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Overview of the HCPB Research Activities in EUROfusion

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Overview of the HCPB Research Activities in EUROfusion

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***Abstract*—In the framework of the EUROfusion's Power Plant Physics and Technology, the Working Package Breeding Blanket aims at investigating 4 different Breeding Blanket (BB) concepts for a EU Demonstration Fusion Reactor (DEMO). One of these concepts is the Helium Cooled Pebble Bed (HCPB) BB, which is based on the use of pebble beds of lithiated ternary compounds and Be or beryllides as tritium breeder and multiplier materials respectively, EUROFER97 as structural steel and He as coolant. This paper aims at giving an overview of the EU HCPB BB R&D being developed at KIT, in collaboration with Wigner-RCP, BUTE-INT and CIEMAT. The paper gives an outline of the HCPB BB design evolution, state-of-the-art, basic functionalities, requirements and performances, and the associated R&D activities in the areas of design, functional materials, manufacturing and testing. Additionally, attention is given also to the activities dedicated to the development of heat transfer augmentation techniques for the FW and the corresponding testing. Due to their nature as design drivers, a brief overview in the R&D of key HCPB interfacing areas is given as well, namely the Tritium Extraction and Recovery system, the Primary Heat Transfer & Power Conversion Systems and Safety topics, as well as some specific activities regarding the integration of in-vessel systems through the BB. As concluding remarks, an outline of the standing challenges and future R&D plans are summarized.**

***Index Terms*—HCPB, tritium breeding, DEMO, EUROfusion**

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I. Introduction

THE EUROfusion Consortium is the entity responsible to coordinate the 30 European fusion laboratories and their Linked Third Parties for the realization of the technical roadmap towards fusion electricity [1] through a Demonstration Fusion Power Plant (DEMO) by mid of this century. In the frame of the EUROfusion's Power Plant Physics and Technology (PPP&T) department, several Working Packages (WP) are defined to address the necessary R&D for the realization of that roadmap. In particular, the WP-Breeding Blanket (WPBB) is dedicated to the development of 4 Breeding Blanket (BB) concepts in 4 sub-WP, namely [2] the Helium Cooled Pebble Bed (HCPB, WP1), the Helium Cooled Lithium Lead (HCLL, WP2), the Water Cooled Lithium Lead (WCLL, WP3) and the Dual-Coolant Lithium Lead (DCLL, WP4).

The WP1-HCPB is itself subdivided in 3 research areas: (1) the definition and description of the HCPB BB system, i.e. rationale for the selection of the architecture in the tokamak, safety and top-level requirements, interface definition, materials, coolant parameters and tritium extraction scheme; (2) HCPB BB Design & Analyses towards the substantiation of a conceptual design and (3) R&D in breeder materials (Li-ceramic tritium breeders and Be-based neutron multipliers).

This paper aims at giving a broad overview of the research activities in the WP1-HCPB, with an additional insight into R&D of key HCPB interfacing systems, which are often additional strong design drivers of the HCPB BB system.

II. HCPB Breeding Blanket Description

A. Functions, interfaces and the need for a holistic design

The BB represents the core of a tokamak-class fusion power plant. This system have the following top-level functions:

1) Tritium breeding: The BB shall regenerate the tritium consumed by the nuclear fusion reactions, ensuring the tritium self-sufficiency of the reactor. The tritium produced shall be able to be extracted and recovered by a dedicated Tritium Extraction and Removal (TER) system.
2) High grade heat extraction: the BB shall be able to transfer the nuclear power generated in the plasma to a cooling medium and transport it to a Primary Heat Transfer System (PHTS) with appropriate coolant parameters in order to be used by a Power Conversion System (PCS), which will produce a net electric output.
3) Shielding: The BB shall be able to shield from thermal and nuclear radiation the parts behind it, mainly the vacuum vessel (VV) and the superconducting magnets.

Due to this variety of functions and its particular position in the tokamak covering the most of the in-vessel surface (80%÷90%), the BB have a high degree of interactivity with many other systems. This is made apparent in the block diagram of a tokamak with a HCPB BB of Fig. 1. The following are key interfacing systems with the HCPB BB system (in parenthesis, section of the paper with the overview of the related R&D): the VV through the BB attachment system (section VIII-A), the Tritium Extraction and Removal (TER) system (section VII-A), the Balance of Plant (BoP) system (section VII-B), the VV Pressure Suppression System (VVPSS) (section VII-C) and in-vessel systems, like the Neutron Beam Injector (NBI)/Electron Cyclotron Heating (ECH), the fueling lines (section VIII-B) and control systems & diagnostics, among others. On top of them, safety aspects are to be taken into account (section VII-C).

All the interfacing systems above pose complex requirements to the design space of the BB, usually with counter-acting effects. Therefore, the conceptual design and related R&D activities of the HCPB cannot be conducted behind these systems and a holistic development of the HCPB must be conducted in close collaboration with all these interfacing systems, as many of them have been already identified as strong potential design drivers.

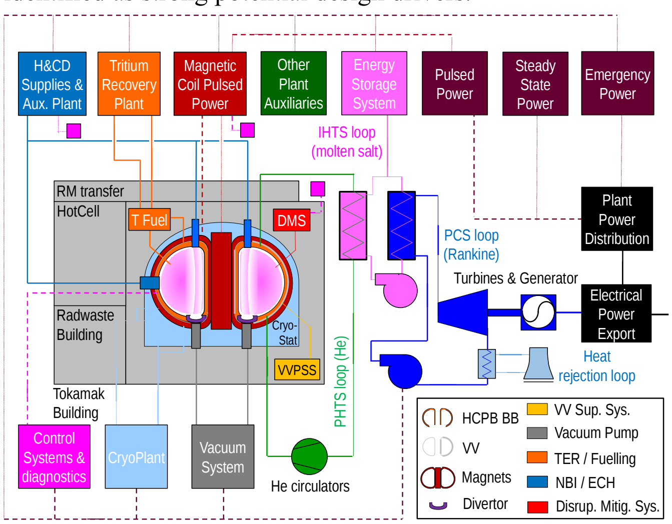

Fig. 1. Tokamak block diagram of the major systems (modified from [3])

B. HCPB Design Description

1) Concept design evolution

The ITER HCPB TBM is based on the so-called "beer-box" concept designed during the Power Plant Conceptual Studies (PPCS) at the beginning of 2000 [4]. This concept has been continued during the initial EFDA PPP&T conceptual DEMO studies during 2011-2013 (e.g. [7], [8], and [14]), revealing that the current definition of DEMO, which has been downscaled from that in PPCS, poses much stringent constraints in the blanket coverage, nuclear performance requirements, technology readiness, safety requirements and FW loads, rendering the "beer-concept" as inappropriate.

Hence, during the initial stage of the EUROfusion's DEMO pre-conceptual phase studies in 2015 a major design revision has been conducted to simplify the concept and to boost its performance figures, leading to a new so-called "sandwich-concept" [6] that can potentially fulfill all BB requirements. This concept is the current architecture of the HCPB.

2) Current architecture

The current architecture of the HCPB BB is adapted to the latest EU DEMO1 baseline BL2015 (R_0 = 9.1 m, a = 2.9 m, A = 3.1, burn-time 2hr, dwell-time 0.5 hr, P_{fusion} = 2037 MW), composed by 18 sectors with 2 IB and 3 OB segments per sector. The segments follow a so-called multi-module modularization, where the IB and OB segments are formed by 7 BB modules. Fig. 2-a) depicts 1 HCPB sector.

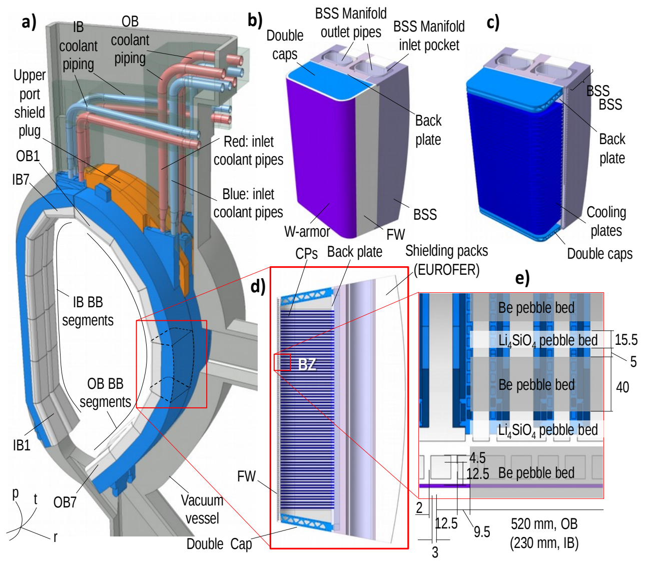

Fig. 2. HCPB BL2015: a) HCPB DEMO1 sector; b) and c) 3D views of the OB4 module; d) section cut of the OB4 module; e) detail of the BZ. The coordinate axis is toroidal: p = poloidal; t = toroidal; r = radial.

