INTERNATIONAL WORKSHOP ON LIQUID METAL BREEDER BLANKETS BOOK OF ABSTRACTS

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CIEMAT, Madrid (Spain)
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Session 1:

PROGRAMMES

IEA Parties Liquid Metal Breeding Blanket R&D Programmes
R&D Program in support of the HCLL-TBM design and testing

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An intense R&D program has to be developed in the next years in order to cope with the HCLL-TBM testing objectives in ITER.
All the major topics will have to be touched in implementing this R&D program: TBM fabrication technology, thermal-hydraulics investigation coupled with MHD effects, instrumentation development, optimization of technology of tritium extraction from PbLi and tritium processing in gas phase, physical-chemistry of the PbLi-materials interaction, consolidation of know-how on high pressure He technology.
In addition, an important part of the R&D effort will have to be focused to the implementation and preliminary validation of models that will be used as tools to plan the experimental campaigns of the TBM testing in ITER.
In this presentation a plan summarizing the evolution of these activities over the next years is provided, emphasizing for each of them the gap to be closed in order to achieve a sound design of the HCLL-TBM and, at the same time, to be compliant with the TBM testing objectives. Relevance of these activities with respect to the DEMO perspectives will be also presented and discussed.
Overview of Japanese R&D Programme on Liquid Metal Blanket

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TBM program is regarded as a most important fusion technology programmes using ITER from an engineering view point in Japan. While Water cooled solid breeder (WCSB) concept is strongly developed as a primary candidate of the ITER TBM in Japan, the JA party also emphasizes basic studies of various advanced blanket concepts so that fusion be further improved as an attractive energy source. Some of the concepts are studied as TBM programs by the ITER Participant Teams, and JA proposes to participate as a Partner for other TBM concepts, which will be lead by other Parties with international cooperation. LiPb based liquid metal blankets are considered for TBMs by other parties. Also, Li-based, and Molten Salt blankets are studied. All of these concepts are mainly studied in universities in Japan.

JA has expressed an interest to participate as the Partner for the LiPb based TBM, when it will be lead by other Parties. Various fundamental researches such as, compatibility tests between LiPb and F82H, tritium control technology such as inventory, permeation, and recovery, and related SiC technology are studied experimentally. MHD and thermofluid, as well as neutronics are extensively studied by both numerical and analytical researches. Some studies are conducted under the TITAN US-Japan collaborative program. National Institute of Fusion Science, Kyushu University, the University of Tokyo, and Kyoto university have these research activities. Kyoto university is operating a LiPb-He dual coolant loop with various test capabilities at the temperature above 900 degree.

JA have the studies on Li- based blanket such as Li-V or helium cooled liquid lithium, and is ready to participate as a partner if other Party will take a lead. R&Ds on Vanadium alloy has been conducted at the NIFS and universities. Liquid lithium is studied in the universities mainly for the application to IFMIF. The conceptual design of the TBM has been performed.

Besides liquid metal, basic studies on molten salt blanket concepts have been extensively conducted under US-Japan collaboration program, and conceptual design was made for Helical reactor.
Overview of US Liquid Metal Blanket R&D Activities
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In the US, liquid Metal Blanket R&D is carried out as part of a broader research program on Fusion Nuclear Science and Technology (FNST). The US FNST research activities continue to focus on the most important technical issues with high scientific content and substantial potential for an improved vision of a fusion energy system. The US science-based framework for development of FNST includes two broad stages: 1- an extensive R&D program of modeling and experiments in non-fusion facilities, e.g. laboratory experiments, fission reactors, and accelerator based neutron sources, and 2- Integrated testing of fusion nuclear components in DT plasma-based testing facilities in which a true fusion environment can be simulated. Analysis of the ITER design and operating conditions show that ITER can be utilized only to obtain data on early prompt nuclear responses. While ITER TBM is important, another new dedicated DT fusion facility is required to address FNST engineering feasibility issues and nuclear component development- such a facility is called Fusion Nuclear Science Facility (FNSF). The US is now exploring design options for FNSF and requirements for testing blankets, PFC, Tritium, and other nuclear components in the fusion environment. The Dual Coolant lead-Lithium (DCLL) blanket is currently the reference liquid metal blanket concept as an example of a pathway toward high temperature operation while using reduced activation ferritic steel (RAFS). Current key research areas in the US on liquid metal blanket R&D include:

- MHD flow Dynamics for liquid metal blankets (UCLA; Hypercomp)
- Interfacial phenomena, MHD Heat and Mass Transfer, Corrosion, Tritium Transport (UCLA; SBIRS: HyPerCom, Ultramet, Hypertherm)
- Compatibility, Corrosion experiments (ORNL, SBIRs: Ultramet, Hypertherm)
- Tritium permeation and recovery (INL, UCLA)
- Safety analysis and modeling (INL)
- Flow-Channel Insert (FCI) SiC material/component development & properties (ORNL, PNL; SBIRS: Ultramet, Hypertherm)
- Irradiation effects in RAFM steels and SiC (Materials Program )
- Integrated modeling / Virtual TBM (UCLA, UW; SBIR: Hypercomp)
- Beryllium armor joining to RAF/M steel (UCLA)

Details of progress on key technical aspects of the US R&D for Liquid Metal Blankets will also be discussed in several other presentations in this workshop (See, for example, presentations by Calderoni, Merrill, Smolentsev, and Wong)
Overview of the LLCB TBM R&D activities in India

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India has proposed Lead–Lithium cooled Ceramic Breeder (LLCB) as the blanket concept for its DEMO reactor. The LLCB blanket concept consists of lithium titanate as ceramic breeder (CB) material in the form of packed pebble beds and Pb–Li eutectic as multiplier, breeder, and coolant for the CB zones. The outer box is made of Indian Reduced Activation Martensitic Steel (IN-RAFMS) and cooled by high-pressure helium. The LLCB TBM will be tested from the first phase of ITER operation (H-H phase) in one-half of a ITER port no-2. The tests in ITER include the simultaneous function of all subsystems including the TBM as well as its ancillary system. The tritium produced in Pb–Li and ceramic breeder zones will be extracted by separate external ancillary systems. The R&D activities have been initiated in all critical areas related to TBM development at various research centers in India. The primary focus of LLCB TBM R&D is in Lead-Lithium technologies, Lithium titanate pebble manufacturing, tritium extraction technologies and development of Reduced Activation Ferritic Martensitic Steels and its fabricability studies. This paper will provide a overview of LLCB TBM R&D activities in India.

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RF for many years has been developing Li-V breeding blanket for tests in ITER after revealing its high potential performance in DEMO-S reactor study of 1998-2000. Due to lack of financial resources and adequate support from international community, RF has been switching since 2009 to the development of lead lithium ceramic breeder (LLCB) test blanket module (TBM) proposed by Indian TBM Team. It uses lead lithium eutectic (LL) and ceramics as breeder materials and helium and LL as coolants. Conceptual design of LLCB blanket for RF DEMO-S reactor was finished to show some advantages over pure LL blanket. Following this analysis the TBM concept for ITER tests is being developed as a support to Indian TBM Team. R&D program for LLCB TBM has been worked out including the following aspects: (1) Corrosion tests of ferritic martensitic steels in LL: (1.A) Rotating disc facilities for screening tests of steels with different type of electro-insulating/anti corrosion barriers. (TLL= 350-550°C, VLL up to 0.5 m/s), (1.B) LL loop corrosion tests. (2) LL technology development (LL loop): (2.A) Development of sensors for oxygen and hydrogen isotopes., (2.B) Development of technique for control of impurities. (3) MHD/HT tests of TBM elements and TBM mock-ups: (3.A) Existing facility with NaK loop (B~ 1T in a volume 600x120x170 mm, LM flow rate up to 20 m3/h), (3.B) Transformation of this facility into LL one. Plans for a magnet with 3-4 T in a volume 800×500×150 mm., (4) Development of Tritium Production Rate monitoring system for TBM., (4.A) Facility (on the basis of IVV-2M reactor) for generated tritium measurements by samples delivery from TBM to out of reactor measurement system. (5) Development of in-reactor loop with LL for tests of tritium production and extraction, tritium permeation and structure materials compatibility. (6) Hydrogen isotope permeation through structural materials study: (6.A) Out of reactor (100-500°C, gas pressures 3-130 kPa) and in-reactor (IVV-2M) facilities.

