different layers. The outer layers (~ 2-6 μm thick on tile 1 and 10-16 μm on tile 3) contain mostly carbon together with deuterium and smaller amount of beryllium. The films underneath the surface layer are very rich in beryllium (thickness of 2 to 14 μm on tile 1 and of 12 to 21 μm on tile 3). During some operation phase at the end of the campaign (e.g. the last three months at lower vessel temperature, or plasma operation in helium) chemical erosion was reduced at the inner divertor.

The amount of Be on the tiles 1 and 3 was estimated. Assuming uniform concentrations toroidally, the total amount of beryllium deposited during the 1999-2001 campaign at the inner divertor is 22 g with an uncertainty level of +/- 40%, which does not include possible toroidal asymmetry.

In contrast to this result, the film in the shadowed region on tile 4 is pure carbon, with a very high deuterium content, and with a well-marked interface to the CFC substrate. Beryllium content in the deposit is very low. The film is ~85 μm thick, with a similar composition to that previously found for the flaking deposits at the inner louvres.

The primary source of beryllium in JET is the periodic evaporations which deposit Be in essentially the main chamber. The major source of carbon is also considered to be the main vessel walls. According to the results obtained, Be and C are transported towards the upper tiles of the inner divertor where the beryllium accumulates. Carbon, after deposition, is re-eroded through chemical sputtering and transported towards the inner base tiles. This chemical sputtering is apparently reduced by a change in plasma operating conditions. Whether the sputtering is influenced by the temperature of the vessel walls, or by some other plasma parameter, currently remains unclear.

The amount of deposition is being estimated from the beryllium build-up found on tiles 1 and 3, which are the only sinks for beryllium. This result (~22g), combined with the spectroscopically observed proportion of Be in the impurity influx into the divertor (~7% of the carbon influx) leads to 400g of deposited carbon. This result agrees well with spectroscopic estimations of a wall carbon source of 390-480g and a beryllium wall source of 20g.

1.2. ^{13}C INJECTION : A DEDICATED TRANSPORT EXPERIMENT

In order to confirm experimental transport observations previously obtained in JET, a special injection of $^{13}CH^4$ in the plasma boundary was scheduled in the last day of discharges prior to the 2001 shutdown. $^{13}CH^4$ was injected from the top of the machine (2.8g of ^{13}C) in 12 identical ohmic pulses.

The deposits on Tile 1 were thin enough to be able to use the ion beam technique Rutherford Back-Scattering (RBS) to analyse through the layers. Figure 2 shows a RBS spectrum measured from the bottom of tile 1. This spectrum was simulated using three different layers (confirmed by SIMS): the two layers referred to in the previous section, and a very thin surface layer containing the ^{13}C. The composition of the deposit is given in Table 1. The surface analysis showed there is at least 100 times more ^{13}C in the inner divertor than in the outer, confirming the transport of impurities from outboard to inboard.

1.3. FLAKES CHARACTERISATION

In JET a very large fraction of tritium is retained in co-deposits in the vicinity of the water cooled louvers adjacent to the inner divertor, not in line of sight of the plasma. These co-deposits sometimes become so thick that they spall off to form flakes, many of which fall into sub-divertor zones. Flakes were collected via a cyclone vacuum cleaner from underneath octant 5 and sent to FZK to be analysed [4].

The flakes have an average diameter of 400mm but occasionally flakes having length of 20mm were also collected. They are saturated with hydrogen isotopes. Optical spectroscopy reveals a layer structure coming from a sequential deposition process. Density measurements give an average flake density of (1.69±0.02) g/cm^3.

The average tritium activity is 1.06 TBq/g. measured via calorimetry. This value was confirmed by heating specimen with air at 800°C and collecting the liberated tritiated water in bubblers containing water. During this experiment, a rapid release of T is observed first, followed by a much slower but steady liberation over a period of many hours. This suggests that a fraction of tritium is strongly bonded and only liberated after full oxidation of the flake.

At the end of the Remote Tile Exchange 3g was not accounted for, which must be in the remaining flakes. With the same content of tritium as presented above (3.3mg of tritium per gram of flakes), 950g of flakes must be present in the vessel. This is confirmed via endoscope inspection of the sub-divertor volume where large quantities of flakes are observed.

Determination of the Specific Surface Area (SSA) of such material is of crucial importance for safety issues since the SSA has a tremendous importance in terms of reactivity. When such a reactive compound containing high amounts of tritium is exposed to air or moist air in a loss of coolant accident, oxidation could occur with a rapid release of T compounds. Brunauer-Emmett-Teller (BET) measurements were performed at TLK and a SSA value of (4.7±0.3) m^2/g was obtained.