Each module is a box with boundaries formed by an actively cooled U-shaped First Wall (FW), 2 so-called double caps and the Back Supporting Structure (BSS). The breeder zone (BZ) is the volume delimited by the FW, the double caps and the backplate, Fig. 2-d), and it is formed by a sandwich-like structure of radial-toroidally actively-cooled cooling plates (CPs), which forms the volumes of alternating ceramic breeder (CB) and neutron multiplier material (NMM) polydispersed pebble beds, as shown in Fig. 2-e).

3) Materials

The reference CB material is Li_4SiO_4 in form of pebbles with sizes between Ø0.25 mm ÷ Ø0.63 mm following a lognormal distribution with mean ~Ø0.4 mm. The reference NMM is Be also in form of pebbles of ~Ø1 mm. Both functional materials are the same ones that are selected for the HCPB TBM in ITER. However, in current R&D an advanced CB mixture of Li_4SiO_4 and Li_2TiO_3, which significantly improves the crush load of the pebbles, is being developed by KIT in collaboration with CIEMAT. Similarly, R&D is also being performed in KIT to develop advanced Be-Ti with higher resistance to oxidation and steam reaction. More details about these R&D activities are given later in section IV-A and IV-B.

EUROFER97 is the reference Reduced Activation Ferritic-

Martensitic (RAFM) structural steel for the BBs in DEMO (as well as for the ITER EU TBMs). However, KIT continues the R&D in advanced RAFM steels in the frame of the EUROfusion WP-Materials and indeed a promising variant of EUROFER97 has been recently developed by only applying a thermomechanical treatment, leading to an improvement of the creep strength of standard EUROFER97 [9].

Regarding the operational temperatures, on one side the Li_4SiO_4 pebbles require temperatures higher than ~600 °C to have high tritium release rates but lower than 920 °C to avoid microstructure changes due to thermal creep effects. On the other side, the temperature in Be should be high enough to maximize the tritium release, but should be kept under 650 °C to avoid loss of mechanical properties and excessive swelling [17]. The safe operational window for the EUROFER97 steel is between 350 °C to 550 °C [24][25], where the lower limit is imposed by the DBTT shift under irradiation and the upper limit is driven by the loss of creep strength. Regarding the upper temperature bound, advances in the EUROFER97 as the one mentioned previously could push this limit to ~650 °C.

4) Tritium extraction

In order to extract the tritium produced in the CB (and partly in the Be), a low pressure purge gas (0.2 MPa) of He (carrier gas) with an addition of 0.1% wt H_2 (doping agent) sweeps the pebble beds independently. The doping agent is of especial importance, as it is the promoter of the isotopic exchange to form HT, which will be carried out of the BB towards the TER system. The purge gas chemistry is a current area of research and other additives, such as H_2O is being considered due to its potential to reduce to negligible values the tritium permeation into the coolant, as the isotopic exchange of tritium with H_2O forms non-permeating tritiated water species. Other details of the R&D on the HCPB TER system are given in section VII-A.

5) Coolant

He gas at 8 MPa is chosen as the coolant. The choice for He is given by its chemical inertness and transparency to neutrons, as well as to its single phase in its operation window for the HCPB. The inlet temperature is set to 300 °C, which, given the ΔT in the steel thickness, it is necessary at least to keep an average temperature in the steel >350 °C as shown in section II-B-3. The outlet temperature is set to 500 °C to avoid potential temperature peaks larger than 550 °C (section II-B-3.

While this operational window can be seen along the history of past designs of the HCPB, research is being conducted to check the suitability to increase the outlet temperature while satisfying the stress criteria given in the codes and standards (C&S) with the current EUROFER97. Recent R&D in advanced EUROFER97 batches (see II-B-3) may allow outlet temperatures of 600 °C÷650 C° and therefore turbine inlet temperatures of >500 °C, more according to current Advanced Gas Reactors (AGR) or High Temperature Reactors (HTR). Additional gains in the high outlet temperature will have a positive impact in the pressure drops, size of the heat exchangers and the plant efficiency.

The coolant pressure setting is a result of a trade-off between the BB pressure drop and the BB P-stresses and it has a decisive influence in the pressure evolution of the BB segments (in an in-box LOCA), the VV and the expansion volume (during an in-vessel LOCA), the tokamak building (in an ex-vessel LOCA) and, to some extent, in the chronic leakages. The continuous improvements in the BB pressure drops in the recent years has motivated an investigation on the possibility to reduce the coolant pressure to HTRs conditions (6÷7 MPa).

III. HCPB Development: Design and Analysis

The cornerstone of the HCPB development lays on the design and analysis. These activities are being performed at KIT, in collaboration with BUTE-INT and the Wigner RCP. Basically 3 types of analyses are performed, neutronic, thermohydraulic (TH) and thermomechanic (TM), which are often intercalated by some CAD design work. These analyses are normally done sequentially, conforming a design cycle, as described in III-A. However, an integrated multiphysics approach is highly desirable, as it would allow the automatisation of the design cycle by some optimization algorithm. A research in this multiphysics coupling is being progressed [15].

A. Design cycle

The flow chart of the HCPB design cycle is shown in Fig. 3. The design cycle starts by assuming a preliminary conceptual design of the HCPB. From the beginning it has been identified that the so-called in-box LOCA event is a main design driver, being therefore the first step of the design cycle. At this point a global, detailed thermal field is not yet available, therefore the assessment normally involves only the evaluation of primary stresses P at some conservative temperature levels (e.g. 500 °C). In a later stage, the secondary stresses Q are to be included in the assessment, as required in the RCC-MRx code. Once the structure of the BB satisfies the defined stress limits for the damage modes involving P stresses, the cycle goes through an analysis campaign comprising neutronics, TH and TM analyses. If the acceptance criteria after each of these analyses are not met, a design modification is performed.

The complexity of this sequential design cycle resides in that the decision on a new design have to take into account many different (often counteracting) engineering judgments on the following fields: (1) manufacturing and assembly feasibility (e.g. realistic component fabrication and welding processes), (2) safety (e.g. Be inventory minimization), (3) BB reliability and availability (e.g. minimization of number of welds), (4) neutronics (e.g. steel volume fraction minimization), (5) TH (e.g. minimization of flow speeds to minimize pressure drops), (5) economics (e.g. ^6Li enrichment minimization). Additionally, there is the risk that a design modification will significantly alter the outcome of former analyses. Therefore, an engineering judgment on the impact after a design modification is conducted and a decision on where to resume the design cycle is taken. A brief description of these analyses and main results are reported in sections III-

B to III-D.

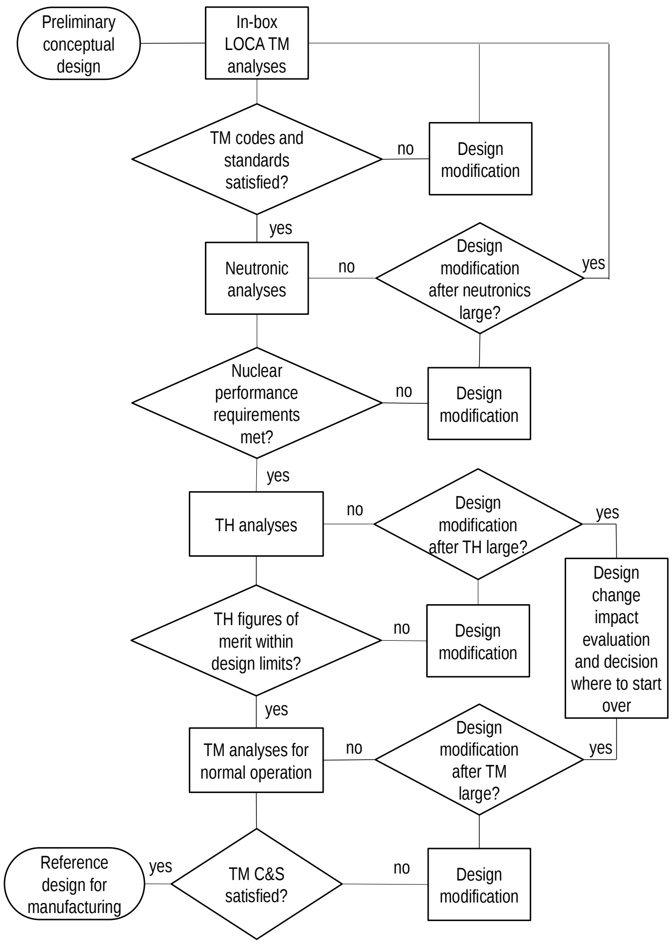

Fig. 3. Design and analysis cycle of the HCPB.

B. Neutron transport analyses

These basic analyses are a key input for the TH analyses. Neutronic analyses are usually performed on half of a tokamak sector by means of the MCNP5 [26] code with nuclear data from FENDL-2.1 library.

For the current sandwich-concept of the HCPB [6], two variants are envisaged (Fig. 4): (1) HCPB with large BZ ("HCPB-long") aimed at maximizing TBR and (2) HCPB with small BZ ("HCPB-short") aimed at reducing the inventory of functional materials (especially Be), as well as offering the possibility to reduce the thickness of the segments (especially at the OB), with a possible positive impact in plasma stability.

