TRITIUM IN FUSION: R&D IN THE EU

R. Lässer[a], A. Antipenkov[b], N. Bekris[c], L.V. Boccaccini[b], C.J. Caldwell-Nichols[b], I. Cristescu[b], Ch. Day[b], M. Gasparotto[a], M. Glugla[b], G. Dell'Orco[a], F. Gabriel[d], O. Gastaldi[e], Ch. Grisolia[e], S. Knipe[f], A. J. Magielsen[g], A. Möslang[b], D. Murdoch[a], R. Pearce[a], A. Perevezentsev[f], G. Piazza[a], Y. Poitevin[a], I. Ricapito[g], J.-F. Salavy[d], L.A. Sedano[i], M. Zmitko[a]

[a]*EFDA Close Support Unit, Boltzmannstrasse 2, D-85748 Garching, Germany*
[b]*Forschungszentrum Karlsruhe, P.O. Box 3640, Karlsruhe, D-76021, Germany*
[c]*EFDA Close Support Unit, Culham Science Centre, Abingdon, Oxon OX14 43DB, UK*
[d]*Commissariat à l'Energie Atomique, CEA/Saclay, F-91191 Gif-surYvette, France*
[e]*Commissariat à l'Energie Atomique, CEA/Cadarache, F-13108 Saint Paul Lez Durance, France*
[f]*Culham Science Centre, UKAEA, Abingdon, Oxon, OX14 3DB, UK*
[g]*NRG, P.O. Box 25, 1755 ZG Petten, The Netherlands*
[h]*ENEA Brasimone, I-40032 Camugnano, Bologna, Italy*
[i]*CIEMAT, Complutense 22, 28040 Madrid, Spain*

Tritium as one of the two fuel components for fusion power plays a special role in any fusion device. Due to its volatile character, radioactivity and easy incorporation as HTO it needs to be controlled with special care and due to its scarcity on earth it has to be produced in-situ in future fusion power plants. The paper discusses the present tritium R&D activities in fusion ongoing in the EU and presents the various processes/techniques envisaged for controlling tritium in future fusion reactors focusing mainly on the issues of breeding blankets and the fuel cycle in DEMO.

I. INTRODUCTION

The safe processing and control plus the effective generation of tritium are key issues in future fusion power plants (FPP) to minimise releases of tritium into the environment and as natural tritium is very rare. The power (and hence the electricity generation) in FPPs comes from the fusion of deuterium and tritium nuclei which yields 14.1 MeV neutrons and 3.5 MeV alpha particles. These energetic helium nuclei are also used to heat the plasma. A very small fraction of the tritium in a plasma is burnt up in this reaction, the remainder is evacuated and then re-injected into the fusion plasma after removal of impurities and isotopic adjustment of the Q_2 composition. This paper presents the R&D activities ongoing in the EU for structural and functional materials for confinement and production of tritium, tritium extraction and recovery systems of the breeding blankets (BB), the ITER and DEMO fuel cycles (FC), tritium accountancy, plus tritium in plasma facing components, IFMIF and wastes. In the EU five types of BBs have been studied within the power plant conceptual studies (PPCS)[1, 2]: the water cooled lithium lead (WCLL) BB, the helium cooled lithium lead (HCLL) BB, the helium cooled pebble bed (HCPB) BB, the dual coolant (DC) BB and the self-cooled (SC) BB. Presently most of the R&D is performed for the HCLL and HCPB BB concepts as they have the lowest development risk. These two concepts have their own specifically designed BB and auxiliary systems, such as the tritium extraction/removal (TES/TRS) systems to collect the tritium bred in the BBs, He coolant system (HCS) to remove the nuclear heat and the coolant purification system (CPS) for detritiation of the coolant He.

II. TRITIUM IN BREEDER BLANKET MATERIALS

The DT-neutrons transport most of their energy through the first wall (FW) of the BBs which almost fully surround the plasma. The purpose of the BBs is three fold: i) to produce by fission of Li atoms the tritium required as new fuel for the fusion plasma, ii) to transfer the thermal energy of the DT-neutrons and associated fission reactions to the He coolant for electricity generation and iii) to protect the radiation sensitive superconducting magnets.

The European BBs consist of two main categories of materials, namely the structural materials (EUROFER, SiC_f/SiC), the functional materials such as Li ceramic breeders (Li_4SiO_4, Li_2TiO_3) or liquid metal (Pb-15.7at%Li), the neutron multipliers (Be, Pb), anti-permeation/anti-corrosion (AP/AC) layers, coolants (He, Pb-Li), etc.