Separate presentations on these R&D topics are foreseen at the Workshop.
Progress in Lithium-Lead breeder TBM R&D in China

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The liquid metal LiPb breeder blanket concept has been explored extensively in China as a design option of DEMO blankets for fusion power reactors. And a dual functional lithium–lead (DFLL) test blanket module (TBM) concept has been proposed for testing in ITER to demonstrate the technologies of DEMO LiPb blanket concepts. The conceptual design and preliminary performance analysis for the DFLL-TBM have been completed. And a series of R&D Activities relevant to the LiPb blanket design and technology development have been carried out recently in China. Design and analysis tools for the fusion reactor and blanket have been developed in the area of the physics and engineering calculation, computer modeling and simulation, database management system and virtual fusion reactor plantform, etc. A multi-functional integrated four-dimensional neutronics simulation system, named VisualBUS, with the main functions of automatic modeling, neutronics/thermohydraulics/MHD coupled calculation, visualized analysis and virtual roaming is being developed. The database management system, including nuclear data, material data, component data and plasma data, is being performed. The progress on the blanket technology development mainly focused on the China low activation martensitic steel (CLAM) development and TBM fabrication, liquid LiPb loops construction and experiment, functional coatings development and SiCf/SiC composites fabrication, and the construction of high intensified neutron generator (HING). Ton level melting of CLAM steel has been obtained already, and the exploration for the fabrication and manufacture technique of reduced-size TBM are being performed. Series LiPb loops with thermal convection and forced convection under different temperatures have been constructed and operated for the corrosion experiment and MHD test. Also, the high intensified neutron generator with D-T neutron rate up to 1010–1013 n/sec is being built.

S1-I6
KO Programme on LM Breeder Blankets

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1. Introduction

Korea has proposed and designed a Helium Cooled Molten Lithium (HCML) Test Blanket Module (TBM) to be tested in the International Thermonuclear Experimental Reactor (ITER) [1-4]. Ferrite Martensitic (FM) steel is used as the structural material and helium (He) is used as a coolant to cool the first wall (FW) and breeding zone. Liquid lithium (Li) is circulated for tritium breeding, not for the purpose of cooling.

The main purpose for developing the TBM is to develop the design technology for the DEMO and fusion reactor and it should be proved by experiment in the ITER. Therefore, we have developed the design scheme and related codes including the safety analysis for obtaining the license for testing in the ITER. Several technologies to be installed at the ITER have been developed simultaneously, fabrication, a liquid breeder, He cooling, tritium extraction and so forth.

2. Development of the design scheme and related codes

For the design of the TBM itself, a 3D CAD was developed with CATIA V5 and thermal-hydraulic/mechanical performance was evaluated with ANSYS codes; -CFX and -mechanical classic version. In the neutronic analysis, MCCARD and MCNP were used and ATILLA code has been prepared for the 3D analysis. For the accident analysis, decay heat and activation materials were obtained through MCCARD and MCNP codes. Transient performance after-accident was evaluated with ANSYS and MARS-GCR (Multi-dimensional Analysis of Reactor Safety for Gas Cooled Reactor) and GAMMA (GAs Multi-component Mixture Analysis) codes for the TBM temperature and coolant behavior, respectively. For the liquid breeder, the safety analysis was not performed because the related codes, such as the MARS-FR and GAMMA-FR developments, were not completed yet.

3. Development of the key technologies for developing the KO TBM

The FW is an important component which faces the plasma directly and, therefore, it is subjected to high heat and neutron loads. The FW is composed of a beryllium (Be) layer as a plasma-facing material and FM steel as a structure material. In order to develop the fabrication technology for a TBM structure, several mock-ups, especially for the FW channels, were fabricated with a HIP (Hot Isostatic Pressing), which was developed in a similar way as the development of the ITER blanket FW in Korea. For joining FMS to FMS, mock-ups were fabricated with an HIP (1050 °C, 100 MPa, 2 hours). For joining Be to FMS, two mock-ups were fabricated with the same method (580 °C, 100 MPa, 2 hours) using different
interlayers. Then, in order to evaluate the integrity of the fabricated mock-ups, they were tested at the high heat flux (HHF) test facilities, KoHLTs (Korea Heat Load Test) under 1.0 and 0.5 MW/m² heat fluxes of up to 1000 cycles. Since there was no delamination or failure in the mock-ups, it could be concluded that the fabrication methods were successful.

The performance analysis for thermal-hydraulics and safety analysis for an accident by loss of a coolant for the KO TBM have been carried out with a commercial CFD code, ANSYS-11 and system codes such as MARS-GCR and GAMMA. In order to verify these codes, especially for the ANSYS-11 and GAMMA by the comparison with the experimental results, a basic thermal-hydraulic test with a high pressure nitrogen gas loop up to 6 MPa pressure and 1,000°C temperature was performed. In the experiment, a TBM FW mock-up made from the same material as the KO TBM, FM steel, was used and the test was performed under the conditions of pressures of 20 and 36 bar and flow rates of 0.75 and 0.92 kg/min. As one-side of the mock-up was heated to 230°C, the wall temperatures were measured by installed thermocouples. Experimental results show a strong parity with the codes’ ones in the test conditions. An additional test with higher pressure and temperature has been prepared for the future.

In order to develop the liquid breeder technology, the analysis methods of its behavior under electro-magnetic field (MHD, magneto-hydrodynamics), compatibility with structural material, and key-components such as the electro-magnetic pump are essential. The experimental loop with PbLi was established at KAERI for performing the essential experiments mentioned above. The design parameters are as follows; over 250°C of temperature, 0.5MPa of pressure, up to 60 lpm of flow rate, 2 T of magnetic field in the magnet.

REFERENCES

Session 2:

LiPb PRODUCTION

LiPb, QA production and characterization
Application of MHD Technology for Production of Lead/Lithium Eutectic Alloy

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Abstract

Lead based eutectic alloys such as lead-lithium, lead-bismuth and lead-gold are considered as perspective candidate working materials for different nuclear plants both from point of view of nuclear physics and as cooling and heat transfer agents. At the Institute of Physics of University of Latvia (IPUL) the technology and corresponding equipment basing on magneto-hydrodynamic effects for mixing of liquid metals for production of Pb15.7at%Li eutectic alloy have been developed and used successfully for its production.

The main objective of the work was the development of technology for production of large quantities of Pb15.7%Li eutectic alloy having relatively high purity and at the same time being advantageous and competitive for economics. As a benchmark the production 3 ÷ 5 tons of alloy per month was accepted. The first planned and produced eutectic alloy quantity was 6 tons. The bigger part of it was delivered for the tests on European Breeding Blanket Test Facility at ENEA, Brasimone, Italy. The production of eutectic alloy was realized in Latvia by Hidrovats Ltd. using the equipment and technology developed at IPUL. Advantageous technology for production of chemically pure alloy at acceptable cost realized by Hidrovats Ltd. will be recommended for industrial production.
Production of Pb-Li eutectic: cover gases or molten salts during melting?

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In order to produce eutectic Pb-Li alloys, it has been developed a reactor with the aim of melt protection via cover gases or molten salts/cover gas mixture. All tests were carried out using a Pb-Li alloy (18.5 at. % Li) establishing the loss of Li for the fusion process. In these studies we have determined the influence of the furnace atmosphere, slag, crucible material, and thermal cycles on the Li content of the obtained alloys. The temperatures of the isotherm of the thermal cycles applied were in the range of 350 - 800 °C. The crucibles material used were C, SiO2 and SiC. The temperature and melt protection conditions selected induced microstructural and compositional changes into the alloy, which could enable or not, its use as material for nuclear applications. The changes originated by the different conditions used in the fusion process were evaluated by OM (Optical Microscopy), SEM (Scanning Electron Microscopy), DSC (Differential Scanning Calorimetry), XRD (X-Ray Diffraction), and ICP-MS (Inductively Coupled Plasma Mass Spectroscopy).