The most recent neutronic campaign on the HCPB is reported in [16]. The TBR values for the HCPB-long and HCPB-short are ~1.26 and ~1.15, which satisfies the current requirements. At the moment there are still large uncertainties in the architecture of DEMO and some of its very basic parameters, like the aspect ratio or fusion power, are not yet fixed. Moreover, the number of in-vessel systems and their size, materials and configuration is not well understood and even the discussion on the divertor configuration (single or double) is still open. Therefore, the current TBR margin seems large enough cover these possible unfavorable cases, which is a very desirable characteristic. Other nuclear performance figures such as the neutron shielding, dpa damage in the VV and nuclear heating in superconducting coils fulfills the requirements [16].

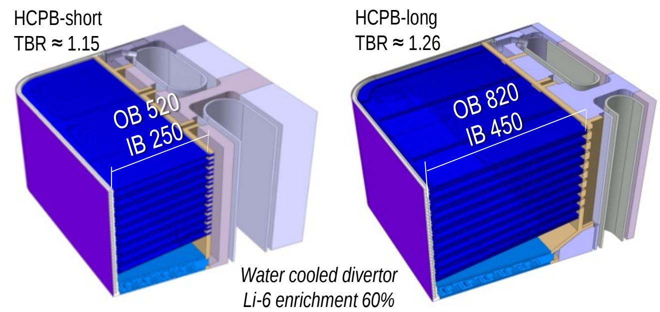

Fig. 4. Current HCPB designs pursued for the EU-DEMO. Left: HCPB-short (reference design). Right: HCPB-long (alternative design)

In addition to the calculation of blanket performance figures, some research has been performed in order to broaden the palette of possible CB and NMM materials. Some results are anticipated in [16], but a dedicated paper is planned to report the details and results of this investigation.

C. Thermohydraulic analyses

The TH analyses aim at obtaining thermal fields of the BB system. This thermal field is used as input for further TM analyses, but also to evaluate the correct operational temperature windows in the pebble beds.

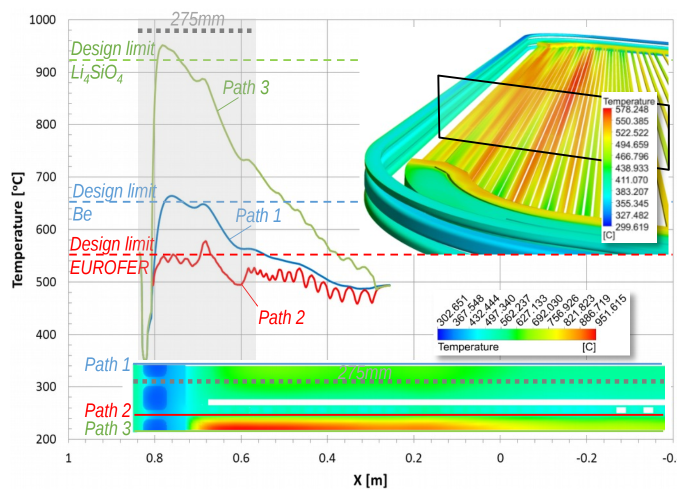

Fig. 5. TH CFD analysis on the reference HCPB design; on a radial-poloidal mid-plane of the equatorial OB module.

Fig. 5 shows a radial-poloidal contour plot of the temperature at normal operation during the burn-time, obtained with CFD at the mid-plane of the equatorial OB module, assuming a FW heat flux of 0.5 WM/m². It can be observed that globally the HCPB temperatures are kept within the defined temperature windows, with the exception of very local hot spots of less than 5% from the design limits. More details and other related analyses can be found in [6], [11] and [12].

D. Thermo-mechanical analyses

Normal and off-normal scenarios are considered in this type of analyses. In both cases, a stress linearization is conducted and then an assessment with respect RCC-MRx is performed.

1) Off-normal condition: In-box LOCA

As stated in III-A, the analysis of the in-box LOCA of the segments strongly drives the design of the BB from an initial stage. The elimination of the vertical stiffening grids with respect to the "beer-box" concept results in an unacceptable stress-state at ITER-like caps. Hence, these caps have been redesigned as a "double-cap" with a Warren truss-like in-between [5], Fig. 6, being able to withstand a pressure of 9 MPa (8 MPa coolant pressure +10% uncertainties).

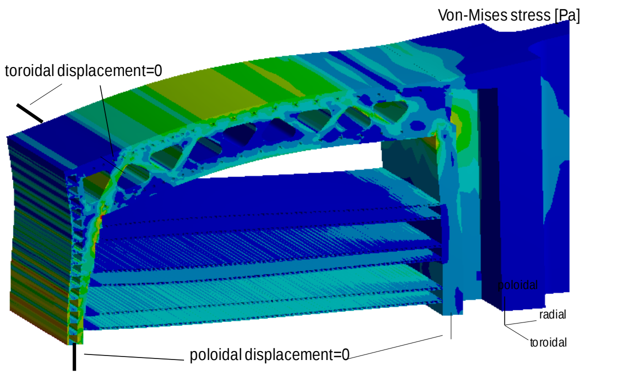

Fig. 6. In-box LOCA in the top-mid-plane region of the equatorial OB module

The safety categorization of such an event is still a topic of current discussion in the TBM Consortium. For DEMO, the current consensus is not to give any safety function to the in-vessel systems and therefore, the level to be considered is D. However, the authors consider that the safety classification is a decision of the Safety Body in a last instance. Therefore and as a conservative assumption, this accident is viewed as Cat III and assessed at a Level C after RCC-MRx for the HCPB, until more consolidated will be known, especially from ITER.

2) Off-normal condition: disruption events

Currently only the electromagnetic forces coming from a central disruption event [27] have been considered for the TM assessment of the segments under this off-normal condition. Some details of these studies are given in section VIII-B.

3) Normal operation conditions

Once a thermal map in the BB is obtained, a full assessment of the stress-state under normal operation is required and to be assessed with the selected C&S, i.e. RCC-MRx [10], as it is done for the HCPB-TBM in ITER.

Due to the resource-intensive nature of these analyses and the need to perform often as many iterations as needed until obtaining a design that fulfills the stress criteria in the C&S, a full-scale model of the segments is normally not computationally and timely feasible and detailed assessment is done in a sub-region with appropriate, often conservative assumptions. Fig. 7 shows as an example a "unit BZ slice" TM model (von Mises plot) of the mid-plane of the equatorial OB4 module, comprising 2 FW channels, 1 CP and half of the Be and CB volumes, applying *P+Q* loads during the burn-time of a pulse. The assessment includes monotonic and fatigue analyses. More detailed results can be found in [6] and [13].

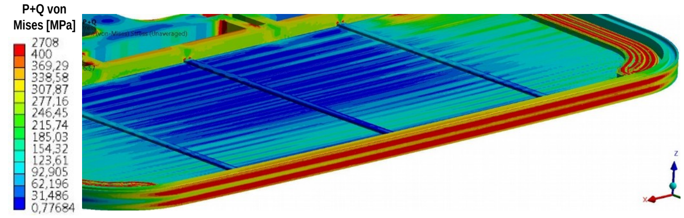

Fig. 7. TM analysis of a unit BZ slice of the HCPB OB4: von Mises plot after applying P+Q loads in the burn-time phase of a DEMO1 pulse.

E. Pebble-beds thermomechanical modelling

Due to their discrete nature and significantly different material properties with respect to the steel volumes where they are enclosed, the use of pebble beds as functional material in solid BB as the HCPB poses a question about the thermal control of the BZ materials. Therefore, a continuous R&D in the area of the thermomechanical modeling of the pebble beds has been carried out at KIT, in close collaboration with the University of Sydney (Australia) and the IITM (India).

Historically, 2 approaches have been pursued: (1) phenomenological approach, based on the use of continuum elastoplastic models executed in Finite Element (FE) codes [18][21] and the Discrete Element Modeling (DEM) [20], based on the direct modeling of assemblies of single pebbles. Due to the steady increase in computational resources, the DEM has gained importance, as it is able to relate macroscopic behavior with the micro response at the pebble scale. As an example, Fig. 8 shows the DEM investigation on the influence of the particle's shape on the pebble bed's mechanical behavior. This research has been performed with assemblies of ellipsoidal particles (Fig. 8-left), in which each particle is approximated as the union of 3 collinear spheres of equal radius. Recent investigations have shown that the DEM models are able to capture behavior such as the reduction of the bed axial stress (Fig. 8-right) at a given strain level [23], the reduction of the pebble bed effective thermal conductivity due to the Smoluchowski effect and the effects of the cyclic loading on the mechanical behavior of the pebble beds [22].