II.A. Structural Materials of Breeding Blankets

The structural materials foreseen in the EU for the HCPB and HCLL concepts are the reduced activation ferritic/martensitic (RAFM) steel EUROFER, a 9% Cr alloy with W, V, Ta content in the wt% range of 1.0-1.2, 0.15-0.25 and 0.10-0.14, respectively. Reduced activation

of these steels is achieved by removing Ni and replacing Mo and Nb with W and Ta, respectively. For tritium applications important properties are solubility, diffusivity and permeability[3] of hydrogen in EUROFER. Traps influence the hydrogen behaviour in RAFM steels[4, 5]. In a fusion environment defect and defect clusters will be created which modify the microstructure and type and number of traps. The tritium permeability in EUROFER as a function of traps and high radiation dose is still to be determined. In steel hydrogen permeation is generally high requiring AP layers on hot metal surfaces in contact with the hydrogen.

II.B. Functional Materials of Breeding Blankets

II.B.1. Li-ceramics

Li ceramics are used as pebbles. The pebble beds are contained and cooled by cooling plates (CP) with internal channels through which He flows with temperatures between 400 and 500°C. The maximum He temperature is approximately 500°C to respect the upper design limit of 550°C for EUROFER. The Li ceramic pebbles should show low swelling with damage, phase stability up to temperatures of 920°C in all contacting gases, good integrity up to high burn-ups to avoid the creation of dust, short tritium residence time, etc. The height of the pebble beds is chosen to keep the maximum temperature below around 920°C. Relatively thin Li ceramic beds are required (1 cm thick under DEMO conditions) to avoid too high temperatures because of the low thermal conductivity of Li-ceramic pebble beds, even in the presence of the purge gas He, and the nuclear heating in the Li pebbles. On the other hand, high temperatures are desirable to increase the tritium release. The pebbles are packed with a porosity of 36%. The gaps between the pebbles must stay open so that the purge gas can transport the released tritium to the TES. Strong reduction of this porosity due to pebble fragmentation or creep can endanger this purge function and must be avoided. The in-pile tritium releases of various types of Li ceramics, e.g. Li_2TiO_3 (Ref. 6), were measured in the High Flux Reactor (HFR), Petten, as a function of reactor power, temperature and purge gas flow and composition. The influence of strong magnetic fields (to simulate the BB environment) on the tritium release in Li-ceramics was also studied[7].

II.B.2. Liquid Pb-15.7at%Li

Differences in the experimental results for hydrogen solubility and diffusion in Pb-15.7at%Li were observed between measurements performed in the past[8] and more recently[9]. The differences are so significant for the tritium control in the HCLL BB concept that a further study was recently launched. Due to the low hydrogen solubility and the presence of dissolved hydrogen in Pb-Li, EUROFER easily absorbs tritium from the PbLi. The PbLi is circulated slowly through the BB to keep corrosion and magneto-hydro-dynamic (MHD) effects small. Laminar flow prevails in most inner parts of the BB. Modelling of the permeation through the mid-plane HCLL BB CPs showed that due to the flow a concentration boundary layer is created in contact with the EUROFER structures which reduces the permeation into the He coolant and acts as a permeation reduction layer with a permeation reduction factor (PRF) of 30 (Ref. 10). To reduce tritium permeation even more the use of further AP layers is considered, preferably on the He coolant side (oxide layers created by adding H_2/H_2O to the He coolant show good self healing potentiality), but also on the PbLi side (mainly Al based coatings). The AP layers in contact with the PbLi in the HCLL BBs should also act as an AC coating. Indeed, PbLi attacks EUROFER near the upper temperature window. Some high corrosion rates have been observed (90 μm/y (Ref. 11) at 480°C and 30 cm/s Pb-Li flow) near the upper temperature window. Further critical issues are the precipitation of the corrosion products and possible influences on the PbLi flow.