Cover gases as N2, can lead to modifications in the Pb-Li alloy due to the Li3N generation, which could act as protective film avoiding the loss of Li from the melt. On the other hand, the eutectic molten salts as LiCl-KCl have a higher protective power than the cover gases. However, the difference in chemical potential of the Li between both inter-phases, generate the removal of the Li from the Pb-Li alloy to the salt and their subsequent reaction.
Session 3:

T TRANSPORT PROPERTIES in LiPb

Tritium transport properties in LiPb eutectic (diffusivity, solubility, mass-exchange coefficients and MHD effects)
Isotope effects on solubility and diffusivity of hydrogen isotopes in Li-Pb eutectic alloy

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The present study is designed to estimate the mass-transfer properties of tritium (T) in Li-Pb alloys more correctly as a promising blanket material of fusion reactors. The experimental study is performed using a permeation cell container, where single- or two-components of hydrogen isotopes permeate through static Li17Pb83 eutectic alloy layers of 1 cm, 2 cm and 3 cm in thickness placed on a α-Fe plate. The concentrations of hydrogen isotopes permeated are detected by gas chromatography and a quadrupole mass spectrometer. Diffusion-limiting permeation is confirmed. Solubility, diffusivity, permeability and isotope exchange rates of hydrogen isotopes in Li-Pb are determined using the permeation cell. Parts of the results were presented in ISFNT-9 and ISFRM-14 of 2009. Topics of discussed and quantified there were (1) solubility, diffusivity and permeability of H or D in Li-Pb, (2) comparison between previous data and our ones, (3) isotope effects among hydrogen isotopes, (4) wettability at interface between Li-Pb and solid wall, (5) impurity effect on mass-transport properties, (6) effects of different Li compositions in Li-Pb alloys and chemical activity of Li in Li-Pb. Our recent data on the above topics are added in the Madrid workshop. We focus on simultaneous permeation of two-component hydrogen isotopes and clarify isotopic differences in diffusivity, solubility and permeability. The isotope effect of permeability is almost independent of temperature and its ratio between H and D in Li-Pb permeability is 0.7, which equals to a square root of its mass ratio. The value corresponds to the classical isotope effect. Therefore, the value of 0.58 is expected for the ratio between H and T in Li-Pb. Since the chemical activity of Li in the Li-Pb alloy is very small, the isotope effect is considered to equal to classical one. At present, we set up a new experimental apparatus using a system of fluidized LiPb, where hydrogen permeates Li-Pb flowing in tube. The details of the experimental apparatus are presented in the Madrid workshop.

S3-I1
The interaction of tritium with the lithium bearing coolant and breeder material is one of the most important physical processes in determining the feasibility and the attractiveness of a fusion energy system because it is fundamentally linked with all aspects of plant operation, from fueling (tritium breeding ratio, tritium availability, etc) to power extraction (heat transfer capability, heat cycle efficiency, etc) to safety (tritium inventory, tritium release, etc). This work presents a critical assessment of R&D programs aimed at the determination of tritium transport properties in lead lithium alloys, with focus on solubility. Completed activities in the US, Europe and Japan available in the literature are discussed with the aim of understanding the technical reasons for the wide scatter of collected results. Ongoing programs are introduced and updates on findings are presented when available. In particular, the experimental work started in the US within the INL Fusion Safety Program is presented in detail. Updated results on hydrogen solubility tests and the necessary modification of the facility for tritium experiments are discussed. Finally, modelling activities in support of data analysis are introduced as well as preliminary analysis for the conceptual design of a forced convection loop to test tritium extraction concepts.
Tritium solubility (Sieverts’ constant) in LiPb eutectic is a key magnitude to assess on T-control at Breeding Blanket (BB) level and for the design of BB auxiliary systems. Experimental values in current literature show a very large scattering with a broad band of two orders of magnitude. The low value of H-isotopes solubility in LiPb eutectic and parasitic effects in the measurement can explain the difficulties for developing accurate instruments sensitive enough to finish with these discrepancies.

Two different absorption-desorption measurements (Sievert’s reversible method) for H$_2$ and D$_2$ in the temperature range of (300-500ºC) for two different samples (one provided by commercial Stachow and the manufactured at IPUL-labs) are proposed from measurements done at UPV and CIEMAT (using a commercial PCT2000PRO® equipment manufactured by SETARAM Inst. Co., France). Results are compared and discussed. Parasitic effects in these measurements are identified and tentatively corrected. Both measurements confirm low solution energies for hydrogen in lead lithium are confirmed in coherence with global solubility database in literature.

Key aspects of the Quality Assurance for the solubility measurement are discussed together with issues to be solved for the needed standardisation of measurement of Hydrogen solubility in lead-lithium and other low hydrogen absorbing materials.

Ongoing activities towards the design of a multipurpose measurement are reported and discussed.

S3-O2

This work is founded by the Spanish National Project on Breeding Blanket Technologies TECNO_FUS through CONSOLIDER-INGENIO 2010 Programme.
Session 4:

TRITIUM CONTROL & SENSING TECHNOLOGIES

Tritium sensing and other control and processing technologies in LiPb
Tritium sensing and processing technologies in lead-lithium

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In the Helium Cooled Lithium Lead (HCLL) blanket for DEMO, where a liquid metal (eutectic lead-lithium alloy) is used as a breeder material, a correct and reliable management of tritium is a critical issue. Tritium relevant technologies will be tested in HCLL-TBM (Test Blanket Module) in ITER with the main aim to validate modelling tools and to getting experience with the different processing systems.

The principal auxiliary circuits of HCLL blanket aimed to process tritium are TES (Tritium Extraction System) and CPS (Coolant Purification System). TES is aimed to extract tritium from the flowing breeder and to process tritium up to the delivery to the systems belonging to DEMO and ITER inner fuel cycle, while CPS has to extract the permeated tritium from the primary circuit, keeping controlled the chemistry of the coolant.

Although TES could be based on different technologies (e.g. gas liquid contactors, getters or permeators) the determination of hydrogen isotopes concentration in the liquid metal is mandatory for its correct design, testing and operation.

Main options for HCLL TES and CPS technologies will be presented in this work together with sensors for tritium measurements in lead-lithium.

S4-I1
Hydrodynamic Study of Wire-mesh Corrugated Packed Bed for Tritium Extraction from Lead-Lithium (PbLi)
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Two types of columns were investigated for Indian LLCB tritium extraction system mass transfer performance evaluation from liquid lead-lithium (Pb17Li) breeder. It was analyzed that packed columns are having higher mass transfer efficiency than bubble columns. This was shown in our earlier paper on tritium extractor design [Ref.1]. In lieu of that hydrodynamic studies of packed bed have been taken up for lead-lithium-helium system to find the extractor liquid hold-up and pumping power requirement. Further, comparison of hydrostatic pressure drop for bubble column and packed column has been done to find the effect on system pumping head requirement. It is found from hydraulic studies that bubble columns not only require higher pressure drop but also higher liquid hold-up in the system compared to packed column. Higher holdup will result in higher equilibrium time. The hydrodynamic study of generalized corrugated wire mesh structured packed bed with countercurrent gas-liquid flow has been chosen for comparison. Semi-empirical correlations with specific packing geometry has been used to calculate hydrodynamic parameters such as flow regime identification, static and dynamic liquid hold-up and overall pressure drop. Packing geometry, hydraulic parameters like hydraulic diameter, effective interfacial area and gas liquid superficial velocities are identified and correlated for flow regime identification of Indian LLCB system.