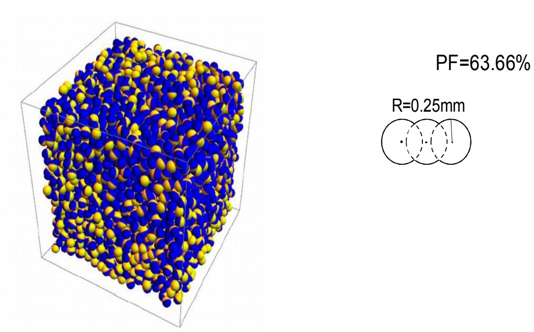

Fig. 8. R&D on pebble beds thermomechanics. Left: random packing of non-spherical pebbles. Right: uniaxial compression of non-spherical pebbles [23].

Although not yet formally implemented in the EUROfusion activities, this area is thought to be a key R&D part of the WP1 activities in the near future. More details on these investigations can be found in [18], [19], [20], [22] and [23].

IV. R&D IN FUNCTIONAL MATERIALS

A. Ceramic breeder materials

The reference CB compound for the ITER HCPB-TBM is Li_4SiO_4, produced by a melt-spray process. While this material is still considered as a reference for DEMO, its rupture strength is lower than that in Li_2TiO_3, which is another well-known option as CB. However, the Li density in Li_2TiO_3 is lower than that of Li_4SiO_4, which results in lower tritium breeding performances than Li_4SiO_4. In order to increase mechanical properties of Li_4SiO_4, additions of Li_2TiO_3 are introduced, resulting in the so-called "advanced CB".

The R&D in this area is led by KIT in collaboration with CIEMAT and it is focused on: (1) production feasibility and potential industrial-scale up, (2) evaluation of optimum compositions, (3) reprocessing studies and activation, (4) long term stability, (5) thermal conductivity, (6) pebble bed thermomechanical modelling, (7) tritium loading and release, (8) behavior under electron and gamma irradiation and (9) compatibility with EUROFER97.

1) Production of advanced ceramic breeder materials

The demonstration of the production process of advanced CB is performed at the Karlsruhe Lithium Orthosilicate (KALOS) facility [28][29] (Fig. 9). The production is based on a melt processing at 1350÷1400 °C and droplet generation by natural jet decay. The generated droplets are solidified into pebbles by applying a liquid nitrogen spray. The jet formation dynamics are monitored and characterized using a high-speed camera in conjunction with dedicated image processing algorithms.

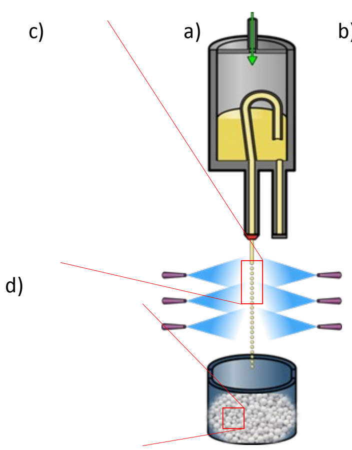

Fig. 9. Production of advanced CB in KALOS: (a) schematic view of the process, (b) picture of KALOS in operation, (c) capture of the melt jet decay using a high speed camera, (d) optical micrograph of the produced pebbles.

Li_2TiO_3 additions in KALOS are possible up to 40 mol%. An eutectic has been identified at ~25 mol% of Li_2TiO_3. At higher Li_2TiO_3 contents, it is observed (Fig. 10) that the microstructure of the CB is increasingly dominated by the primary Li_2TiO_3 and therefore gains in the rupture strength are expected. Indeed, while neutronic studies have shown that the TBR reduction due to the use of advanced CB with additions up to 40 mol% Li_2TiO_3 is of only ~1% with respect to using 100% Li_4SiO_4 (Fig. 11) the increase in crush load is >400%.

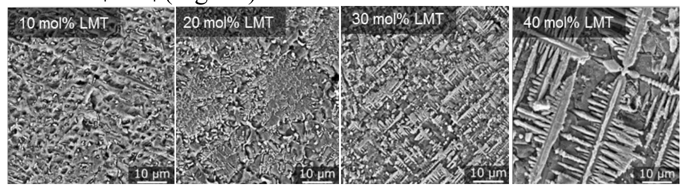

Fig. 10. Microstructure of advanced CB at increasing Li_2TiO_3 content

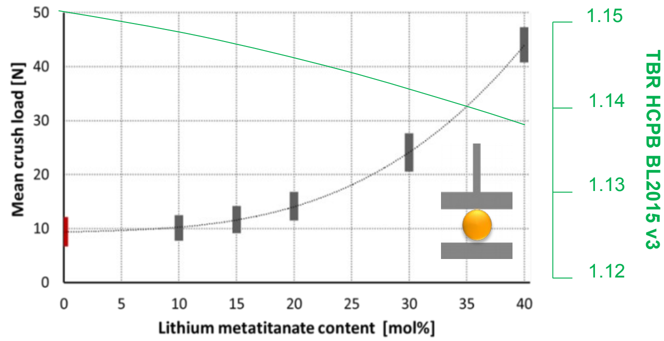

Fig. 11. Crush load of advanced CB with increasing Li_2TiO_3 content

2) Characterization of advanced ceramic breeder materials

A key property for the TH analyses is the effective thermal conductivity of pebble beds as a function of the temperature and the compressive state of the bed, which is normally expressed in terms of inelastic volumetric strain. In this regard, a dedicated facility based on Hot Wire Method (HWM) [30] has been commissioned at KIT and thermal conductivity experiments at relevant HCPB temperatures and compressive states are being conducted (Fig. 12). The results are reported in [31] and have been used to set a minimum purge gas pressure of ~0.2 MPa to help reducing the tritium permeation to the coolant, as it has been observed that at lower pressures the conductivity drops significantly due to the Smoluchowski effect.

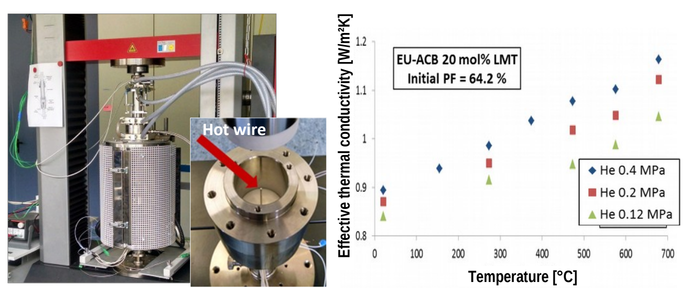

Fig. 12. Thermal conductivity experiments for CB materials. Left: HWM facility at KIT. Right: thermal conductivity experiments with advanced CB.

Also, R&D is being conducted in CIEMAT regarding the chemical characterization of advanced CBs in unirradiated and under ionizing radiation environment. The goal of the tests in unirradiated conditions is to investigate the compatibility between EUROFER97 and different CB batches produced by industry and KIT (SCHOTT and KALOS respectively), JAEA in Japan and CIEMAT, using different purge gas doping agents, namely 0.2%H_2 and 0.012%H_2O.

Due to the current lack of a facility producing a neutron spectrum relevant to DEMO (e.g. a DEMO-Oriented Neutron

Source, DONES), the approach in CIEMAT to simulate a similar environment is pursued in 2 steps:

1) Up to 2016: electron irradiation in a Van-der-Graaf (VdG) accelerator, and γ-irradiation facility (Nayade).
2) From 2017: use of high energy ion sources (proton beam irradiation facility at a MeV range) to simulate the CB damage with neutron irradiation.

For the chemical characterization with both purge gas chemistries temperature levels relevant to the interfacial temperatures expected in DEMO are set (350 °C and 550 °C) under gamma (4 MGy in Nayade) and electron irradiation (176.4 kGy in VdG facility).

Regarding the unirradiated tests, Fig. 13 shows the experimental set-up: the tubular furnace in Fig. 13-a) and Fig. 13-b), a schematic view of the set up in Fig. 13-c), the multi-chamber reactor in Fig. 13-d) and 2 example micrographs of the CB fabricated by CIEMAT after 336 hr and 3696 hr exposed to He+0.2%H_2 at 550 °C in Fig. 13-e).

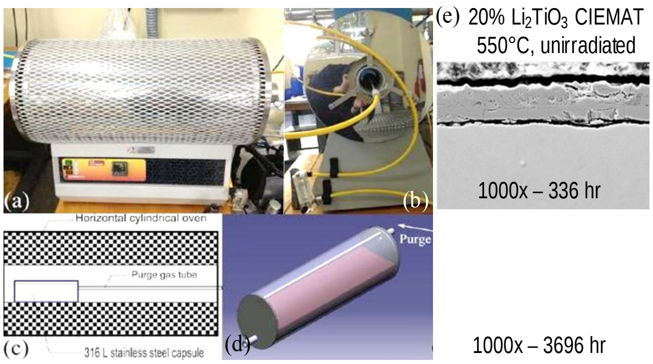

Fig. 13. Corrosion characterization of CB and EUROFER97 (unirradiated): a) and b) front and side view respectively of the tubular furnace; c) schematics of the experimental set-up; d) multi-chamber reactor; e) micrographs after 336 hr and 3696 hr of CB produced by CIEMAT tested under 550 °C with He+0.2%H_2 atmosphere and no irradiation.