II.B.3. Neutron Multipliers

Be pebbles of 1 mm diameter are used in pebble beds of the HCPB BB concept as neutron multipliers to increase the efficiency for tritium generation and to achieve an overall tritium breeding ratio (TBR) greater than one. The volume of these beds is about 3 to 4 times larger than for the Li ceramics. Neutrons can create large amounts of He and tritium in the Be pebbles causing significant swelling and the evolution of a dense dislocation network. In particular the tritium inventory increases with neutron dose and raises concern of unacceptable tritium releases under certain accident scenarios. Special emphasis is given to the production of pebbles with very small grains (e.g. via sintering). The hope is that with increasing dose a dislocation network with channels and openings up to the Be surfaces will evolve offering the tritium short diffusion paths for release into the purge gas. The maximum design temperature of Be in the HCPB concept is about 650°C. In addition the new material class of beryllides ($Be_{12}Ti$, $Be_{12}V$) is gaining increased interest due to their superior swelling characteristics, reactivity with air and steam and tritium release. In the EU the production of larger pieces of $Be_{12}Ti$ (Ref.12) with high phase purity is now possible despite its inherent brittle character. This material will be used in future trials to produce beryllide pebbles by the rotating electrode method (REM), one of the future DEMO materials R&D activities between Japan and the EU within the Broader Approach (BA) agreement.

II.B.4. Antipermeation/anticorrosion Layers

The materials studied in the EU to reduce permeation and corrosion in EUROFER are Al (Al_2O_3)-, Er (Er_2O_3)-, H_2/H_20 based oxides and W-based coatings, respectively. The properties of these layers should include good adhesion to the substrate, stability under cyclic conditions, self-healing capability for defects and cracks, compatibility with

contacting media, low hydrogen permeation, low corrosion rates, irradiation resistance, no long term activation and no prevention of recycling coated materials. To date almost all studies with these coatings have been performed without tritium. AP/AC layers for EUROFER should be available for testing in the so-called test blanket modules (TBMs) at the end of the first ten years of ITER operation, will be likely required for the second phase of ITER and could be mandatory for future FPPs. Much effort is still needed to fabricate these AP/AC coatings by efficient production methods for the surfaces of BBs in contact with hydrogen or PbLi and to test their effectiveness and lifetime under fusion conditions.

III. ITER FUEL CYCLE

III.A. Inner ITER Fuel Cycle

The ITER FC and its main subsystems such as vacuum systems, tokamak exhaust processing (TEP), isotope separation system (ISS), storage and delivery system (SDS), water detritiation system (WDS), fuelling systems and analytical system have been described extensively[13]. Many improvements - recently proposed by the participant teams (PTs), the EU, India, Japan, South Korea and the United States, involved in the in-kind supply of FC components - have been discussed in the ITER design review[14]. Previously the EU performed the R&D for the whole FC. Since the agreement of the PTs on the procurement packages, the EU is now focusing on the in-kind components, such as cryopumps, WDS, ISS and leak detection for the ITER FC. For the ITER vacuum pumping systems, the EU is performing the following activities: 1) design and fabrication of a 1:1 torus cryopump; 2) update of the TIMO test facility to test the full size cryopump; 3) study of electromechanical loads on the torus cryopump; 4) development of a vacuum conductance code for all flow regimes (laminar, molecular and transitional) in the ITER divertor; 5) construction of a rig to perform benchmark tests for the pumping code; 6) initial design studies for the diagnostic neutral beam cryopump and cold valve boxes. The recent EU activities for the ITER tritium plant are: 1) commissioning and testing the WDS and ISS prototypical systems at Tritium Laboratory Karlsruhe (TLK); 2) updating the ITER ISS and WDS integrated design; 3) determination of the lowest tritium hold-ups in different packing for the ISS; 4) endurance tests of WDS components; 5) conceptual design on ITER air and vent detritiation systems (ADS/VDS); 6) studies of the tritium plant building layout; 7) finalizing activities with TEP, etc.