Ref.1: ‘Experimental Design of Tritium Extraction loop from Lead-Lithium Eutectic’, Sadhana
Mohan, Kalyan Bhanja, K.C. Sandeep, Fusion Engineering and Design (under press)
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S4-O1
Tritium monitoring in lead-lithium eutectic and ceramic ITER TBM and DEMO breeding blanket

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The experimental demonstration of tritium breeding represents a major milestone for controlled fusion reactor engineering. Tritium Breeding Ratio (TBR) is a main parameter characterizing the breeding process as the ratio of the amount tritium produced in the fusion reactor to the amount of tritium that burned up in the reactor plasma.

A concept and lay-out diagram of tritium breeding monitoring and experimental estimation of the TBR in DEMO and ITER-TBM for ceramic and eutectic blankets are discussed.

Methods are proposed for the experimental estimation of the TBR and the tritium-breeding dynamic parameters in a Tritium Breeding Modules (TBM) of the ITER. They are based on tritium and neutron flux measurements under ITER plasma experiments and on the use lithium ortho-silicate, lithium carbonate, eutectic samples and the neutron detectors.

Different sample $^6\text{Li}$ isotope enrichments are used in the detectors. The detectors are conceived for the Tritium Breeding Zone (TBZ) of the TBM and their signals connected with the TBM monitor channels and the specific TBM Control Room in ITER. Pneumatic and mechanic methods are considered to deliver and to extract the samples for the TBM’s TBZ using the channels during plasma operational pauses.

Results of the channel parameter calculations and comparison of the pneumatic and mechanic systems are presented.
A fast and efficient recovery of bred tritium is a major milestone of tritium breeding technologies R&D for the demonstration of a fusion reactor tritium self-sufficiency. Diverse tritium recovery technologies from lead-lithium eutectic have been investigated with different degree of qualification. Permeator Against Vacuum (PAV) runs as a single-step process for tritium on-line recovery, acts as passive systems allowing to be thermally governed can be easily in-pipe integrated in LiPb loop systems and can be conceived with high compactness. An optimal design of a PAV requires a detailed hydraulic design optimization for established operational ranges (HCLL at low mm/s velocity or DCLL in the ranges of tens of cm/s).

Key design issues determining PAV efficiency under LiPb alloy in laminar regime are well-known.

An optimal PAV design is proposed with detailed design parameterization of tritium recovery efficiency at two velocity ranges from numerical simulation based on properly developed Openfoam® CFD code BelFoam® customized solver.

CIEMAT and SENER Ingeniería y Sistemas S.A. are collaborating to build up a technological demonstrator to qualify a technique to recover Tritium from LiPb at operational ranges of the HCLL TBM of ITER. The objective of this demonstrator is to qualify the technology of Permeation against vacuum for continuous in-line tritium recovery.

The aim of this paper is to show the experimental set up design for the experiment and the particularities of the trials.
Development of a H-concentration sensors in PbLi eutectic alloy based on solid state electrolytes

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Molten lithium lead eutectic (Pb15.7(2)Li) is under consideration for tritium breeding material as well as primary coolant in DEMO blankets and in the ITER Test Blanket Modules R&D Programs. The determination of hydrogen isotopes concentration in lead-lithium represents a key issue for tritium monitoring and safe control in blanket systems and results essential to experimentally certify tritium self-sufficiency.

Potentiometric hydrogen sensors for molten lithium-lead eutectic have been designed at the Electrochemical Methods Lab at Institut Quimic de Sarria (IQS) at Barcelona. These sensors are based on the use of solid state electrolytes in order to separate the working and the reference electrodes. Working as Proton Exchange Membranes (PEM) the main advantage of using this type of electrolytes is that they are selective and potentially provide nearly instantaneous H in-solution calibrated signals.

The following compounds have been synthesized in order to be used as solid state electrolytes in the hydrogen sensors: SrCe0.9Y0.1O3-α, Sr3CaZr0.9Ta1,1O8,55, SrCe0.95Yb0.05O3-α, CaZn0.9In0.1O3-α, Ba3Y(Ca1,18Nb1,82)O9-α. The surface characteristics of the powder and sintered ceramics have been studied by SEM-EDS.

In addition, potentiometric measurements of the ceramic elements synthesized have been performed at different hydrogen concentrations. In these first experimental campaign, a fixed and known hydrogen pressure has been used in the reference electrode.

Foundations, potentialities and first qualifications are proposed and discussed.

S4-O4

This work is founded by the Spanish National Project on Breeding Blanket Technologies TECNO_FUS through CONSOLIDER-INGENIO 2010 Programme.
RF Proposals for Liquid Breeders Containing Eutectic Lithium-Lead

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Lithium-lead is candidate tritium breeder for controlled fusion reactors. The material was discussed and investigated for Russian project reactors. Now lithium-lead is proposed for Russian DEMO.

Problem of tritium extraction from Li-Pb was a main target to RF research in a frame of DEMO activity. Molecular distillation and helium bubble methods were adopted for tritium extraction. Distillation column is placed in a container together with the primary cooling system. Tritium concentration is provided around 1-100 appm at exit of the system and accumulated on getter. The extraction system is discussed.

As for lithium-lead tritium permeation through metallic walls was a critical problem in RF fusion reactor designs. The permeation through stainless steel walls was measured by out-of-pile experiments with Li17Pb83 molten allow in the temperature range from 600 to 750K. Results of the experiments are presented.

New facility for investigation of hydrogen isotope permeation through stainless steels has been assembled and technology schema of it is presented.
Session 5:

Coatings & channel inserts

Insulating aspects, anti-corrosion coating, anti-permeation barriers and channel inserts development
Insulating Coating Development for Anti-corrosion and T-permeation Barriers

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Reduced ferritic-martensitic steels are foreseen in future fusion technology as structural material which has to withstand long-term exposure to the heavily corrosive environment of liquid lead-lithium breeder in the HCLL design. In earlier work corrosion testing has shown that material dissolution by Pb-15.7Li reaches dramatically values of about 300 - 400 μm/year at flow rates of > 0.1 m/s at 550°C. This large amount of corrosion products is transported in a closed system and will be deposited in form of precipitates at cooler and/or other critical system relevant positions. In the corrosion loop PICOLO these precipitates caused loop blockages after relatively short times, representing a high risk in running of a TBM in ITER.

Thus, to guarantee reliable function of TBM’s, anti-corrosion barriers have to be implemented. Previous studies (e.g. Hot-Dip Aluminisation) showed that Al-based coatings have such anti-corrosion and also T-permeation reduction behavior. However, industrially relevant coating technologies are missing for complex shaped components and the special requirements in Pb-15.7Li environment. Additionally, the deposited Al scale thicknesses had to be reduced by a factor of roughly ten compared to Hot-Dip process to fulfill low activation criteria. Electrochemical deposition of metallic films from aqueous systems is well known e.g. in noble metal technology, however, it will fail in aluminization due to the chemical activity of Al in water and the formation of e.g. hydroxides instead of metallic films. The development of Al deposition methods from non-aqueous systems succeeded in investigating two different types of electrolytes. The first one is a process, based on organic aromates as electrolyte and the second variant uses organic salt melts (ionic liquids), which are liquid at around room temperature. Both electrolyte types allow deposition of metallic Al scales in high reproducible and controllable thicknesses. Each type has specific characteristics and advantages e.g. in coating inner surfaces or undercutted surfaces. All deposited Al scales have to be heat treated afterwards to form the protective surface scales. The deposition, the developed heat treatment, the surface scale formation will be discussed and first corrosion testing in Pb-15.7 Li will be shown in comparison to Hot-Dip Aluminization process.