The most remarkable observation has been the difference on the corrosion behavior between different CBs under same conditions: while the CBs fabricated by melt-spraying process (e.g. KALOS) show a negligible corrosion layer, the ones by emulsion or spray-dryer are significantly corroded. Also, although the corrosion potential of He+H_2O is expected to be greater than for He+H_2, other factors like the specific surface of the pebbles or atmosphere that is generated (outgassing of e.g. CO_2 from the pebbles can alter the purge gas composition) are also important in the corrosion mechanisms. More details on the chemical characterization tests and results can be found in [32].

Regarding the irradiated tests, CIEMAT conducts R&D on the characterization of the tritium release in advanced CB under ionizing radiation by analyzing the absorption/desorption characteristics. Deuterium is used instead of tritium in order to simplify the experimental set-up, without loss of relevance due to the similar isotope behaviors. The goal of these investigations is to demonstrate the viability of the desorption tests as an alternative method to assess the hydrogen transport in CB candidate materials.

Fig. 14 depicts the deuterium release test set-up of KALOS' advanced CB in the VdG facility. The batches of advanced CB to be tested, Fig. 14-a), are firstly dehydrated at 400 °C in vacuum. Next, they are loaded with deuterium: they are placed into a steel container, Fig. 14-b), kept in deuterium atmosphere at 1.3 bar under ionizing radiation in the VdG test rig, Fig. 14-c) and Fig. 14-d). Here, the deuterium loading takes place at different temperatures and dose conditions. Finally, the deuterium is desorbed (thermally-induced) up to 800 °C at 10 K/min, also at variable irradiation conditions.

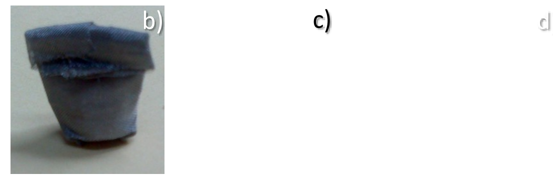

Fig. 14. Deuterium release under ionizing electron radiation: a) advanced CB produced in KALOS; b) steel container with advanced CB; c) desorption sample holder; d) position of the sample holder inside the VdG facility.

Fig. 15 shows some results from the deuterium release experiments in the VdG facility. It can be observed that adding Li_2TiO_3 does not significantly influence the desoprtion rate, yet it is marginally improved. Also, the ionizing irradiation and the temperature have positive effects in the desorption characteristics. More results are reported in [33] and [34].

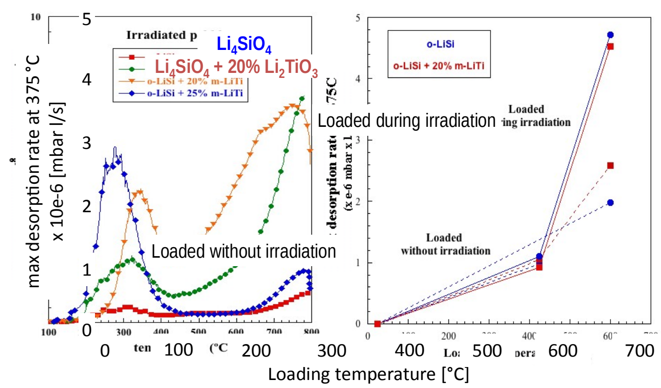

Fig. 15. Deuterium absorption/desorption tests under ionizing irradiation.

B. Neutron multiplier materials

The R&D performed in KIT on NMM is focused on the characterization of reference ITER and DEMO material and the development and characterization of advanced NMM based on Be intermetallic alloys (beryllides).

The development of NMM goes along 3 major routes:

1) Characterization of the reference ITER material, which is 1mm Be pebbles produced by Rotating Electrode Method (REM) by NGK Insulators Ltd., Japan (within several F4E contracts and WP1-HCPB in EUROfusion).
2) Due to the limited scalability of the REM, other processes are being investigated. A screening of possible cost-effective alternatives (particularly ball milling rounding of crushed billets and scrap from fluoride reduction) was undertaken within the FPA380-A3-SG01 contract for F4E.
3) Development and characterization of advanced beryllides (within WP1-HCPB in EUROfusion).

1) Characterization of irradiated pebbles

The performance of NGK pebbles under irradiation has been characterized within two HIgh DOse irradiation of BEryllium campaigns (HIDOBE) at the target irradiation temperatures of 425 °C, 525 °C, 650 °C and 750 °C: HIDOBE-01 up to generation of 3000 appm He and HIDOBE-02 with 6000 appm He, corresponding to ~1/3 of that at the DEMO end-of-life. Results of the post irradiation experiments can be shortly summarized as followed:

1) Creep: pebbles irradiated at 650 °C and 750 °C creep-deform easier than irradiated at 425 °C and 525 °C [35].
2) Swelling: double for pebbles irradiated in HIDOBE-02, linearly increasing with temperature up to 4% at 650 °C. Post irradiation annealing suggests that swelling accelerates significantly above 650 °C [36] and material breaks when the relative volume increase is 25-30% [37].
3) TEM investigations revealed formation of large bubbles along grain boundaries [38].
4) Tritium release tests after irradiation revealed a gradual decrease of the amount of tritium retained with increasing irradiation temperature: about 40% of generated tritium remains trapped at 600 °C [39].
5) Long-lived activation of Be pebbles: origin mainly due ^{60}Co, ^{55}Fe and ^{54}Mn impurities.

2) Characterization of unirradiated pebbles

Thorough characterization of unirradiated beryllium pebbles by various methods has been performed, namely:

1) Optical microscopy, showing frequent formation of central shrinkage pores or distributed technological porosity within pebbles produced by REM [40].
2) Tritium release tests after thermally induced tritium loading, demonstrating higher tritium uptake by the Be-pebble batch with higher internal porosity [40].
3) Experiments on oxidation in synthetic air, indicating significant oxidation above 800 °C and excessive reaction with water vapour starting above 1050 °C, being the reactivity of Be pebbles higher the larger the porosity.
4) Compatibility studies with EUROFER9, revealing the absence of interdiffusion layer at temperatures <600 °C. Also, the effect of post welding heat treatment (0.5h at 900 °C), which is to be performed after filling the BB modules with Be-pebbles, has been simulated and showed no interaction layer at contact zones between pebbles and surrounding EUROFER97.

3) Development of advanced beryllides

KIT in collaboration with TU Berlin has developed a hot extrusion method to fabricate compacted Be-Ti semi-products in form of rods from Be-Ti powder mixtures (Fig. 16-left) [41]. The rods are a two-phase material, therefore an homogeneization annealing is required to obtain a one-phase beryllide. Extruded two-phase rods have sufficient strength for production of beryllide pebbles by REM. The insert right-down demonstrates $Be_{12}Ti$ pebbles (Fig. 16-right) produced by REM in Rokkasho, Japan, from the rods fabricated by KIT.

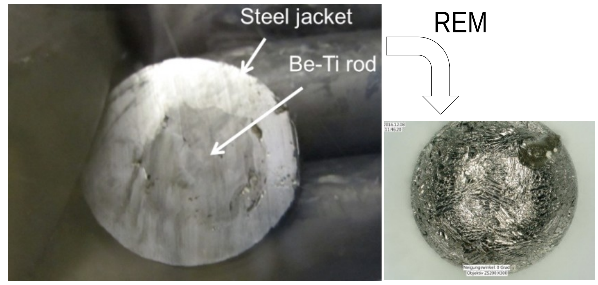

Fig. 16. Fabrication of Be-Ti. Left: hot-extruded rod of Be-Ti in steel jacket. Right: Be-Ti pebble produced by REM.

V. R&D in Manufacturing and Mock-ups Testing

The goal of the manufacturing R&D is the realization of the HCPB BB key components by developing, standardizing and qualifying different processes, preparing specifications and placing industrial contracts towards the establishment of long-term development collaborations.

Fundamental technologies investigated for the HCPB BB subcomponent manufacturing are e.g. Electrical Discharge Machining (EDM), cold forming of plates and recently the introduction of Additive Manufacturing (e.g Selective Laser Sintering, SLS). From the point of view of the joining techniques, Electron Beam (EB) is considered as a reference process. The different processes can be used separately, but the fabrication of some components may need to combine some of them. E.g., straight cooling channels with constant cross section are manufactured by drilling a pilot hole in the center and subsequently cutting the final channel section by EDM. Also the internal channel surface can be equipped with structures manufactured e.g. by EDM for heat transfer augmentation [43] (see related topic in section VI-B).

Deep-hole drilling can be performed for typical channels with length-to-diameter ratios ~150 (e.g. ~2200 mm long FW with channel ∅15 mm) and ~300 for channels in cooling plates in BZ [45]. Much larger ratios can be reached by realizing pilot holes for the EDM in a process chain consisting of milling, EB-welding, and High Isostatic Pressure (HIP) bonding [44].

