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R. Lässer, N. Bekris, A. C. Bell, C. Caldwell-Nichols, I. Cristescu,
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R. Lässer[1], N. Bekris[1], A. C. Bell[2], C. Caldwell-Nichols[1], I. Cristescu[1], S. Ciattaglia[3], P. Coad[2], Ch. Day[1], J. Likonen[4], D.K. Murdoch[5], S. Rosanvallon[6] and F. Scaffidi-Argentina[3]

[1] *Association Euratom/Forschungszentrum Karlsruhe, Tritium Labor, Postfach 3640, D-76021 Karlsruhe, Germany*
[2] *Association Euratom/UKAEA, Culham Science Centre, Abingdon OX14 3DB, UK*
[3] *EFDA Close Support Unit, Culham Science Centre, Abingdon OX14 3DB, UK*
[4] *Association Euratom/TEKES, VTT Processes, P.O. Box 1608, FIN-02044 VTT Finland*
[5] *EFDA Close Support Unit, MPI für Plasmaphysik, D-85748 Garching, Germany*
[6] *CEA, Centre de Cadarache, F-13108 Saint Paul Lez Durance, France*

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ABSTRACT.
The JET fusion machine is the only device capable of operation with tritium and of handling Be and therefore best suited to study tritium and fusion related issues. JET is now managed under the EFDA Agreement and experiments, enhancements and technology issues are organized as Scientific and Technical (S/T) tasks within various Task Forces (TF). A large variety of different activities is performed within the Fusion Technology (FT) Task Force. A brief summary of FT-studies related to tritium is given presenting topics such as tritium inventory in flakes and tiles, tritium depth profiles in PFCs, development of a facility for full recovery of the tritium in flakes and tiles, erosion and deposition of plasma exposed materials determined by surface sensitive techniques, development of detritiation techniques for different waste categories, characterization of packing/catalyst combinations for liquid phase catalytic exchange column and design studies of a water detritiation plant using either a solid polymer or KOH electrolyser, test of ITER cryo-panels in the JET Active Gas Handling System, etc.

1. INTRODUCTION

The JET machine is unique because it is the largest fusion device in operation, has Be handling capabilities, can achieve plasma configurations nearest to ITER scenarios, allows the study of the largest variety of heating possibilities and is best suited to test ITER diagnostics. Its scientific success has been recently summarized [1]. In addition, the JET tokamak was built to handle tritium, a limited and an extended tritium campaign were performed in 1991 [2] and 1997 [3], respectively, and remote handling tools have been developed and are used for the replacement of components inside the machine in a Be and tritium contaminated environment and in a high radiation.

The exploitation of the JET facilities is managed under the European Fusion Development Agreement (EFDA) by the Associated Leader for JET and the Close Support Unit (CSU) in Culham. Scientific and Technical (S/T) tasks for experiments, enhancements and technology issues are performed in various Task Forces (TF). In the Fusion Technology Task Force (FT-TF) mainly S/T tasks with respect to tritium in the tokamak; tritium process and waste management; plasma facing components; engineering; safety; and test beds; are performed involving the JET Operator (UKAEA) and several EU Laboratories. 27 S/T tasks were/are performed within FT-TF in the years 2000 to 2002, 19 new ones are being started for the work programme 2003.

This paper reports about tritium related tasks performed within the FT-TF whereas the other FT-tasks are summarized in another paper [4].

2. TRITIUM IN FLAKES AND TILES

During the 1997 Deuterium-Tritium Experiment (DTE1) tritium accountancy performed within the Active Gas Handling System (AGHS) [5] revealed clearly that a large fraction (up to 40% [6]) of the tritium injected into the tokamak was at least temporarily trapped in the machine. Carbon layers co-deposited with up to 80% hydrogen [7] had been observed before DTE1. These layers are

believed to be the main sinks of the injected tritium. Various FT tasks were started to enlighten these problems: e.g. determination of the tritium inventories in tiles and flakes and of their chemical composition, collection of flakes remotely with a special vacuum cyclone, characterization of the areas of erosion or deposition on the plasma facing components with various surface analytical techniques. In the following the tritium inventory in tiles and flakes as well as the determination of erosion and deposition areas in the divertor are briefly described.