S5-I1
Lithium titanate, Li$_2$TiO$_3$, is a candidate tritium breeding material due to its reasonable lithium atom density, low propensity to activation, good tritium release performance and chemical stability. Lithium titanates as pebbles or pellets are to be used as tritium breeder in the test blanket modules. In the TBM environment, irradiation of lithium-6 isotope in Li$_2$TiO$_3$ by fast neutrons will generate energetic tritons (2.7 MeV) and helium ions (2.1 MeV) by the $^6$Li(n,α)$^3$H reaction. Molten lithium-lead eutectic alloy (Li0.17Pb0.83) is a proposed multipurpose material to be used as a coolant, neutron multiplier and also tritium breeder. The alloy is useful due to the low lithium and lead vapor pressures in the blanket operating temperature range of 350-550°C, moderate melting temperature of Pb-Li eutectic, and good heat-transfer properties under the condition of a fast circulation loop. India has proposed a LLCB (Lead-Lithium Cooled Ceramic Breeder) design for its TBM in ITER. The materials involved are Li$_2$TiO$_3$, Pb-Li and ferritic martensitic steel. In this particular LLCB design, it is imperative to ascertain the compatibility of lithium titanate pebbles with the liquid coolant material in order to predict the loop behavior in an event of accidental breaching of the containment. Initial thermodynamic predictions were made using FACTSAGE 5.6 equilibrium module.

Subsequent validation experiments were done in a differential thermal analyzer (DTA), high temperature XRD and furnaces. The equilibrium calculations were performed for temperatures up to 1000°C for possible interaction between Pb, Li and Pb-Li eutectic with the lithium titanate ceramic, for both possible rise in temperature due to reaction and for formation of new products. DTA experiments were also carried out up to 1000°C to assess the heat effects. Larger quantity of Li$_2$TiO$_3$ and Li0.17Pb0.83 mixtures were also heated in a steel crucible under argon in a silicon carbide heated furnace up to 1000°C. The DTA and the furnace products were subsequently analyzed by XRD to identify new phase formation. A high temperature XRD was done in He atmosphere to indicate any phase change in the vicinity of 1000°C. The results of thermodynamic calculations were correlated with experimental information. The interactions between the alloy coolant and the breeder bed material up to 1000°C are presented in the paper.
Corrosion study of Li-based blanket materials in RF

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Lithium and lead-lithium alloy are considered as coolants and tritium breeding materials for fusion blankets. Compatibility of structure materials - vanadium alloy and ferritic-martensitic steels with lithium and lead-lithium alloy correspondingly is an important issue. In 2009 tests of V-4Ti-4Cr specimens in lithium at Rotating Disk Facility (RDF) were started in IPPE. Besides, several specimens were tested in ampoule, i.e. in static lithium, at the same conditions.

Rotating disk method is known for material science specialists, but there are some questions related to comparison of the results with other methods of testing. This point is discussed in the report. It was shown that rotating disk results may compare with results of tests of a flat plate installed in a circulating loop.

Three RDF were constructed, filled with lithium, and the first stage of testing was carried out. Some results will be presented in the report. A number of specimens were exposed in ampoules with lithium at temperatures 450 and 600°C during 750 and 1500 hours. They were analyzed and the results will be presented too. Design of the RDF and ampoules, and a procedure of filling them with lithium are described. All tests were performed in molybdenum cups, to prevent a contact of lithium with stainless steel. The methods used for analysis of samples are gravimetric, roentgen structure, electron microscopy, micro hardness. Samples for RDF were shaped as flat disks and for ampoules – as foils 100 micrometers thick.

After testing of vanadium alloy in lithium molybdenum caps of RDF and ampoules were cleaned and prepared to test steel samples in lead-lithium eutectic alloy. Composite discs consisted of six sectors each for screening tests of ferritic steel with different types of electro-insulating/anti corrosion barriers are being prepared. Design of the composite disc is presented.

The presentation includes some schemes and photos of the RDF, ampoules and samples, and comments of the results obtained.

S5-O2
Development of Buoyancy Driven Lead-Lithium Loops for Corrosion Studies

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India has been allocated half-a-port by ITER to test its Lead-Lithium Ceramic Breeder (LLCB) concept of the TBM in which lead-lithium eutectic liquid acts as a coolant, neutron multiplier and breeder. The corrosion of the structural materials due to the flowing hot eutectic lead-lithium liquid coolant obviously needs to be evaluated for the safe functioning of the LLCB TBM. Keeping this need in view, buoyancy driven lead-lithium loops have been set-up in our labs. One loop that has been set up primarily for the corrosion studies; has one litre of the liquid lead-lithium alloy in it. Temperature of the hot leg of the loop is maintained at 500 °C and that of the cold leg of the loop at 400 °C. Velocity of the eutectic liquid, obtained in the loop, was about 6 cm/s. The first set of experiments is carried out on, P91 (the 9Cr-1Mo steel) to test the working of the loop. Plain coupons as well as tensile test specimens of P91 steel were placed in both the legs of the loop and evaluated, after various durations of exposure to the flowing liquid, for their microstructure, composition and tensile properties using techniques such as optical microscopy, SEM-EDAX, XRD, chemical analysis, and tensile & hardness testing. The changes in the composition of the eutectic liquid due to corrosion/erosion/dissolution of the samples in the two legs and the tubes carrying the eutectic liquid were also recorded at regular intervals of time. The present paper discusses the results of these efforts.
Silicon Carbide Tritium Permeation Barrier for Steel Structural Components


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Chemical vapor deposited (CVD) silicon carbide (SiC) does not lose tritium permeability resistance under radiation. Ultramet is developing a material system based on fully dense CVD SiC as the tritium barrier, bonded to ferritic steel using a SiC foam or a ductile metallic foam layer, which serves as a compliant interlayer between the steel and the CVD SiC tritium barrier. The thin, dense SiC barrier is vapor formed on and integrally bonded to one face of the foam structure, while the opposite face of the foam can be readily bonded to ferritic steel. The composite structure offers significant advantages over current aluminized coatings including high resistance to thermal- and radiation-induced stress, lower tritium diffusivity and solubility, and compatibility with molten lead-lithium breeder/coolant. Ultramet will optimize processing for bonding the CVD SiC tritium barrier/foam to ferritic steel. Mechanical and thermal cycling behavior is be characterized and prototype components will be constructed. Sandia National Laboratories, supported by Digital Materials Solutions, will define application requirements, establish a design, and perform deuterium permeation testing while Tritium permeation testing will be performed at Idaho National Laboratories.
Flow Assisted Corrosion and Erosion-Corrosion of RAFM Steel in Liquid Breeders

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Study on flow assisted corrosion (FAC) and erosion-corrosion of RAFM JLF-1 steel (Fe-9Cr-2W-0.1C) in liquid breeders of Li, Pb-17Li and Flinak was carried out. The corrosion tests in the flowing liquid breeders at 500°C and 600°C were performed in a stirring pot. The compatibility of JLF-1 steel with the liquid breeders was evaluated by the mass loss measurement of the specimens and metallurgical analysis of the surface after the tests.

It was found that the alloying elements of Fe and Cr in the JLF-1 steel were commonly dissolved into these melts [1, 2]. The mechanism of erosion-corrosion in the liquid metals was made clear as the removal of the corroded surface by the shear stress of the liquid metal flow [3]. The specimens exposed to Flinak flow showed the trace of pitting corrosion caused by electrochemical corrosion [2]. A compatibility model was developed which can evaluate the mass loss of the steel by the mass transfer: Δm_d, erosion-corrosion: Δm_e, and electrochemical corrosion: Δm_v as; Δm_{TOTAL} = Δm_d + Δm_e + Δm_v. The mass loss of the specimens in the corrosion experiments was evaluated by the model. The effect of erosion-corrosion on the total mass loss of the steel in the liquid metals could be larger than that of FAC estimated by mass transfer calculation. The mass loss of the steel by electro-chemical corrosion might be larger than that by the FAC in the Flinak.