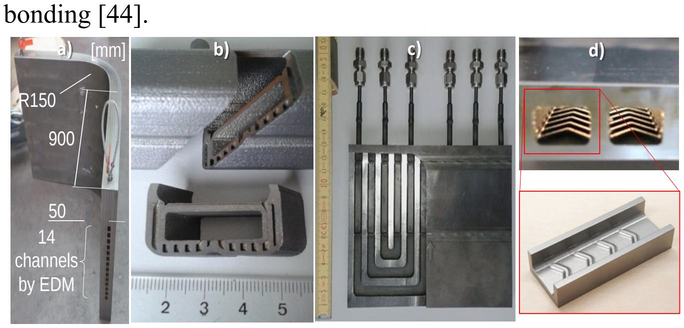

Fig. 17. Technology demonstrators: a) ½ FW mock-up by EDM; b) multichannel divertor mock-up produced by SLS; c) hybrid production of complex parts by SLS and conventional CNC machining joined by EB welding; d) Rib-roughened FW cooling channels through die sink EDM.

Non-planar plates with internal channel structures are realized by cold forming e.g. to create L or U-shaped plates [38] and [45]. Fig. 17-a) shows a relevant scale demonstrator of ½ FW (straight length before forming = 900 mm, 14x channels 10x15 mm², bending angle 90°, ext. radius

~ 200 mm) combining EDM for the cooling channels and cold forming.

For the realization of flow channels with complex structures/geometries, SLS has been investigated using metal powder with composition relevant to that of EUROFER97, and a preliminary qualification has shown very similar (80%÷90%) values to reference EUROFER97, with relevant macro- and micro-structures. Demonstrators of complex multi-channel, multi-wall structures have been manufactured by SLS [46], Fig. 17-b). Also, hybrid components have been produced by joining with EB-welding complex parts produced with SLS and simpler ones by standard machining, Fig. 17-c). Future R&D considers the implementation of SLS into codes and standards.

In order to increase the heat removal rate of the FW, heat transfer augmentation techniques based on rib-roughened structures are to be applied on the plasma side of the cooling channels (see section VI-B). The fabrication feasibility of such structures has been demonstrated with the use of die-sink EDM [42]. Fig. 17-d) shows an example of 2 rib-roughened FW cooling channels, with an augmented detail of the ribs.

Regarding mock-ups testing, dedicated facilities exist at KIT for large mock-ups and medium temperatures up to 500 °C (HELOKA, [47][48]) and for small mock-ups at high temperatures up to 800 °C (KATHELO, [49]), both at 8 MPa. E.g., planned testing involves the experiment of a full-scaled FW with 2 channels and V-rib-roughened structures (see VI-B) at 1 MW/m² heat flux. Other tests are being planned to test the integral TH behavior of a segment, mainly for the validation of TH codes build for the HCPB.

VI. R&D in Heat Transfer Augmentation for the FW

A. The FW issue

Key differences of DEMO with respect to ITER are that the former has to prove a net electric output, together with the demonstration of tritium self-sufficiency. These two functional requirements completely change the approach for dealing with high heat fluxes on the FW of both reactors: while ITER can afford thick water-cooled FWs on high conductive structures (CuCrZr) with heat flux removal capabilities of 4.7 MW/m² [55] under a low neutron fluence, DEMO has to maximize the heat removal capability with a thin FW in order to avoid parasitic absorption and ensure the tritium breeding capabilities and being made of a much lower conductivity material. This pose a technological limit of 1÷1.5 MW/m² to the FW heat transfer capability, regardless of the coolant medium, as it has been demonstrated that the EUROFER97 thermal conductivity [50] and TM considerations [65][56] are the limiting factors.

B. Heat transfer augmentation design, analysis and testing

Due to the technological constraints exposed above, an effort is being conducted in order to design the FW so that the sum of the FW heat fluxes from the charged particle and the radiation do not exceed these limits [54][55].

On the other side, a heat transfer rate in the coolant is needed to be able to dissipate about ~1÷1.5 MW/m². Gas cooling and in particular helium cannot reach high heat transfer capabilities at high speeds without resulting in large pressure drops and circulating power, unlike uncompressible media such as water. However, high heat transfer rates can be also alternatively achieved by means of heat transfer augmentation techniques. In this regard, a dedicated R&D program is being conducted in KIT and in particular, in the use of rib-roughened channel walls ([50][51][52][53]), which, as it has been demonstrated in [50], the benefits on heat transfer augmentation by rib-roughened walls are higher than by simply increasing the gas speed in terms of pressure drop.

Fig. 18-bottom show the CFD results after the exploration of different rib-roughened geometries for the FW [50].

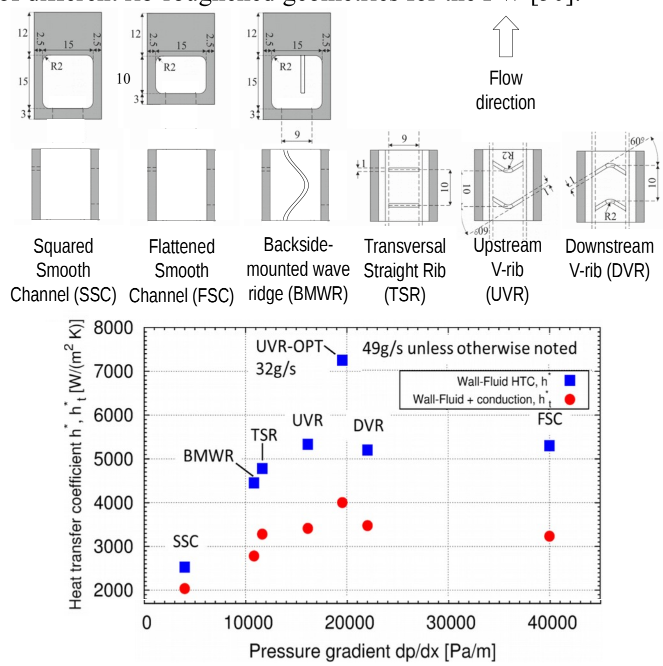

Fig. 18. FW heat transfer augmentation by rib-roughened structures. Top: different rib geometries. Bottom: heat transfer coefficient vs. pressure drop for the different rib geometries.

It has been found that the V-ribs show the best performance ratio between heat transfer and pressure drop. These structures have been successfully applied by EDM on a relevant FW mock-up (Fig. 18-top). Currently research is being performed in order to validate the CFD results in a dedicated facility (HETREX) and a future test in HELOKA at 1 MW/m².

VII. HCPB Key Interfaces

A. R&D on the HCPB Tritium Extraction and Removal

The development of the TER system for the HCPB BB is supported by various activities carried out in the frame of the WPBB. For the definition of the functional requirements and the design requirements, the modeling of the tritium transport phenomena, together with the development of the tritium control strategy, play a major role. These developments are complementary with the experimental activities aiming to quantifying the efficiency of the technologies that have a potential for application at a DEMO scale.

A dedicated assessment based on a multi-criteria decision analysis has been carried out in order to identify the most suitable technology to be used for the HCPB TER in DEMO [57]. This assessment has shown that the cryogenic approach is a mature and demonstrated method and hence it has been selected as a baseline technology, while the combination of permeation on zeolite membrane and catalysis is still at a R&D level and it is kept as back-up solution.

Depending mainly on the purge gas chemistry and the temperature, the tritium is released and extracted from the CB pebbles in its both molecular (Q_2) and oxidized (Q_2O) forms (Q meaning one of the hydrogen isotopes ^{1}H, ^{2}H, or ^{3}H). Several studies have been carried out to accurately predict the respective amount of both species (to be expressed for example as the Q_2O/Q_2 ratio), which strongly depends on many parameters such as the CB material, the temperature and especially the purge gas composition. For the design of the HCPB TER for both the reference and back-up solutions, the following compositions of the purge gas and are being used: H_2 = 95.8%; Q_2 = 0.73%; H_2O = 3.47%; Q_2O = 0.03%.

The cryogenic approach (Fig. 19-left) works on the basis of trapping/adsorption of Q_2O in the Reactive Molecular Sieve Bed (RMSB) and the adsorption of Q_2 in the Cryogenic Molecular Sieve Bed (CMSB) at cryogenic temperatures (77K). Tritium is recovered from the RMSB via catalytic isotope exchange between the Q_2O and a hydrogen/deuterium gas and from the CMSB by regeneration at temperature above 400 K. The tritium recovered from the RMSB and CMSB is further processed in the Tritium Plant.

The permeation on zeolite membrane approach (Fig. 19-right) uses a cascade of zeolite membranes as a pre-concentration stage. In this stage a He enriched phase is obtained at one side of the cascade and reused in the purge gas flow, while Q_2O and Q_2 enriched phases are collected on the other side. The tritium is recovered later from the Q_2O and Q_2 enriched stream by a catalytic membrane reactor (PERMCAT).