2.1 TRITIUM LEVELS ON THE SURFACE AND IN THE BULK OF THE MKIIA DIVERTOR TILES

Figure 1 shows the surface tritium concentrations [8] in MBq/cm^2 for the MkIIA divertor. The results are obtained for 1mm thick discs cut from the surface, but the activity is mainly located in the co-deposited layers at the surface. Figure 2 presents tritium depth profiles of three different locations for tile 3. In the bulk the tritium concentration is far lower than in the co-deposited layer, decreases at least up to the middle of the tile and increases again towards the other surface. This behaviour can qualitatively be explained by the porosity of the CFC tiles, the diffusion of tritium and possibly by the uptake of tritium due to pressures of up to 30 Pa in the machine during regenerations of the torus cryo-pumps. In the case of graphite tiles the depth profiles show in general a far larger drop of activity from the surface to the bulk, very flat profiles within the whole tile and no increase near the not plasma exposed surfaces. The tritium inventories for the divertor, inner wall and outer poloidal wall tiles were determined to be only 60, 20.4 and 50.5 mg [9], respectively, clearly showing that the tiles can not be considered as the main sink of the unaccounted 3.4 g tritium at the end of the Remote Tile Exchange (RTE) shutdown.

2.2 FLAKES

During the RTE shutdown 150 g of flakes were collected with a remotely handled vacuum cleaner, mainly from the water cooled louvres. The flakes were found to have high specific activities of 1.2 TBqg^{-1} and showed high tritium off-gassing rates (2.6 GBqh^{-1}g^{-1}) [10], especially when handled in an oxygen containing atmosphere. Calorimetric measurements were disturbed by an unusually slowly disappearing heat production which was attributed to chemical reactions with oxygen. This again showed that these new materials are not stable and are a dangerous source for uncontrolled tritium releases. 0.93 g of the flakes were sent to the Tritium Laboratory Karlsruhe (TLK) and in three years storage at room temperature 0.50 TBq tritium was released. The remaining tritium inventory was determined to be 0.60 TBq by calorimetry. This gives an initial atomic tritium concentration in the flakes of 1.5% assuming a D/C ratio of 75% [7].

During the 2001 JET shutdown approximately 40 g of flakes were collected from the subdivertor region of one JET octant.

A dedicated facility and glove box is being built [11] in the Active Gas Handling System (AGHS) for the determination of the specific activity of flakes and dust by calorimetry and for recovery of

the tritium by thermal treatment or full combustion using only one sample container for both experiments to reduce handling of the flakes.

2.3 EROSION AND DEPOSITION IN A POLOIDAL SET OF DIVERTOR TILES

To study erosion/deposition CFC tiles with special coatings (C+10% B layer (thickness ~2.5μm) on a thin Re interlayer; or W layers (thickness ~3.4μm)) or with mechanical markers were installed in the divertor and at selected positions of the Inner Wall Guard Limiter and Outer Poloidal Limiter. The mechanical markers allow determination of large scale erosion/deposition via a micrometer, whereas the special coatings serve as an indication of the substrate when deposits are analysed (e.g. by secondary ion mass spectometry (SIMS)) or as a clear indication for erosion when their thickness is reduced. Figure 3 presents the erosion/deposition measured by micrometer and SIMS for MkIIGB divertor tiles. In the top section of Figure 3 the locations for the micrometer measurements are given by arrows. Both measurements agree in their trends very well. Clear deposition is observed in the inner leg of the divertor, whereas small erosion/deposition is detected in the outer with the exception of the sloped area of the tile 6 where large deposition occurs. Repeated micrometer measurements lead to reductions of the thickness (indicated by the grey shading in Figure 3) due to the soft structure of the co-deposited layers. Very thick deposits of more than 200μm were found on the floor tiles (position 10 and 16). These deposits are located at the extreme corner of the plasma-accessible region. Further interesting results were also obtained by SIMS, time of flight elastic recoil detection (TOF-ERDA) and Rutherford backscattering spectroscopy (RBS) measurements.