Development of LiPb-SiC High Temperature Blanket

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High temperature LiPb liquid metal blanket combined with SiC/SiC cooling panel is designed and being developed in Kyoto University, combined with the concept of biomass-fusion hybrid reactor, that utilizes fusion heat to convert biomass to hydrogen and liquid fuel. The cooling panel made of SiC composite actively cooled with He can isolate the blanket vessel made of steel based materials, and thus achieves above 900 degree C of liquid metal to transfer high grade heat to the energy utilization plants. This paper describes recent results of the development of SiC-LiPb blanket in Kyoto University based on the above concept. The numerical simulations on neutronics and thermal hydraulics are conducted to investigate the feasibility to achieve the technical goal of this concept, and to optimize the designs. With the operation of the LiPb-experiments, valuable experiences on the actual dual coolant systems are also being obtained, that is quite unique in the world now. The LiPb-SiC-He combination suggests a very attractive blanket concept with realistic development path toward future fusion plants. The compatibility of SiC with LiPb was investigated as a major subject. Loop test as well as rotating disk tests of SiC composites and liquid LiPb up to 900 degree suggested good compatibility, however various interesting phenomena such as metal impurity plating and reaction between Sic and LiPb were observed. Permeability of hydrogen isotopes including tritium was measured with various types of SiC materials. The results suggest that SiC composite have very complicated permeation mechanism and paths for hydrogen isotopes. While powder and fibers of SiC exhibit very low diffusion coefficients for hydrogen, observed permeability varies orders of magnitudes depending on the binding materials and structure. Solubility of hydrogen isotopes in the LiPb was studied under US-Japan collaboration, and pressure and temperature dependence of the solubility and diffusivity of hydrogen was obtained. At the same time, tritium extraction process from liquid LiPb was studied and designed based on the solubility/diffusivity data. This study was performed as a part of the entire system design, because the requirements for tritium recovery system are determined from the tritium safety consideration for the fusion plant that utilizes tritium contaminated heat transfer media in secondary loops and beyond. The experimental data suggests that tritium recovery system that satisfies technical requirements for the high temperature LiPb blanket was possible with modest technical difficulty.
It should be noted that liquid LiPb blanket concepts requires heat exchange processes for LiPb and other heat transfer media such as water or He in order to utilize the heat transferred with LiPb. In Kyoto university, intermediate heat exchanger between LiPb and He was developed with SiC composite, and demonstrated with LiPb-He dual coolant loop up to 950 degree C. The technology tested with this dual loop also indicates that considerable fraction of the technology for LiPb-SiC blanket were common, and thus verified with hands on operation experiences.

The above experimental as well as numerical studies suggest that the LiPb blanket can achieve the extraction of high temperature heat in the near future. The authors propose a development path to realize fusion energy with the biomass-fusion concept as a possible option for the near future target.

S5-O6
Session 6: MODELLING

Mathematical model developments (included safety, irradiation issues and nuclear data)
A key mission for International Thermonuclear Experimental Reactor (ITER) is the integrated testing of blanket concepts suitable for demonstrating fusion power production. Testing in ITER will be accomplished by inserting prototypical blanket modules into designated ports. Because ITER will be a licensed nuclear facility, these test modules will also undergo licensing review and certification. The concept developed in the US is the Dual Coolant Lead Lithium (DCLL) Test Blanket Module (TBM). In order to license this module, a preliminary safety report (PrSR) was completed and submitted to the ITER International Organization (IO). This PrSR presents a comprehensive view of a US DCLL Test Blanket System (TBS) preliminary safety assessment. The work for this PrSR was led by the Idaho National Laboratory (INL) Fusion Safety Program, with contributions from General Atomics, University of California-Los Angeles, and other members of the US TBM team. The MELCOR, TMAP, DKR-PULSAR, and QADMOD computer codes were used to analyze the operational and accidental safety of the DCLL TBM concept. This paper gives a brief overview of some of the predictions obtained for the DCLL TBM from these computer codes, and summarizes some of the safety findings from the DCLL TBM PrSR.

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Magnetic Field Effects on Three Dimensional Stability of Natural Convection Flows in Differentially Heated Cavities

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The parametric study of two dimensional free convection flow in a square cavity is extended in order to cover three dimensional disturbances and their effect on heat transfer. The magnetic field is taken to be perpendicular to the direction of gravity and a temperature gradient parallel to the Hartmann walls is assumed in the cavity cross section. A dimensionless volumetric heat production rate $S$ is also assumed in order to accommodate the energy influx due to neutrons entering the blanket from the core of the plasma reactor. The above flow arrangement is used as a first model of heat transfer in conjunction with the first wall effect, in a cavity that is located in the blanket with liquid lithium as the operating fluid. Mass continuity and the momentum, energy and electric charge conservation equations are solved for. The Galerkin finite element method is employed in order to solve the generalized eigenvalue problem, in the parameter space defined by the dimensionless numbers $Gr$, $Ha$ and $Pr$. The Arnoldi method is used for the calculation of eigenvalues with the highest absolute value. GMRES is applied for the solution of the stability problem pertaining to a specific eigenmode. Parametric continuation is performed in order to obtain the evolution of critical modes as wavenumber $k$, $Gr$ and $Ha$ vary. The unstable eigenmodes are not associated with the modes dominating the 2d dynamics of the cavity and arise as a result of a centrifugal instability due to the curvature of the stream lines in the base flow [1]. In all cases examined: i) three dimensional disturbances are less stable than two dimensional ones, ii) the standing wave mode is superseded by the travelling wave mode as the dominant eigenmode and iii) the critical $Ha$ number increases with increasing $Gr$. A parametric study is performed covering different aspect ratios $A$ of the cavity, higher $Ha$ numbers and smaller volumetric rates of heat production $S$ in order to identify the effect of the magnetic field and strength of the emerging recirculation vortices.

In addition, extensive stability analysis calculations are performed in ducts with electrically conducting walls where Rayleigh - Bernard convection takes place in the presence of a horizontal magnetic field, while a temperature gradient that is parallel to the direction of gravity is applied. The above methodology is used, in order to identify the mechanism for generation of quasi two-dimensional structures and recover the experimental findings reported in [2-3] regarding heat transfer rates.

MHD Modelling activities at UPC aimed at Breeding Blanket Analysis

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In the frame of fusion reactor design definition, the detailed analysis of main flow parameters in liquid metal blankets is of utmost interest. Critical aspects are (1) tritium inventories and permeation rates, (2) heat extraction and maximum temperatures for material specifications and (3) MHD pressure drops. The aim of T4F research group at UPC is to develop a CFD code, based on the OpenFOAM toolbox, able to deal with the main phenomena occurring at blanket channels (MHD coupling, heat transfer and tritium transport) and capable to quantify the above mentioned critical aspects. In parallel, CIMNE research group is developing its own finite element MHD code with the aim of algorithm optimization based on the variational multiscale stabilization techniques. Both codes solve the electromagnetic coupling by means of an electric potential based formulation (inductionless MHD approximation). A full validation of the algorithms has been carried out and comparison of results is exposed. The HCLL/ITER blanket has been studied. At T4F a simplified 3D inlet/outlet channel has been simulated with the aim on analyzing the influence of the U-bend (close to the First Wall) on flow stability and its consequences on tritium permeation. Main results are the presence of a complex buoyant flow mainly due to natural convection and the substantial increase of tritium permeation at low Reynolds numbers. The electromagnetic coupling for the complete module (four inlet channels and four outlet channels) has been analyzed at CIMNE. Main conclusions are that the cooling plates make the flow laminar but there exists a non-uniform distribution of the velocity field in the four outlet channels (in the bottom two there is almost no circulation), which enforces the necessity of a different design strategy. Recently, an overview of modeling challenges related with vertical insulated banana-shape liquid metal channels has been discussed at T4F. This new design of blanket is being considered as a progress of conceptual design refinement of dual-coolant liquid metal blankets (DEMO specifications). Among the studied challenges the most relevant are (1) LM/FCI/wall electrical and thermal coupling and (2) LM buoyancy effects. The paper overviews all the ongoing studies and highlights future needs for blanket modeling.

S6-O2

This work is founded by the Spanish National Project on Breeding Blanket Technologies TECNO_FUS through CONSOLIDER-INGENIO 2010 Programme.
Tritium and helium transport phenomena CFD implementation for fusion applications: BelFoam© solver development

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Tritium breeding in future DT fusion reactors is associated to high helium (He) production rates in breeding blankets. Consequences of helium cavitation may be diverse and encompass a wide range of effects, like the trapping of tritium in the bubbles and the severe reduction of permeation efficiency in surface permeators. Helium bubble nucleation, growth and transport models have been implemented [1] to take into account the complex tritium transport phenomena that take place in liquid metal (LM) breeding blanket channels of future fusion reactors.