1.4. CONCLUSIONS AND FUTURE WORK

Complementary experiments must be undertaken in the 2004-2005 work program in order to understand the remaining open question such as the decrease of the chemical sputtering observed on the inner divertor tiles.

The ^{13}C gas injection has been done again in order to confirm via surface analysis the SOL transport.

As usually noted, the use of CFC as a main chamber plasma facing components leads to several constraints in term of safety due to tritium trapping in remote co-deposited layers. In order to avoid as much as possible this kind of difficulties, ITER is supposed to be operated with a beryllium wall. In the forthcoming JET FT work program, new tasks have been launched in order to assess how an ITER-relevant beryllium wall experiment might be carried out in JET, and what are the consequences foreseen in terms of lost operational space.

The possible options considered are the coating with Be of the existing CFC tiles and the use of Be bulk material tiles.

Operating JET with Be walls has to be considered as a first priority in the following years in order to test how such a material sustains high heat load such as ELMs, and how the tritium accountancy is modified with such a new configuration.

2. DETRITIATION STUDIES

2.1. CURRENT EXPERIMENTAL STATUS

Estimations of ITER in-vessel tritium retention have shown **that regular** in-situ detritiation will be needed during operation. Detritiation processes based on laser or flash lamp are in-situ suitable methods. They are being investigated in the JET configuration in the frame of TF-FT.

After very promising results obtained in laboratory showing a possible cleaning rate of more than $3m^2$ per hour for a 50μm thick deposit, a flash lamp has been mounted on the JET Remote Handling boom and has been used for in vessel tests [5]. Co-deposited layers on inner divertor tiles have been treated within the vessel aiming to reach ablation. The flash lamp is now used in the JET Beryllium Handling Facility for detritiation by heating of poloidal limiter tiles. The efficiency of this technique will be assessed in 2004/2005, via comparison between treated and not treated samples using tritium calorimetric measurements and full combustion to be performed at the Tritium Laboratory at the Research Centre of Karlsruhe (FZK) and ion beam analysis of deuterium on tiles not exposed to deuterium-tritium plasmas.

Fast heating of an exposed surface by a focused laser beam allows temperatures higher than 1000K to be obtained on a thin surface layer [6,7]. Detritiation either by hydrogen desorption from the surface or by ablation of this surface layer results. Low and high repetition rate nanosecond laser benches have been used in the laboratory to study ablation of co-deposited layers on graphite. It was measured that ablation threshold fluencies were different by more than a factor 2 for graphite ($1J\ cm^{-2}$) and co-deposited layers ($0.4J\ cm^{-2}$). Ablation efficiencies of $0.025\mu m/Jcm^{-2}$ for graphite and $0.2\ \mu m/J\ cm^{-2}$ for co-deposited layers were also determined. Therefore, an ablation rate of 1 m^2 per hour is obtained for a 20μm co-deposited layer using a high repetition rate Nd-YAG laser beam of 250 W mean power. In this case, the laser fluency should be $1\ J/cm^2$ to provide the maximum co-deposited layer ablation efficiency of $0.2\mu m/Jcm^{-2}$ and to ensure laser surface cleaning without graphite surface damages.

2.2. CONCLUSIONS AND PERSPECTIVES

Both Laser and Flash Lamp are promising for in situ detritiation.

Based on the experimental results from laboratory tests on detritiation of co-deposited layers, an optimized laser ablation system will be designed and built in 2005-2006. This system will use a high power high pulse repetition rate Nd-YAG laser (200W of mean power) equipped with an optical fiber for beam transportation in the tokamak chamber. Special optical systems for beam focusing and scanning will be developed. This new system will be ready for in-situ JET detritiation at the end of 2005 and will be tested with tritiated tiles in the beryllium handling facility.

3. JET AS A TECHNOLOGICAL TESTING FACILITY

In the frame of TF-FT, JET facilities have been used to investigate the main features of several ITER relevant systems, and to experimentally obtain or collect data relevant for ITER design. In the frame of the design project of a water detritiation facility for JET, key components have been studied, such as the performances of electrolysers using solid or liquid electrolyte, and catalyst/packing mixtures for the Liquid Phase Catalytic Exchange (LPCE) column [8].

Relevant cross-section data have been verified in real D-T neutron spectra, code systems for shutdown dose rate calculation were validated against experimental data, and drawings of a JET octant were used for the development of an automated interface between Computer Aided Design (CAD) and Monte Carlo transport (MCNP) codes.

The prototype cryosorption panels (PCPs) represents one recent example of the use of the means available at JET for ITER component testing.