3. DETRITIATION OF TRITIUM CONTAMINATED MATERIALS

Due to the injection of 36 g tritium into the tokamak during DTE1 the interior of the tokamak, its connected systems (NIB4/NIB8) and of the primary containments in the AGHS are now highly tritium contaminated. Various waste categories can be distinguished depending on the materials, the degree of tritium contamination (e.g. low level waste (LLW <12 kBq/g), intermediate level waste (ILW >12 kBq/g) and the activation levels [12]. Special detritiation techniques have to be developed in order to be able to transfer waste from ILW to LLW and to guarantee an economic and safe final disposal of the materials. Within FT detritiation studies and experimental work are performed for the detritiation of plasma facing graphite and CFC tiles, of tritiated water (see section 4), of the tritium storage materials ZrCo and U, etc. Studies [13] showed that full combustion seems to be the best detritiation procedure for flakes and dust as the volumes concerned are small. Short high heating pulses should be used for decontamination of co-deposited layers on carbon tiles, but also lower temperatures are useful for the tritium removal from surfaces and bulk. For stainless steel the most promising detritiation technique is melting with gas bubbles passing through the liquid. Hydrogen for isotopic swamping and argon are proposed as purge gas.

3.1 DETRITIATION AND DEACTIVATION OF U AND ZRCO TRITIUM STORAGE BEDS

The objective is to process tritium storage beds for safe final disposal. In addition to the tritium hazard that such beds present, uranium beds also present a pyrophoric hazard if the uranium is exposed to oxygen. Therefore the objectives are to remove as much tritium as possible, to make the beds chemically inert and to ensure that any remaining tritium cannot be released. Calorimetric measurements with Amersham-type getter beds filled with uranium and zirconium cobalt have shown that even after several stages of isotopic swamping with deuterium significant amounts of tritium remain in the bed. The total tritium for both types of bed reaches a level of a tenth of a TBq, however the relative amounts in the getter material and primary containment are not known.

Oxidation experiments have been carried out on a JET-type getter bed filled with 84.5 g of zirconium-cobalt. The ZrCo reacted very slowly at room temperatures and the reaction only became significant at 250°C, this can be regarded as a safety bonus for ZrCo. The final experiments were performed at 480°C. Considerable helium-3 was released in these experiments, indicating that it had been trapped in the ZrCo. Tritium was also released, though clearly as HTO, and work is underway to collect and measure this. It is clear that oxidation of the ZrCo was extensive from the amount of He-3 released and that it also further detritiated the ZrCo. Further work will be performed on uranium beds, initially with beds that have not seen tritium to establish the oxidation characteristics.

4. DEVELOPMENT OF A WATER DETRITIATION SYSTEM FOR JET

The JET AGHS does not have a water detritiation plant [5]. Until now, tritiated water collected in AGHS has been sent to Canada for reprocessing. As this route might not be available in the future, design studies of a water detritiation plant for JET have been started.

The most suitable method for water detritiation is the CECE process, a combination of electrolysis and catalytic exchange of hydrogen isotopes in a Liquid Phase Catalytic Exchange (LPCE) column. The intention is to recover tritium in gas phase from tritiated water and to inject enriched tritium into a Cryogenic Distillation for further enrichment.

Laboratory scale experiments and design studies are performed within FT-TF to test the performance of various packing materials/catalysts combinations, to determine the mass transfer parameters between the various aqueous and gaseous species, to check the parameter and the long term behaviour of electrolysers (solid polymer or KOH type), etc.

A model describing in detail the deuterium/tritium transfer between water and hydrogen in the packing and catalyst sections of a catalytic exchange column was developed and is currently being validated by experimental data. In order to completely characterise the separation performances all the isotopic species that can appear in a catalytic exchange column have been considered [14].

The separation performances of four catalyst-packing combinations from different manufacturers (SCK-CEN Belgium; Mendeleev University Russia; Institute of Cryogenics and Isotopic Technologies (ICIT) Romania-TLK; Showa Eng. Japan) have been investigated with respect to

deuterium transfer between gas phase and liquid phase in a LPCE column. The different catalyst/packing materials with the exception of the Japanese one behaved very similarly and well [15].

For simultaneous tritium and deuterium transfer, the performances of catalyst-packing mixtures developed by TLK are currently investigated at TLK. For the LPCE process investigated at low tritium activity (~40 $Bqcm^{-3}$) the experimental data related to tritium transfer from the vapour phase to gaseous hydrogen in a protium rich environment showed a mass transfer coefficient reduced by 30% in comparison to deuterium. In the presence of small deuterium concentration of up to a few percent the mass transfer coefficient for tritium is again lower compared to the value when only tritium is transferred. The mass transfer coefficient for HTO species depends on the concentration and the mass flux of HDO species.