Transport phenomena including tritium absorption into He bubbles, adsorption onto structural material and desorption to system cooling channels have been modelled and implemented [2]. Tritium absorption has been implemented as both diffusion and recombination limited processes and the effect of absorbed tritium on the bubbles volume is modeled through a combined helium tritium equation of state. Tritium conjugate transport between LM and structural material (SM) has been implemented as an iterative process taking into account possible wall He nucleation in the form of micro-bubbles. Models have been implemented in the OpenFOAM® CFD code as a new solver BelFoam©.

BelFoam© capabilities are presented for different 0D, 1D and 2D test and verification cases together with some LM breeding blanket channel cases at nominal conditions. Sensitivity of the models to some key design parameters is analyzed.


S6-O3

This work is founded by the Spanish National Project on Breeding Blanket Technologies TECNO_FUS through CONSOLIDER-INGENIO 2010 Programme.
LIBRETTO-4: Modelling tritium transport in LiPb/EUROFER capsule materials and understanding of release rates under irradiation

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LIBRETTO (LIquid BReede Experiment with Tritium Transport Option), experimental series are designed and accomplished in order to study the effect of neutrons on the tritium permeation through pre oxidized ferritic-martensitic steel EUROFER tube filled with lead lithium eutectic. For LIBRETTO-4 the two experiments (-4/1, -4/2) were identical but operated at two different temperatures (300-350, 500-550 ºC) providing in-pile continuous and comparable data. The plenum of the liquid metal tube is also continuously swept and measured and from these two measurements an in-pile tritium permeation percentage can be obtained and free surface desorption directly measured. The development and validation of tritium transport modelling tools represents a key issue to justify a tritium safe control and management in order to exploit tritium breeding tests at Test Blanket Modules in ITER and to certify by simulated extrapolation tritium self-sufficiency at DEMO breeding blanket scale. The present work focuses on the numeric modelling approaches to reproduce tritium release-rate data, both permeation and desorption at LIBRETTO-4 capsules. By model is understood compartmental tritium transfers between real enclosures and first-order release rates through irradiated materials. Different numeric strategies have being investigated starting from: (1) a simple home-made model in Mathcad13.0 implementing transfers, followed by: (2) a 1-d finite difference model in TMAP7 (today the unique tool with ITER QA pedigree for tritium transport assessments) implementing the complete physics of a tritium release-rate transfer model and finally in: (3) the flexible, modular and versatile numerical tool EcosimPro®. EcosimPro® being developed for tritium transfer modelling at ITER Tritium Plant scale has robustly shown its strong performances for tritium release-rate transfer modelling. On the base of the quality of fitting/understanding obtained from the diverse numeric approaches a clear release-rate physical representation of tritium transfers is obtained and key tritium transport data under irradiation is validated as: (1) diffusion coefficients in lead-lithium eutectic and in EUROFER; (2) Sieverts´ constants, (3) tritium mass transfers across a lead-lithium free swept surface, (4) finite mass exchange coefficients at interfaces and (5) release-rate constant at surfaces. Transport values obtained are reported and discussed.

S6-O4

Modelling work undertaken under the frame of F4E GRT08-2009 and supported by CDTI.
Development of PFD simulation tools for transient tritium transfers and system inventories modeling of HCLL TBMs and DEMO auxiliaries

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The demonstration of tritium self-sufficiency of future DEMO reactors is a major milestone for fusion technology to be tested in ITER Test Blanket Module (TBM) programmes. TBM tritium testing exploitation in ITER needs of validated Process Flow Diagrams (PFD) numerical simulation tools. PFD simulation tools should be as far as possible developed and validated as “predictive”. “Predictive” goal represents a very ambitious goal of tritium fusion technology. A blind testing of the PFD modelling tools prediction capabilities in ITER represents a major test of prediction qualities and of the reliable understanding of tritium transport phenomena under a realistic blanket environment. Furthermore it means the certitude of extrapolation capabilities to DEMO breeding blanket design. The development of similar tools at both, TBM and DEMO operational ranges appear as a reasonable R&D strategy.

PFDs balancing dynamically transferred T atoms below 0.1\% at both, TBM and DEMO scale levels have been developed in CIEMAT and IQS based on TMAP7 (as the unique tritium transport modelling tool having ITER QA pedigree) and commercial Aspen+® software largely used in chemical engineering, respectively. Current PFDs models includes refined coupling of primary coolant chemistry control and EUROFER permeation properties through the surface state of oxidation based on Ellingham’s diagrams.

Aspen+® library of modules are generated as a new set of operating blocks. The modular nature of PFD tool with both TMAP7 and Aspen+® solutions permits system scale sizing analyses of reference operational ranges. TMAP7 and Aspen+® solutions are compared for PFD modelling tool bases at both, ITER-TBM and DEMO scales for HCLL, designs and reference runs.

The scope of the results and the projection of the model capabilities for the design of systems are discussed.

S6-O5

This work is founded by the Spanish National Fusion Programme through IQS/Ciemat contract agreement 08/210.
Magnetohydrodynamic flow in the helium-cooled lead-lithium test blanket for ITER. Mock-up experiments, comparison with numerical simulations and further requirements on the way to ITER

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One blanket concept that will be tested in ITER is the helium-cooled lead lithium (HCLL) blanket. For ITER experiments a detailed insight into the physical phenomena related to the magnetohydrodynamic (MHD) interaction of the electrically conducting breeder material with the plasma-confining magnetic field is mandatory.

Liquid-metal MHD flows in a scaled mock-up of a HCLL test blanket module for ITER have been investigated experimentally and numerically. The used geometry is scaled down by a factor 2 compared to the original dimensions to fit into the gap of the large dipole magnet available in the MEKKA laboratory of the Karlsruhe Institute of Technology. Pressure and electric potential on the surface of the mock-up have been recorded for various magnetic field strengths and flow rates. The experiments confirm theoretical predictions according to which the major contributions to the total pressure drop arise in access pipes and manifolds. Smaller contributions occur when the flow passes through narrow gaps in the back plate or at the first wall of the TBM. Results of electric potential measurements on the surface of the test section yield information about flow distribution in the mock-up. A strong electrical coupling of the various flow domains results in rather uniform flow partitioning in the breeder units in accordance with numerical simulations.

The presented results give good insight into blanket-typical MHD flows in case of uniform applied magnetic fields. Future investigations will focus on MHD flows in non-uniform magnetic fields for studying the effect of the radial variation of the field. Other problems to be considered in future are for instance the MHD effect of anisotropic conductance of walls due to helium channels, functionality of sensors to be used in ITER, TBM draining time during emergency conditions etc. In parallel some research should be dedicated also to fundamental aspects of MHD flows since only such investigations will help improving and validating numerical predictive tools.

S6-O6
Integrated Modeling of MHD Flows, Corrosion/Deposition and Tritium Transport in Liquid-Metal Blankets

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We consider mass transfer processes associated with liquid-metal (LM) flows in fusion reactor blankets, such as corrosion at the material interface, transport of corrosion products and their deposition, and tritium transport. We describe the ongoing work and first validations of new computational tools capable of addressing these phenomena in any type of LM blankets but special emphasis is given to the Dual-Coolant Lead-Lithium (DCLL) blanket [1], which is considered in the US for testing in ITER and for using in the DEMO reactor. In this blanket, eutectic alloy Lead-Lithium (PbLi) circulates slowly as coolant and breeder. The associated mass transfer processes occur in the presence of a strong plasma-confining magnetic field and high temperature gradients, resulting in tight coupling between heat and mass transfer and magnetohydrodynamic (MHD) phenomena. The current activities on development of integrated computational tools include modernization of the original 3D parallel MHD software HIMAG [2] to couple it with new heat and mass transfer solvers. In parallel, a number of research codes (mostly based on quasi-2D approaches) are developed to test various physical models and boundary conditions. Development of new and modification of existing physical/mathematical models, boundary conditions and closure relations (hereinafter “phenomenological modeling”) and integration them in the computational suite is the key element of the ongoing R&D. The new codes have been benchmarked against available experimental data for corrosion of ferritic samples in PbLi with [3] and without [4] a magnetic field. The examples of application of the newly developed software include tritium transport and corrosion of ferritic walls in the poloidal ducts of the DCLL DEMO blanket with the flow channel insert, gravitational mixed convection in a transverse magnetic field caused by volumetric heating by neutrons, and extraction of the corroded material in a magnetic trap. The needs for further R&D in the MHD, Heat&Mass Transfer area and possible collaboration efforts among the IEA parties are also discussed.