3.1. CRYOPANEL STUDIES

The design of the ITER high vacuum system is based on a number of supercritical helium cooled cryosorption pumps providing a high pumping speed and capacity, and fast on-line regeneration. In order to pump helium, which cannot be condensed at the available 5K cooling temperatures, and to help to pump hydrogen, the pumping cryopanels were developed at FZK with coating of activated charcoal granules [9]. After several tests at FZK, a large scale test arrangement was build at JET in the Active Gas Handling System to assess in detail the charcoal-tritium interaction and to derive performance parameters essential to the design of the ITER cryosorption pumps. The new PCPs were first operated under the JET Trace Tritium Experiment to pump gas from the torus and neutral beam injectors. It was observed that the PCP worked well according to the design specifications. Pumping efficiency is very high (estimated for D_2 as 23 mbar l/cm^2) leading to a high amount of pumped gas: 140bar l of D_2 in 245 l of PCP volume.

PCP are presently undergoing further tests with gas feeding from external supplies defined in agreement with the ITER gas exhaust tritium specifications.

After completion of the test, it is planned to investigate the panels at FZK and CEA/Cadarache to determine the tritium levels and to develop specific detritiation methods for these low-level tritiated materials.

CONCLUSIONS

The use of the JET facility is of crucial importance in order to test ITER fusion technology solutions. Within the JET FT-TF activities significant results have been obtained on fuel cycle issues as well as on in-situ detritiation.

The future FT-TF work programmes will focus on key issues needed to validate the current ITER design.

The use at JET of new materials such as beryllium and/or tungsten will be beneficial in defining the future pulse configurations for ITER. In-situ detritiation has to be pursued in order to show that co-deposited layers can be efficiently detritiated. New techniques have to be assessed like oxygen-induced tritium removal or new conditioning techniques using ICRH heating [10] or radiative plasma termination [11].

Important issues concerning neutronics (test of new material or code evaluation), diagnostics tests in fusion neutron environments, and particle source appraisal to control the fuel cycle can be addressed using the JET facility.

All these issues have to be studied in presence of tritium, therefore new tritium campaigns at JET would be of great benefit.

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Layer	D, at.%	Be, at.%	^{12}C, at.%	^{13}C, at.%	O, at.%	Cr, at.%
1	11	5	27	30	27	0
2	30	5	53	0	10	2
3	5	31	33	0	28	3

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Table 1: Composition of the deposit at the bottom of tile 1 exposed in 1999-2001 measured with RBS.

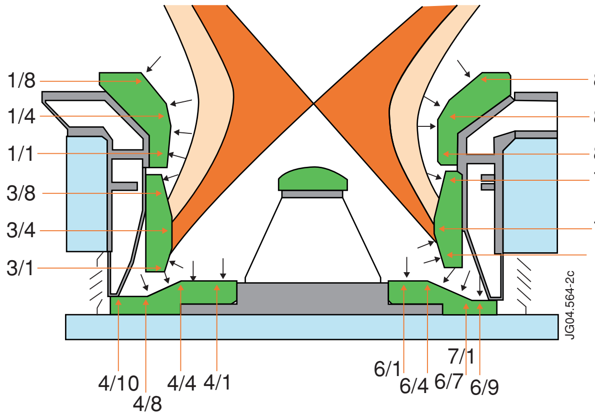

Figure 1: The JET MkIIGB divertor tile set. The samples for SIMS and IBA measurements are indicated with numbers. The first number in the sample code refers to divertor tile and the second one to the position in the tile.

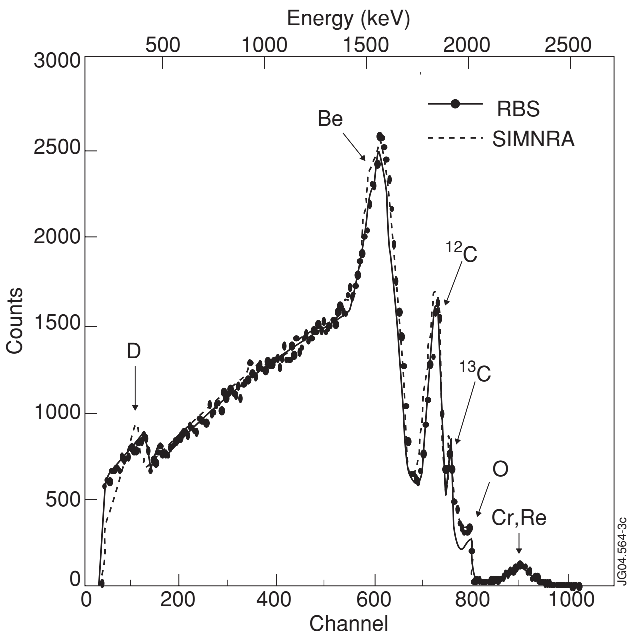

Figure 2: RBS spectrum and SIMNRA simulation from the bottom of tile 1 which was exposed in 1999-2001.