III.B. Tritium in the Auxiliary Systems of TBM Systems

For the first 10 years of ITER the tritium for DT operations will come from external sources (e.g. CANDU fission reactors). TBMs[15, 16] of the HCLL and HCPB BB concepts will be installed in ITER from Day 1 to gain experience for future use in DEMO. As only three horizontal ports will be available, the amount of tritium bred will be negligible compared with the needs of ITER. The tritium from TBMs needs to be removed via TES and TRS of the HCPB and HCLL concept, respectively. Various processes depending on the chemical nature of the tritium in these loops are available. In the HCPB concept the tritium bred in the Li-ceramics is released into the purge gas and taken to the TES, as HT and in small amounts as HTO (3.2% HTO is the design value[17] for a reference gas containing 0.1% hydrogen in 0.11 MPa He). HTO can be adsorbed in molecular sieve beds at room temperature or frozen in cold traps at lower temperatures. As high specific tritium concentrations in the water are expected, the reduction of the highly tritiated water to hydrogen should be considered as early in the process as possible. Hydrogen gas can be trapped in molecular sieve beds at 77 K, in metal getter beds or removed from the He purge by means of Pd-Ag permeators or ceramic protonic conductors[18]. In the EU trapping of water vapour in a cold trap[19] and of hydrogen in a 77 K cold molecular sieve bed has been published[20]. For the HCLL concept the liquid Pb-Li with the dissolved tritium is circulated through TRS, an external packed column, where He entering at the bottom flows in counter-flow to the liquid Pb-Li. By optimising the packing type and column dimensions a large contact surface between the He filled gas volumes and the PbLi liquid is achieved and this stimulates tritium transport from the liquid into the He. The tritium in the He is then collected in a TES, as above. A new liquid PbLi loop to determine the best parameters for TRS was recently installed[21], but only for use with H_2 and D_2. Due to the small amounts of tritium produced in these TBMs and the limited space in ITER, the TES and TRS need to be selected with care. The TESs shall allow accurate tritium accountancy for comparison with the values deduced from neutron measurements and neutronics model calculations. In the EU the TESs for the two TBM systems to be installed in ITER are presently being designed. 1.4 kg/s of He need to be circulated in the HCS of a HCPB TBM system to avoid excessive temperatures in the TBM. Despite the low tritium inventories in the TBMs tritium will permeate into the He coolant. The CPS has to remove continuously this tritium, together with impurities, and to add the required ratios of H_2 to H_2O to achieve optimum oxidising conditions on the inner EUROFER surfaces. The same tritium extraction processes can be used in the CPS as for TES. The main differences are the higher He pressure of 8 MPa and the lower HT and HTO concentrations. Two tritium compatible He circulators are presently constructed for testing the TBMs in the European He test facilities HeFUS3[22] and HELOKA[23] before installation in ITER. These will be oil free with magnetic bearings and dry emergency bearings, have a total He leak rate $<10^{-6}$ Pa m^3/s at 10 MPa internal pressure, will supply a He mass flow of 1.4 kg/s at a pressure ratio of 1.13 and be

tritium compatible apart from the electrical insulation (resin) of the stator windings.

IV. DEMO FUEL CYCLE

DEMO will be the first FPP with a self-sufficient tritium BB. Tritium from external sources will be only needed during a short start-up phase. For continuous production of 3.7 GW_{th}, approximately 500 g of tritium will be burnt per day. 550 g/d are to be produced (TBR=1.10) and 50 g/d stored in the long term storage for other uses (e.g. start of other FPPs). About 35 kg of tritium will be injected into DEMO per day. Due to the expected higher burn-up efficiency in DEMO, the DT feed rate to keep the plasma burning is approximately 300 Pa m^3/s, only slightly larger than that foreseen for ITER.

IV.A. Inner DEMO Fuel Cycle

DEMO will operate continuously under best conditions for electricity production. The inner DEMO FC will be similar to ITER but simpler. The hydrogen isotope mixture from the torus permeating through the permeators of the first stage of TEP will have similar isotopic composition to the feed gases injected into the plasma. They can be directly transferred to the fuel injection system and topped up with newly-bred DT (to compensate for the burnt DT), by-passing ISS and SDS of the inner loop and may require only a small isotopic balancing with D_2 and T_2 gases from SDS. The hydrogen gained in the second and third stages of TEP will be sent to the ISS of the inner loop for isotopic enrichment. The ISS can be smaller than for ITER, thus allowing the production of "pure" H_2, D_2 and T_2. These gases can be stored in metal hydride beds equipped with in-situ calorimetry. The change of isotopic composition during desorption is thus negligible in contrast to the ITER storage beds with their 50% D/50% T or 90% T/10% D gas mixtures. The second and third stages of TEP are expected to be also smaller than in ITER. Only the number of permeators in the first stage may need to be increased due to the slightly larger throughput in DEMO. If larger pressures in the DEMO divertor could be accepted, the replacement of cryopumps by large Roots pumps could become possible. This would simplify the inner FC and contribute to the advantages of lower tritium/hydrogen inventories, continuous operation, no temperature cycling of large components, avoiding the use of cryogens, etc.