S6-O7
Monte Carlo method is a preferred technique for fusion neutronics. MCNP [1] is the current standard tool, well established and validated for fusion technology applications. CEA Monte Carlo code TRIPOLI-4®1 [2], developed and applied mainly for fission nuclear reactor technology and shielding applications, needs to be verified & validated for fusion applications.

This work, performed in the framework of the EUROBREED programme supported by the EFDA, is on the verification issue on the example of a computational benchmark for a HCLL DEMO.

The benchmark compares the TRIPOLI and MCNP results in terms of Tritium Breeding Ratio and local surface neutron flux on the First Walls.

The TRIPOLI model is issued from the KIT MCNP model [3] and partly converted using the MCAM software [4].

The results are:

• For TBR: $1.0886 \pm 0.0001$ with a TRIPOLI partly-homogeneous breeding blanket geometry, to compare with the KIT MCNP heterogeneous model at $1.088 \pm 0.0004$ [2]

• For the local surface neutron flux arriving on the First Walls of HCLL DEMO:

  An average error relative to the MCNP KIT reference source of 0.16% with a local maximum at 0.9%.

Thus TRIPOLI-4.6 can be considered as adequate as MCNP for neutron transport and TBR calculations relative to the HCLL DEMO reactor.

Session 7:

PRESENT AND FUTURE TEST PROGRAMMES & FACILITIES

Test programs on the LiPb breeder blankets development: ITER TBM and role of IFMIF
Tritium breeding blanket testing is a critical element in the ITER mission. Mock-ups of DEMO Breeding Blankets (BB), called Test Blanket Modules (TBMs), inserted and tested in ITER in dedicated equatorial ports directly facing the plasma and associated with their own appropriate cooling and tritium extraction systems, are the principal means by which ITER will provide the first experimental answers on the correct performance of the tritium breeding blankets, a still open issue on the path to fusion power.

ITER conditions are different from the expected DEMO conditions. The most important differences are that ITER features a lower neutron load on the first wall (FW) (~30% of the DEMO values) and much lower neutron fluence on the First Wall. Moreover, it features operations with relatively short plasma pulses compared to the quasi-continuous or steady state operation expected in DEMO. Despite these differences, several studies performed by Europe and other Parties have shown that most of the required data can be obtained by the testing of TBMs in ITER, provided they use the same materials as in the DEMO Breeder Blanket. For instance, neutronics, thermal-mechanics and tritium breeding/extraction performances (as well as some of possible coupling phenomena) can be checked during this testing bringing essential data for the design of future Breeder Blankets.

The Helium-Cooled Lithium-Lead (HCLL) TBM shall be installed in the equatorial port 16 of ITER directly facing the plasma. The HCLL TBM testing strategy is to test different versions of the TBM adapted to perform specific experiments in the different fields such as neutronics, thermo-mechanics, magneto-hydrodynamics (MHD) and electromagnetic (EM), tritium control and management. Furthermore, these tests have been adapted to the operational plan of ITER that foresees different plasma phases with very different operating conditions, from the initial H/He phase (without neutrons) to a high-duty D-T phase after several years of operations, passing through the D phase and the low-duty D-T phase.

The presentation will summarize the present status of the HCLL TBMs test program in ITER, illustrating in various fields how it is planned to maximize test objectives through a series of dedicated experiments. Remaining issues to be addressed through future R&D for maximizing the outcome of the HCLL TBM program will be discussed.
Status Report on the US DCLL ITER TBM


Under the US Fusion Nuclear Science and Technology Development program, we have selected the Dual Coolant Lead Lithium concept (DCLL) as our primary Test Blanket Module (TBM) for testing in ITER. The US serves, with support from Korea, as the Interface Coordinator for port No. 18. The DCLL blanket concept has the potential to be a high-performance DEMO blanket design with a projected thermal efficiency of >40%. Reduced activation ferritic/martensitic (RAF/M) steel is used as the structural material. Helium is used to cool the first wall and blanket structure, and the self-cooled breeder PbLi is circulated for power and tritium extraction. A SiC-based flow channel insert is used in the large poloidal channels as a means for magnetohydrodynamic pressure drop reduction from the circulating liquid PbLi and as a thermal insulator to separate the high-temperature PbLi (~650°C to 700°C) from the RAF/M structure. The RAF/M material must operate at temperatures above 350°C but less than 550°C. We have been developing the mechanical design, and performing neutronics, structural and thermal hydraulics analyses of the DCLL TBM module. Progress on related R&D needs to address critical issues will be reported separately at this meeting. To prepare for the testing in ITER, we have provided input to the ITER TBM Report Preliminary on Safety (RPrS) report. This paper will be a summary report on the progress and results of recent work, including the revised helium loop arrangement and different areas of analysis in support of the DCLL TBM development.

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RF Experimental Facilities for Thermal Hydraulic Tests of LLCB TBM Mock-ups

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Since 2009 RF has been developing lead-lithium ceramic breeder (LLCB) test blanket module (TBM) proposed by India TBM Team. In this concept LL and Li-containing ceramics are breeder materials, He is the coolant of the first wall and LL is the breeder zone coolant. Thermal hydraulic tests of LLCB TBM elements and TBM mock-ups are necessary to validate MHD/HT codes and check the design prior to manufacturing of TBM for in- ITER tests.

First stage of the mock-up tests will be performed on NaK in magnetic field ~1–1.5 T for the following TBM elements: poloidal ducts in uniform and inclined magnetic field; collector between round pipe and rectangular duct; round duct in non uniform magnetic field of large extension. Uniform magnetic field region of the existing magnet has dimensions 600(pol)x120(rad)x170(tor) mm. NaK is more convenient for experiments because of its low temperature (50–100 0C) and existing technique for velocity measurement. Also one can reach higher Hartmann numbers on NaK in comparison with LL with the same magnetic field induction. Mock-ups conceptual design for thermal hydraulic tests is presented. Electric heater simulates heat flow from ceramics and allows preliminary estimation of thermal gravitation flow, to be followed by tests in LL.

LL loop with low-activation ferritic steel as structural material for the next stage of the tests is under design. Superconducting magnet with induction 3-4 T in volume 800x500x120 mm is foreseen for this loop and its conceptual design is made.

S7-O1
Role of IFMIF on the development of the liquid breeder blankets

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The International Fusion Materials Irradiation Facility (IFMIF) aims to test and qualify advanced materials at relevant fusion irradiation conditions. The Engineering Validation and Engineering Design Activities (EVEDA) are being currently developed in the framework of the Broader Approach Agreement. During this phase, the main irradiation modules are being designed, focusing on the current candidate materials for different concepts of fusion reactors, such as structural materials for blankets and first wall, functional materials for ceramic, liquid breeders and diagnostics, and other.

In this paper, a set up for irradiation experiments on functional materials for liquid breeders (Liquid Breeder Validation Module, LBVM) is proposed. The irradiation conditions of the medium flux region of IFMIF in terms of dpa, gas production, etc. are analyzed and compared to the DEMO fusion reactor, to demonstrate the suitability of IFMIF to perform such experiments and to find out the best configuration for them.

The conceptual design of the module and some of the foreseen experiments in materials such as antipermeation, anticorrosion and electrical insulating coatings are also presented and described as well as their main requirements and features.