In order to investigate the simultaneous deuterium and tritium transfer under true JET and ITER conditions a new CECE facility with a LPCE column is being installed at TLK. Two solid polymer electrolysers with a nominal flow-rates of 1 m^3/h each are used and a decontamination factor between 10^3–10^4 is expected.

Preliminary [16] and detailed design studies for a water detritiation plant for JET are also performed in view of the construction of such plant if required.

5. USE OF AGHS AS A TEST BED FOR ITER

Special ITER components such as a PERMCAT reactor [17] which in the present design is the main component of the third clean-up loop of the ITER Tritium Exhaust Processing System and cryo-panels of the same design as those to be used in the ITER cryo-pumps will be tested in the AGHS at JET. The PERMCAT has been manufactured, sent to JET and shall be installed and tested at the AGHS during 2002.

5.1 TEST OF CRYO-PANELS IN AGHS

The purpose of the test is to reveal any showstoppers to the ITER cryopumping concept arising from tritium interaction. The ITER panel design is validated in all aspects of operation except the performance under tritium [18]. In the field of tritium cryosorption on panel structures, only one small scale phenomenological experiment limited to liquid nitrogen temperatures has been performed until now [19].

Therefore, a prototype cryosorption pump (PCP) will be installed in Cryogenic Forevacuum (CF) of the JET AGHS. The cryosorption panel test arrangement is of ITER relevant design in terms of geometry and dimension, coating and sorbent material. The panel surface area corresponds to one ITER panel, 28 of which are installed in each ITER pump. The active surface is by a factor of 300 larger than that for the former small scale experiment. The PCP unit is currently under manufacturing at the Research Centre Karlsruhe (FZK) and industry and will be ready for installation at JET before the end of 2002.

The main objectives of the experimental programme are to clarify the pumping performance and mechanism for tritium and tritiated gases when compared to the other hydrogen isotopes, to study

the regeneration behaviour of cryosorption panels loaded with tritiated gases, and to estimate the different tritium inventories. In a first test stage, the PCP will be separated from the torus matrix line, while in the second stage, the PCP will be connected with the torus and will be employed as regular cryopump during the planned trace tritium campaign in 2003. This test aims to assess the dynamic performance of the cryopump under pulsed tokamak operation conditions. A more detailed description of the PCP programme can be found in [20].

CONCLUSIONS

Fusion Technology activities at JET, launched in April 2000, provide relevant contributions to JET operation and ITER design in the following areas ('FT Topics'):

- Tritium distribution in the Tokamak: tritium distribution in tiles and flakes;
- Tritium processes and waste management: design of a water detritiation system, relevant component tests and improvements, test of ITER prototype components in the JET AGHS, development of detritiation techniques for different JET waste categories;
- Plasma facing components: characterisation of tiles, model validation of PFC behaviour under plasma loads;
- Engineering: improvement of In-Vessel inspection techniques based on Laser technology, qualification of optical fibres, evaluation of Remote Handling operating experience at JET;
- Safety: In-Vessel neutronics activation model validation, operating experience data collection and relevant component failure rate evaluation.

Until now 27 S/T tasks were/are performed within FT-TF, while 19 new ones are being started in the frame of the work programme 2003. A few tasks have been delayed mainly because of the re-schedule of the JET operation plan (e.g. re-schedule of shutdown activities and/or experimental campaigns, use of the test bed facilities for JET operation, modifications of AGHS for the Trace Tritium Experiment in 2003, etc.).

The tasks on tritium (e.g. determination of the tritium distribution in tiles and flakes inside the JET machine, development of detritiation methods and waste disposal routes for various solids (graphite, stainless steel, uranium, etc.) and liquids, the development of a water detritiation plant, etc.) play the major role (about 70%) in the activity of the FT-TF. They are especially important both for the operation and decommissioning of JET and for the design of the tritium system and tritium licensing aspects of ITER.

Furthermore, they provide the essential background for the waste disposal of tritiated ITER components and can validate the ITER design choices through tests performed at JET.