IV.B. Tritium Collection Loops of the DEMO BB Systems

The processes for tritium collection in DEMO BBs will be similar to the ones discussed above. The main differences are that in the helium coolant system (HCS) of the DEMO BBs huge He flows (up to 3000 kg/s) with small concentrations of H_2, HT, H_2O and HTO have to be processed and larger amounts of highly tritiated hydrogen and water will be collected on a daily basis. The hydrogen will be sent to a special ISS designed for the needs of the BB collection loops. In the collected water specific activities of up to 5 g tritium per kg water are expected. Detritiation of this water could be achieved by means of i) PERMCATs[24], ii) palladium membrane reactors (PMR)[25], iii) ceramic protonic conductors and iv) metal powders capable of reducing HTO to HT and being reactivated. If the tritium concentration in the water needs only to be reduced by a few orders of magnitude, the highly tritiated water could be injected into a liquid phase catalytic exchange (LPCE) column at the top and the water with lower concentrations extracted from the bottom and sent into the BB WDS which is used for detritiation of low tritiated water. Each of these five processes for treating highly tritiated water requires further dedicated R&D. Decomposition of the water vapour instead of its collection as water if possible seems to be essential. Impurities in the He gas of CPS and TES will be collected from time to time and processed in a dedicated impurity processing system of the BB using the same processes as in the second and third stages of ITER TEP. Permeated hydrogen will be sent to the BB ISS. The main product DT of this ISS will be directed to the fuelling streams to compensate for the tritium burnt, whereas the 50 g surplus will be stored as pure tritium in the long term storage part of SDS. In this way a clear separation of the processing duties of the inner loop and the BB collection loops is achieved. DEMO can be kept in operation as long as tritium from the long term storage system is available even when difficulties in the supply of the DT gas from the BB may have occurred. ISS and WDS will be operated as a combined system as in ITER. In this way the hydrogen released from ISS can be further detritiated in the LPCE column of WDS and then be reused as isotopic exchange gas in TES/TRS or as feed gas to the CPS to establish the required ratio of H_2 over H_2O for oxidation of the inner EUROFER surfaces. Another issue is the fabrication of the large tritium compatible components needed for handling the huge HCS He flows in DEMO, such as He circulators, economisers, heaters, etc.

IV.C. Tritium Accountancy in DEMO

In general licensing authorities require at least an annual full accountancy of all tritium inventories on site. This requirement can be fulfilled by performing pressure, volume, temperature and tritium concentration (pVT-c) or calorimetric measurements on batches during a shut-down period. Due to the continuous operation in DEMO and the large tritium amounts bred in the BB collection loops, injected into the vacuum vessel and processed in the inner FC, in-situ dynamic accountancy[26] is needed for operational needs. Flows and compositions of the gases in the BB and inner FC will be continuously monitored and controlled at selected points between subsystems. Part of these

diagnostics needs to be developed and their accuracy, reproducibility, reliability, stability, etc. assessed.

V. TRITIUM IN PLASMA FACING COMPONENTS

Significant tritium trapping in co-deposited layers and plasma facing components (PFCs) is now a fact proven in TFTR and JET. A large R&D effort[27] was performed recently to determine the places in fusion devices where erosion of PFC occurs or where the eroded material is deposited and to assess the hydrogen trapped in various locations. In-situ diagnostics[28] for deposited layers were developed to gain information on a shot by shot basis. The trapping of tritium in ITER and future FPPs is of concern as it may require a shutdown until the internal tritium inventory is reduced again to acceptable values. In ITER Be will be the plasma facing material and CFC tiles will be used for divertor areas with the highest heat flux, whereas the other surfaces of the divertor are covered with W tiles. During plasma operation, Be will be eroded, deposited on and re-eroded from the CFC tiles and will come in contact with W. The coverage of the divertor tiles with Be will be far from homogenous. Further, new very brittle phases can occur with W and the melting point of these new "alloys" decreases drastically by addition of Be. With graphite present the estimate of the tritium inventory will be difficult. Clearly detritiation techniques of the plasma facing components and the high heat flux components are required and a large R&D effort is still ongoing. Detritiation techniques addressed in the EU include the use of lasers[29], flash lamp[30], plasma torch, etc. Remote detritiation exercises of tiles inside the JET vacuum vessel were also performed successfully[31], but on a limited scale. During these processes the creation of dust needs to be avoided as this would contribute to the existing dust issue in ITER. In summary, the inventories of tritium and dust inside the vacuum vessel will increase with the number of shots and both are to be restricted due to safety considerations. A large R&D effort with good results has been made in existing machines, further R&D is needed. The final proof will be made during the ITER H-phase with new diagnostics to assess the hydrogen inventories in-situ.