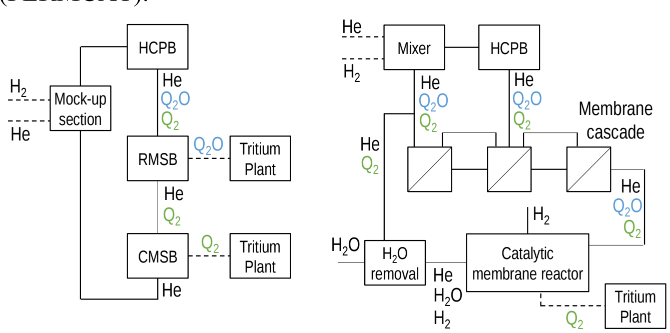

Fig. 19. HCPB TER system. Left: reference design based on cryogenic approach. Right: back-up design based on zeolite membranes and PERMCAT.

Both methods offer some advantages and disadvantages: while the permeation on zeolite membranes offers a simultaneous and continuous treatment of Q_2O and Q_2, minimizing the tritium inventory, nevertheless, at the expense of using many cascade stages resulting in many components, a complex control and large power consumption. On the other side, the cryogenic approach is a mature technology but relatively bulky and with a large liquid N_2 consumption. Several synergies have been identified at a BB design level to assist the TER to achieve a more balanced system, namely: (1) study a reduction of the current flow rate of purge gas (10000 Nm³/hr) and (2) study the use of alternative purge gas chemistries, using denser dopants and/or carrier gases.

B. R&D on the HCPB Balance of Plant (BoP)

The BoP conforms a key interface system to the HCPB BB. As stated in [5], the conceptual design of the BoP of DEMO requires not only engineering and modeling activities, but also factors as efficiency, safety, costs and especially Reliability, Availability, Maintainability and Inspectability (RAMI), where the technology readiness of the system's components will be assessed. Therefore, in order to build a credible BoP, the BB has to accept several additional restrictions in order to allow an efficient BoP system with low R&D needs, using proven and mature technologies where possible.

The current conceptual design configuration for the DEMO HCPB BoP is shown schematically in Fig. 20, with typical temperatures of the plasma burn-time during a pulse.

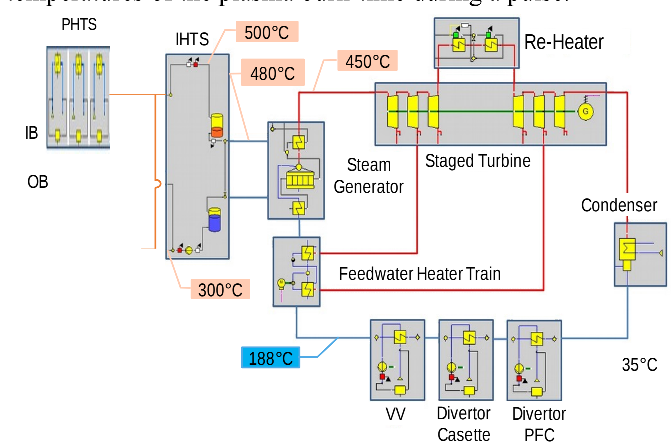

Fig. 20. Schematic representation of the HCPB BoP system.

The BB heat is transferred to the PCS (Rankine cycle) via PHTS and Intermediate Heat Transfer System (IHTS). The IHTS comprises an Energy Storage System (ESS) of molten salt, similarly as in Concentrated Solar Power (CSP) plants, in order to accumulate part (~20%) of the heat of the BB during the burn-time, thus releasing it during the dwell-time. The heat accumulated during the burn-time is enough to replace the missing BB heat (the HCPB decay heat of only ~1%) in the dwell time. This avoids a non-continuous supply of steam to the PCS and therefore of electricity to the net, which would imply a pulsating operation of the power train (turbine and generator). Similar situation exist in fossil power plants, where the use of an ESS is presently under consideration to extend the lifetime of the components by reducing plant transients. The amount of solar salt to accumulate such heat is ~11300 ton, which is a typical size for CSPs plants.

Based on the current DEMO baseline (18 sectors), the heat in the HCPB BB (2389 MW) is collected from the IB (3x 235 MW) and OB (6x 303 MW). The ratio of IB-to-OB number of loops is selected in order to (approximately) match the ratio of the power transferred in the IB and OB, i.e. ~70%

in OB and ~30% in IB. On the other side, each PHTS loop will be connected to the IHTS (i.e. the ESS) via an Intermediate Heat Exchanger (IHX). Then, the total number of loops (3 IB + 6 OB) is driven to keep a reasonable power (i.e. size) of these IHXs, which is considered to be ~300 MW. The number of circulators per loop is also a function of the current state-of-the-art on the technology of helium circulators, which currently is in the range of 5÷6 MW. For the current total current circulating power of ~150 MW (~18 MW per loop), a tandem of 2 circulators of ~8 MW (i.e. 16 circulators per reactor) is predefined, as this power is considered to be a reasonable extrapolation from the current state-of-the-art. Here the importance of keeping the pressure drop in the PHTS becomes clear, and in particular in the BB system as the highest contributor to the pressure drops, resulting in a challenging additional constraint to the design of the HCPB BB. The tandem approach of the circulators also guaranties 50 % cooling power in case of a compressor failure.

Aside to the BB, the heat sources in the VV (86 MW), the divertor cassettes (115.2 MW) and the divertor PFCs (136 MW) are added to the PCS as an additional feedwater heating, having each its own water pump and heat exchanger. In order to ensure that the correct amount of heat is being transferred to the PCS and that the returning water to each of these three heat sources has its predefined parameters at any phase of the DEMO operation, feedwater flow at the three heat exchangers is being split into a main flow and a by-pass flow.

Staged turbines with high pressure (HP) and low pressure (LP) stages have been assumed. Two steam re-heaters are assumed using hot solar salt from the ESS between the three HP stages and the three LP stages, to avoid condensation in the LP stages. The return temperature of the solar salt should always be controlled in order not to drop down the temperature of the solar salt excessively (<220 °C), to avoid risk of freezing.

The net electrical power produced with this configuration is ~659 MW during the burn-phase and 713 MW during the dwell time. The higher electricity production during the dwell time is due to the consideration of the He mass flow reduction, as in the dwell time the BB heat source is only from decay heat. Taking into account this assumption, the solar salt mass could be reduced to ~9000 ton.

These numbers lead to a PCS thermal efficiency of about 36%, which is lower than other gas-cooled reactors like AGR or HTR (40÷42%). The reason of the lower efficiency resides in the lower temperature and pressure of the steam at the inlet of the HP turbine. Although the outlet temperature of the HCPB is 500 °C, the steam inlet temperature at the HP turbine is lowered to 450 °C (in comparison e.g. to the 538°C of AGR reactors [58]) due to the intercalation of the IHTS between the PHTS and the PCS. However, this is still higher than the values achieved in a PWR (inlet HP turbine at 274 °C at 59 bar [58]). This lower temperature limits also the steam pressure to ~60 bar at the HP turbine in DEMO (compared e.g. to 167 bar in an AGR [58]). Therefore, research is being conducted to evaluate the possibility of increasing the BB outlet temperature (hence also the inlet steam turbine temperature) and further reducing the pressure drop of the HCPB BB in order to reach thermal efficiency values closer to a AGRs or HTRs (~40%). As discussed in section II-B-3, if the improved high temperature properties of the advanced EUROFER97 will be confirmed also under irradiation conditions, a significant increase in the inlet turbine temperature and pressure can be envisaged, making efficiencies of >40% more tangible.

C. R&D on HCPB safety analyses

Safety studies accompanying the design and development of the HCPB BB are essential to ensure the fulfillment of the BB safety requirements while being compatible with the BB functionalities. Based on the Functional Failure Mode Effect Analysis (FFMEA) results for the HCPB blanket system, Postulated Initiating Events (PIEs) have been identified and consequently critical event sequences have been selected [59].

KIT focuses on the safety analyses for the representative accidental sequences of the Design Basis Accidents (DBAs) in the current pre-conceptual design phase. DBAs such as loss of flow accident (LOFA), in-box LOCA and in-vessel LOCA have been investigated. For the previous ITER-like HCPB design in 2014, the FW has been modeled using CFD code ANSYS CFX [60] and system code RELAP5-3D [61]. for LOFA and LOCA analyses with respect to various mitigation actions, such as the cooling circuit redundancy, the circulator redundancy, the plasma shutdown and their combination. The results have been further used to characterize the behaviour of BB relevant parameters under the fast transients characterizing such accidents, as well as to identify a suitable experimental setup to validate the numerical models [62].

In the deterministic accident analysis, a LOCA in a BB module has been investigated for three representative accident scenarios using MELCOR186 for fusion [63]: (1) in-box LOCA with failure of one horizontal plate, (2) in-vessel LOCA with failure of the FW channels, and (3) in-box LOCA to the purge gas system with failure of one CP of the BZ. Due to the constant design progress, each BB LOCA is updated for the most current version of the HCPB BB (Fig.2) and currently being integrated in one associated OB loop of the PHTS. For all cases, different scenarios are simulated with respect to break sizes in the BZ and the FW and plasma shutdown conditions. Results of the mass flow rate, pressure, He temperature, structure temperature and mass are evaluated. The plasma disruption inducing high heat flux load on the FW has an impact on the FW peak temperature and its duration. In this respect, the possibility to use advanced EUROFER97 (II-B-3 is recommended from the safety point of view. The modeled loop is also valid for further DBAs such as ex-vessel LOCA, LOFA, loss of power, etc.