The main aims of the further development of the FT activity at JET in 2003 and beyond are:

- Demonstration of ITER/JET relevant tritium technologies;

- Studies for a detailed design of a water detritiation plant for JET and ITER;
- Demonstration of efficient and economic disposal routes of further tritium contaminated waste categories for JET and ITER;
- Characterisation of PFC/Be FW behaviour;
- Collection and interpretation of further JET operating experience data for ITER engineering design and safety analysis.

ACKNOWLEDGEMENTS

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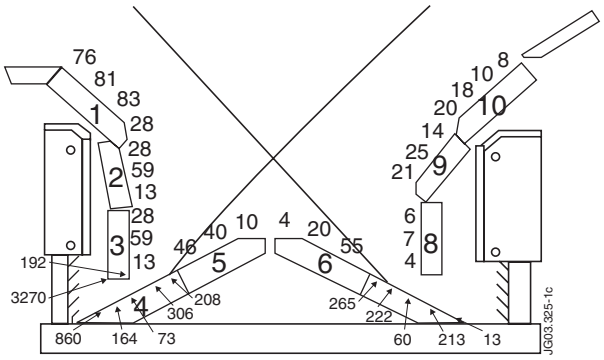

Figure 1: Cross section of JET MkIIA divertor with tritium surface concentrations given in MBqcm^{-2}

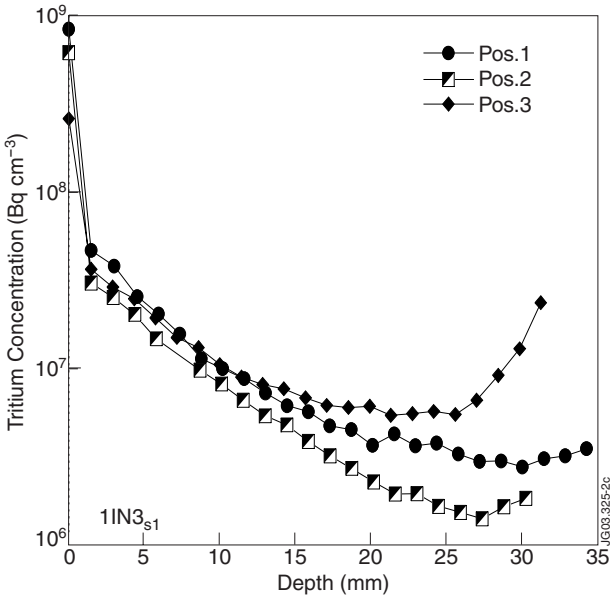

Figure 2: Tritium depth profiles of MkIIA divertor tile No. 3 (see Figure 1) at three different positions.

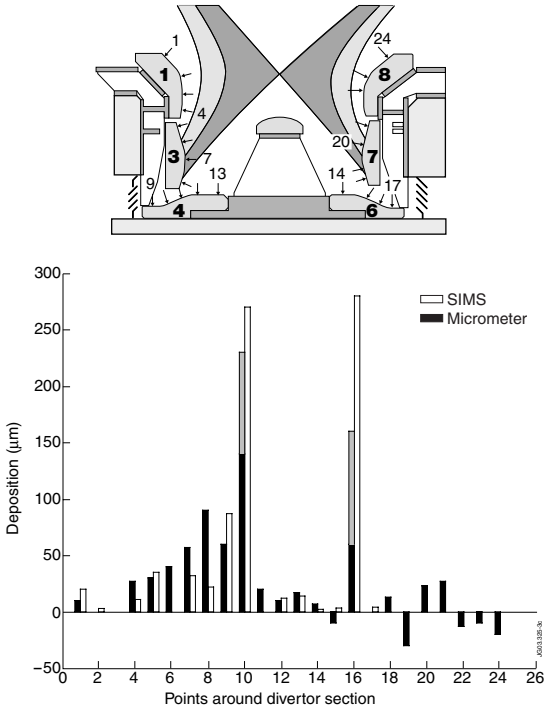

Figure 3: Bottom: Erosion/deposition measured with micrometer and SIMS. Grey shading in micrometer results shows decreased thickness after successive measurements. Top: Cross section of JET MkIIGB divertor showing the position of mechanical slots used in micrometer measurements.