VI. TRITIUM IN IFMIF

IFMIF[32] is the second pillar of the fast track to fusion power. IFMIF will produce a fusion-like neutron spectrum and will be able to produce 50 dpa (displacement per atom) per full power year with the correct appm He to dpa ratio of 10 to 12. The operation of IFMIF is essential for validation of the structural and functional materials to be used in FPPs and for their licensing. 40 MeV deuterons at 250 mA are shot into a fast moving Li jet to remove the 10 MW heat of the beam. In the (d,Li) stripping reactions neutrons are produced and also tritium which is gettered by the Li. Also, via the same process as in the BBs, a small fraction of these neutrons produces tritium via fission of the Li in the jet and in the so-called Li quench tank. The production of tritium in IFMIF under normal operation conditions (10 MW) is approximately 7.5 g/fpy (Ref. 33). To avoid tritium releases during possible Li spillages the tritium needs to be removed continuously from the Li via hot getters. The required tritium recovery system for IFMIF is to be tested as part of the activities within the BA. Most of these Li target activities will be performed by JA, but the EU is offering contributions to the various clean-up systems in the Li loop.

VII. TRITIUM IN WASTE OF FUSION DEVICES

The handling of tritium in waste is important during operation of machines using tritium, especially during their decommissioning. Papers discussing the R&D in the EU on conditioning various types of tritium contaminated waste produced and collected in fusion devices for long term storage were recently published[34, 35].

VIII. CONCLUSIONS

In the EU R&D on tritium related issues has been performed mainly at the Tritium Laboratory (TLK) of the Forschungszentrum Karlsruhe; in the Active Gas Handling System (AGHS) at JET in Culham, U.K.; at CEA Cadarache, France; ENEA Brasimone and Frascati, Italy; and SCK-CEN in Mol, Belgium. These facilities were the main contributors to the successful operation of the tritium campaigns at JET and the leading role of the EU in the design of the ITER FC. In future these facilities will help i) to address the questions and requirements of licensing authorities or originating from operation of the various tritium processing components in ITER and ii) to develop more efficient and safe processes for tritium FCs as the conditions for tritium control and management will become even more stringent in future FPPs.

ACKNOWLEDGMENTS

This work, supported by the European Communities, 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. D. MAISONNIER et al, "A conceptual study of commercial fusion power plants, Final report of the European fusion power plant conceptual study (PPCS)", EFDA-RP-RE-5.0 (2005).
2. P. SARDAIN et al, "The European power plant conceptual study: Helium-cooled lithium-lead reactor concept", *Fus. Eng. Des.* 81 (2006) 2673-2678.