The accident is initialized during the normal operation the pulse flat top, hence the He inventory of the modeled OB loop at that time can be determined. The large He quantity (~9.5 ton in all 6 OB loops) is a challenge for dimensioning of the expansion volume (EV) in case of the in-vessel LOCA. A combined VVPSS and EV concept is being explored for both water and He to protect the VV integrity [64]. This issue has triggered some studies also at a BB design level and, currently, studies are being currently conducted to check the possibility to reduce the He coolant pressure to HTR values (6÷7 MPa). Also, this issue has revealed that former configurations considering pipework through the upper and lower ports are not convenient, as they increase the length of the piping and therefore the overall He inventory.

The worst plasma event due to a runaway electron event which can lead also to an in-vessel LOCA event due to the FW failure is being investigated as well. To study accidental scenarios during the pulsed operation, the HCPB blanket concept in the associated PHTS and auxiliary systems of BoP will be modeled using RELAP5-3D in the next step.

VIII. Systems Integration

In the current EU DEMO pre-conceptual phase it has been identified [5] that the large number of systems to be integrated into the VV and their complex interdependencies plays a decisive role in their design, as the requirements of all of them will likely drive their design. The current strategy to integrate these systems has been to share the effort between the different WPs in the WPBB [68]. In the particular case of the WP1, the integration activities are regarding the development and integration in the HCPB of a BB attachment system and the integration studies of the pellet injection fuelling lines.

A. EM analyses and BB attachment system integration

Due to the electrical and ferromagnetic properties of the EUROFER97, static electromagnetic (EM) forces (in the following called Maxwell forces) as well as transient Lorentz forces can be induced in the BB segments, especially during off-normal events involving plasma disruptions.

The BB segments are connected to the VV by the so-called BB attachment system, which has the task to support the weight of the blankets, to allow the thermal expansion of the blankets and to resist seismic loads as well as EM loads during normal and off-normal operation. In doing so, it has to limit the deformation of the segments in order to ensure their structural integrity while permitting their thermal expansion. In addition, the design of the BB attachment system has to be compatible with the numerous requirements from the remote handling operations for the installation and removal the segments from the VV, minimizing the plant maintenance schedule to ensure the maximum plant availability.

Different concepts have been developed over the past years, which are all based on an arrangement of different attachment elements comprising key and slot combinations, as well as other support structures. A sector of the DEMO reactor is shown in Fig. 21-left with supports at the bottom, keys at top and center of the IB segments and keys at top and pads at center of the OB segments. A BB transporter is used to manipulate the BB segments during remote handling operations. The present design of the attachment system is described in detail in [66].

The design process is supported by the investigation of the structural integrity of the BB segments under different normal and off-normal operating conditions, which include plasma disruptions and ex-vessel LOCAs. During a plasma disruption, high Lorentz forces have to be considered in addition to Maxwell forces, the later also present under normal operations. EM analyses based on a schematized FE model have been conducted to obtain the Lorentz forces during a simplified central plasma disruption [27]. These analyses show very high radial and poloidal moments acting on the segments. The structural assessment of the BB segments with the present design of the BB attachment system under normal operation and a central plasma disruption including Maxwell and Lorentz forces is reported in [66].

The structural integrity of the BB segments during an ex-vessel LOCA and the effect on the VV are discussed in [67]. During this event, there is an additional increase of the temperature on the BB segments, leading to an additional thermal expansion. Therefore, the attachment system has to accommodate this additional deformation in order to limit the reaction forces on the VV [64].

B. Pellet injection fueling lines integration

The so-called fueling lines in DEMO are the system of tracks guiding the nuclear fusion fuel ice pellets of deuterium and tritium from outside to the inside of the VV, penetrating the BB system and delivering them to the center of the burning plasma. The current design of the fuelling lines for DEMO envisages a guiding track in each sector, located in-between two IB segments. Fig. 21-right depicts the integration of a fuelling line guiding track through two IB segments. These guiding tracks pose additional constraints in the design of the BB, which must provide local openings to ensure an adequate clearance for these guiding tracks and the ice pellets. Studies are being conducted to check how close the ending tip of the guiding tracks can get to the plasma side and at the same time which is the minimum clearance that the guiding tracks can allow, in order to minimize the neutron streaming.

A detailed summary of these integration activities and for other systems with different BB concepts can be found in [68].

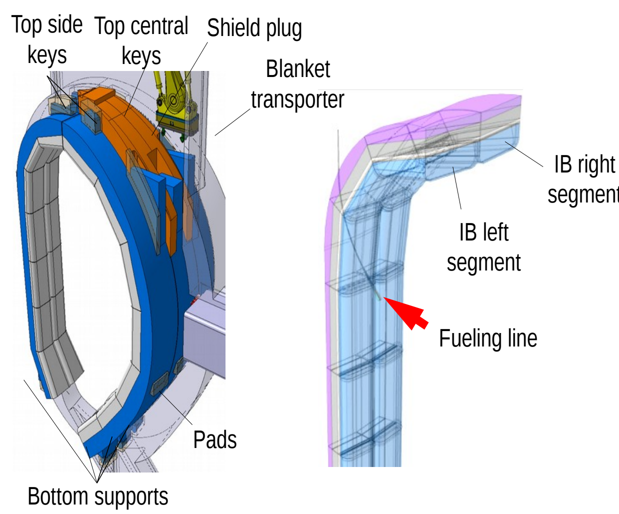

Fig. 21. Integration studies. Left: BB attachment system. Right: fuelling lines.

IX. Standing Challenges and Future R&D Plans

As stated in the EU Fusion Roadmap [1] and highlighted by in the R&D development plan in EUROfusion [5], a "pragmatic approach" is pursued and for the EU DEMO, meaning that mature technologies should be taken as basis for the system's integration or only small extrapolations of them, whenever possible. For a helium-cooled BB such as the HCPB, the interfacing system that can potentially benefit from already established technology is the BoP and therefore, current available He technology, mainly a heritage from the long, successful AGR-program in UK, should be assumed. However, the current circulating power to run a HCPB-DEMO is ~150 MW, which is about 2÷3 times that of AGR or HTR-like reactors, posing a technological question on the possibility to use current off-the-shelf solutions. As the main contributor to the circulating power is still the BB, future design investigations will be directed to optimize the HCPB in order to convert the BB into a "near-term" system, with circulating powers closer to gas-cooled reactors, i.e. 60÷70 MW. This will likely force a rather strong conceptual change of the current HCPB, so that the heat transfer between the pebble beds and the coolant will be driven mainly due to an increase of heat transfer area, instead of relying on high heat transfer rates by having large flow speeds, as it has been historically the case, resulting in large penalties in pressure drops. Similarly, the heat transfer capability of the FW will be also augmented through rib-roughened turbulence promoters, instead of by purely increasing the flow speeds, in an attempt to cope with the large high heat fluxes while keeping relatively low pressure drops.

If this target can be achieved, the next step will be the investigation on the limits for a reduction of the coolant pressure and evaluation of the chain of advantages mainly related to safety point of view in comparison with the main disadvantage, which is the increase on pressure drop.

Also, research will be conducted to check the possibility to exit the BB at a higher temperature while still meeting the C&S. This will help (1) to increase the temperature difference in the IHXs, which will increase the heat transfer rate while keeping a more compact device, thus reducing the pressure drops through the IHX and further reducing the circulating power and (2) to increase the net cycle efficiency. Also, a tighter collaboration with WP-MAT will take place, in order to evaluate the readiness of the advanced EUROFER97 and its impact in the coolant temperature range for a boost in the thermal and net plant efficiency.

Further R&D for the development of CB and NMM will be conducted, with the mid/long term perspective of performing in-pile irradiation tests. The possibility to include in the program additional high tritium-performance CB materials will be assessed. In the case of Be, an additional research effort will be conducted in order to better understand the tritium transport phenomena in the Be/beryllide pebbles. Also, current KIT research in the area of the pebble bed thermomechanics will be includd in the EUROfusion program, in order to reach a full understanding of the HCPB behavior during operation.

X. Summary

The current roadmap for fusion electricity in Europe foresees the beginning of the construction of DEMO in less than 20 years. This poses new requirements in the design of the HCPB BB, whose design cannot be anymore approached as a standalone system for a long-term device, but it has to be confronted as one of the most highly integrated systems in DEMO and with the increasing need to be compatible with nowadays state-of-the art technologies. This is reflected in this paper, which briefly outlines the current R&D activities in the frame of EUROfusion not only from the point of view of the BB design, analyses and functional materials development, but also with especial emphasis to all the interfacing systems and functional areas that are gradually acquiring a key role as design drivers of the HCPB BB system.

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