3. A. AIELLO et al, "Hydrogen isotopes permeability in Eurofer 97 martensitic steel", *Fus. Sci. Tech.* 41 (2002) 872-876.
4. E. SERRA et al, "Influence of traps on the deuterium behavior in the low activation martensitic steels F82H and Batman", *J. Nucl. Mater.* 245 (1997) 108-114.
5. G.A. ESTEBAN et al, Hydrogen transport and trapping in EUROFER'97, Proc. ICFRM-12, Santa Barbara, California, 2005.
6. M.M.W. PEETERS et al: In-pile Tritium release behaviour of Lithium Metatitanate produced by Extrusion-spheroidisation-sintering process in EXOTIC-9/1 in the High Flux Reactor Petten, *Proc SOFT Warsaw 2006*.
7. G. KIZANE et al, "Behaviour of tritium in breeding materials" *Fus. Eng. Des.* 75-79 (2005) 896-901.
8. F. REITER, "Solubility and diffusivity of hydrogen isotopes in liquid Pb-17Li", *Fus. Eng. Des.* 14 (1991) 207-211.
9. A. AIELLO et al, "Determination of hydrogen solubility in lead lithium using sole device", *Fus. Eng. Des.* 81 (2006) 639-644.
10. F. GABRIEL et al, "2D element modelling of tritium permeation for the HCLL DEMO blanket module", *Proc SOFT, Warsaw 2006*.
11. J. KONYS et al, Corrosion behavior of EUROFER steel in flowing eutectic Pb-17Li alloy ", *J. Nucl. Mater.* 329-333 (2004) 1379-1383.
12. A. MÖSLANG et al, "Production and characterization of titanium beryllides for HIDOBE irradiation", *J. Nucl. Mater*. 367-370 (2007) 1069-1072.
13. *Fus. Sci. Tech.* 41 (2002) 303-1174; 48 (2005) 1-806.
14. D. MURDOCH et al, "ITER Design Review; tritium issues", *this volume*
15. Y. POITEVIN et al, "The Test Blanket Modules project in Europe: From the strategy to the technical plan over next ten years", *Proc SOFT, Warsaw 2006.*
16. J-F. SALAVY et al, "Overview of the last progresses for the European test blanket modules projects", *Proc. SOFT,* Warsaw 2006.
17. M. DALLE DONNE et al, "European HCPB Test Blanket. ITER Design Description Document", Dec.1996, *FZKA 5891*, April 1997, FZK Karlsruhe.
18. Y. KAWAMURA et al, "Extraction of hydrogen from water vapour by hydrogen pump using ceramic protonic conductor", *Fus. Sci. Tech*. 45 (2004) 33-40.
19. N. BEKRIS et al, "Cold trapping of traces of tritiated water from the helium loops of a fusion breeder blanket", *Fus. Eng. Des.* 58-59 (2001) 423-428.
20. S. BELOGLAZOV et al, Semi-technical cryogenic molecular sieve bed for the tritium extraction system of the TBM for ITER, *Fus. Sci. Tech*. 48 (2005) 662-665.
21. A. AIELLO et al, "TRIEX facility: an experimental loop to test tritium extraction systems from lead lithium" *Proc SOFT, Warsaw 2006.*
22. G. DELL'ORCO et al, "HeFUS3-European Helium cooled blanket test facility for DEMO", Proceedings of the 19th SOFT, Lisbon, Portugal, *Fus. Tech.* 1996, 1335-1338.
23. B.E. GHIDERSA et al, "Helium Loop Karlsruhe (HELOKA): A valuable tool for testing and qualifying ITER components and their He cooling circuits", *Fus. Eng. Des.* 81, 8-14 (2006) 1471-1476.
24. M. GLUGLA et al, "Tritium tests with a Permcat reactor for isotopic swamping", Proceedings of the 19th SOFT, Lisbon, Portugal, *Fus. Tech.* 1996, 1193.
25. S.A. BIRDSELL et al, "Ultrahigh Tritium Decontamination of Simulated Fusion Fuel Exhaust using a 2-Stage Palladium Membrane Reactor", *Fus. Tech*. 30 (1996) 905.
26. M. NISHI et al, "Study on tritium accountancy in fusion DEMO plant at JAERI", *Fus. Eng. Des*. 81 (2006) 745-751.
27. M. MAYER et al, "Further insight into the mechanism of hydrocarbon layer formation below the divertor of ASDEX Upgrade, *Nucl. Fusion* 46 (2006) 914-920.
28. H.G. ESSER et al, "Post mortem analysis of a JET quartz microbalance system », *J. Nucl. Mat.* 363-365 (2007) 146-151.
29. C.H.SKINNER et al, "Tritium removal from JET and THTR tiles by a scanning laser", J.Nucl.Mater. 313-316 (2003) 499-503.
    B.SCHWEER et al, "Laser desorption of deuterium retained in re-deposited carbon layers at TEXTOR and JET, J.Nucl.Mater. 337-339 (2005) 570-574.
30. K.K.J. GIBSON et al, "The removal of co-deposited hydrocarbon films from plasma facing components using high-power pulsed flaslamp irradiation", J.Nucl.Mater. 337-339 (2005) 565-569.
31. C. GRISOLIA et al, "In-situ tokamak laser applications for detritiation and co-deposited layers studies", *J. Nucl. Mater.* 363-365 (2007) 1138-1147.
32. IFMIF Comprehensive Design Report (CDR), printed January 2004.
33. S.P. SIMAKOV et al, "Assessment of the ^{3}H and ^{7}Be generation in the IFMIF lithium loop", *J. Nucl. Mater.* 329-333 (2004) 213-217.
34. A. C. BELL et al, "Detritiation processes needed for JET operations and their wider applicability". *Fus. Sci. Tech.* 41 (2002) 626-631.
35. S. ROSANVALLON et al, "Tritium related studies within the JET fusion technology work programme", *Fus. Sci. Tech.* 48 (2005) 268-273.