



EUROfusion

WPBB-CPR(18) 19350

G Federici et al.

An Overview of the EU Breeding Blanket Strategy as Integral Part of the DEMO Design Effort

Preprint of Paper to be submitted for publication in Proceeding of 30th Symposium on Fusion Technology (SOFT)



This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the Euratom research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

This document is intended for publication in the open literature. It is made available on the clear understanding that it may not be further circulated and extracts or references may not be published prior to publication of the original when applicable, or without the consent of the Publications Officer, EUROfusion Programme Management Unit, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK or e-mail Publications.Officer@euro-fusion.org

Enquiries about Copyright and reproduction should be addressed to the Publications Officer, EUROfusion Programme Management Unit, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK or e-mail Publications.Officer@euro-fusion.org

The contents of this preprint and all other EUROfusion Preprints, Reports and Conference Papers are available to view online free at <http://www.euro-fusionscipub.org>. This site has full search facilities and e-mail alert options. In the JET specific papers the diagrams contained within the PDFs on this site are hyperlinked

An Overview of the EU Breeding Blanket Design Strategy as Integral Part of the DEMO Design Effort

G. Federici¹, L. Boccaccini², F. Cismondi¹, M. Gasparotto³, Y. Poitevin⁴

¹ EUROfusion Consortium, Boltzmannstr. 2, Garching 85748, Germany

² Karlsruhe Institute of Technology, Hermann-von-Helmholtz-Platz 1, 76344 Eggenstein-Leopoldshafen, Germany

³ 00046 Grottaferrata (Rome), Italy

⁴ Fusion for Energy, Route de Vinon-sur-Verdon, S 90 046-13067, St Paul Lez Durance, (France)

This paper provides an overview of the newly revised design and development strategy of the DEMO breeding blanket in Europe. This has been defined to take into account the input from the DEMO pre-conceptual design activities, the findings and recommendations of a thorough technical and programmatic assessment of the breeding blanket and the EU ITER TBM programs, recently conducted by an independent expert panel. This work has led to the identification of (i) the most mature and technically sound breeding blanket concepts to be used as “driver” breeding blanket in DEMO and as “advanced”, the latter to be installed and tested in a limited number of properly designed segments, having the potential to be more attractive for future fusion power plants; (ii) the remaining technical gaps and R&D priorities. A number of urgent steps that are required to better align and strengthen the EU TBM and DEMO Breeding Blanket Program as a whole and to aim at an efficient implementation of the work are described in this paper. These include a proposal to change the EU TBM options to be tested in ITER in order to obtain important and useful information from the presently two considered breeders (solid and liquid) and coolants (helium and water) for DEMO.

Keywords - DEMO, Power Plant, Breeding Blanket, Design Integration, Tritium

1. Introduction

As an important part of the Roadmap to Fusion Electricity [1], Europe is conducting a pre-conceptual design study of a Demonstration fusion power plant (DEMO) to come in operation around the middle of this century with the main aims to demonstrate the production of few hundred MWs of net electricity and to demonstrate feasibility of operation with a closed-tritium fuel cycle, including maintenance systems capable of achieving power-plant relevant availability [2]. This is currently viewed as the remaining crucial step towards the exploitation of fusion power after ITER, not only in Europe but by many of the nations engaged in the construction of ITER. The DEMO design and R&D activities in Europe are expected to benefit largely from the experience gained from the design, construction and operation of ITER. Nevertheless, there are still outstanding gaps that need to be timely overcome, requiring a vigorous physics and technology R&D beyond ITER.

The breeding blanket and associated tritium processing system for DEMO comprise a complex system with multiple functions, materials, loads, and environmental conditions that must operate safely and reliably from day-one. Achieving tritium self-sufficiency will be an unescapable requirement for any next-step fusion nuclear facility beyond ITER. However, no fusion blanket has ever been built or tested. Hence, its crucial integrated functions and reliability in DEMO and future power plant are by no means assured. However, the program in Europe benefits of many years of design and R&D, primarily carried out in European Fusion Laboratories. In addition, ITER presents a first and unique opportunity to test the response of breeding blanket materials and representative

component mock-ups, specifically called Test Blanket Modules (TBMs) at relevant operating conditions, in an actual fusion environment, albeit at very low neutron fluences (see for example [3,4] and references therein). Recent work on DEMO pre-conceptual design in Europe has brought forward the need to launch a critical reevaluation of the strategy for breeding blanket design and technology development that minimise the risks that could jeopardize the effort to arrive on time to self-consistent breeding blanket design solutions, addressing both materials and engineering issues and extracting maximum benefit from the ITER (TBM) program.

This paper describes the newly revised design and R&D strategy of the breeding blanket in Europe that has been defined to take into account the input from the DEMO pre-conceptual design activities and the findings/recommendations of a thorough technical and programmatic assessment of the EU DEMO breeding blanket and EU ITER TBM programs, recently conducted by an independent expert panel. This was conducted to identify, among the available options, the most mature and technically sound candidates for breeding blanket concepts to be potentially used as “driver” blanket in DEMO, the remaining technical gaps and to align and strengthen the supporting R&D Program. To ensure a coherent and efficient Program, a change of the EU TBM options to be tested in ITER is proposed in order to obtain important and useful information from the two considered breeders (solid and liquid) and the two coolants (helium and water). Sect. 2 provides an overview of the adopted EU DEMO staged-design approach and the interdependency and technical input expected from ITER. Sect. 3 describes the main blanket design constraints and integration issues in DEMO and the important role of the ITER TBM Program, Sect. 4 introduces the key aspects of

the new DEMO Breeding Blanket/ EU TBM design strategy as Integral Part of the DEMO Design Effort. Sect. 5 provides a brief description of selected and recent achievements about the two most attractive breeding blanket concepts presently considered in Europe

2. DEMO in the EU roadmap

2.1 Programmatic and timeline considerations

At present, the DEMO design has not been formally selected and detailed operational requirements are not yet available. However, the DEMO plant high-level requirements have been defined following interaction with an external stakeholder group composed of experts from industry, utilities, grids, safety, licensing, etc. The design should be capable of producing electricity (up to ~500 MWe), operating with a closed fuel-cycle and to be a facilitating machine between ITER and a future First-of-a-Kind (Foak) commercial fusion power plant (FPP). The approach advocated by the EU fusion roadmap, is to consider in the early design phase a plant concept that would rely as much as possible on mature design solutions and technologies and the knowledge basis acquired with the design, construction and operation of ITER. It is argued that by delaying the design of DEMO in anticipation of the ultimate advances in plasma physics and technology would postpone the realization of fusion indefinitely [1]. Thus, emphasis has been placed from the very beginning on the study of key design integration issues that are foreseen to affect the whole DEMO nuclear plant architecture, arising from remote maintenance, power conversion, safety, licensing, and technology readiness aspects. The risk of postponing integration, assuming that it restricts innovation and inhibits an attractive DEMO plant, is that designers remain oblivious of integration issues and developing design solutions that cannot be integrated in practice. Thus, an early system integration work is essential to develop an understanding of the importance and relative difficulties of various design integration and technological problems to be solved in a DEMO Plant. It provides the context for further design improvements and contribute to guide future R&D. To this extent, contacts were also made with Gen IV fission and ITER to learn from their experience. Both projects emphasized the following aspects: (i) the plant design should drive R&D and not the other way around. (ii) fusion is a nuclear technology and as such, will be assessed with full nuclear scrutiny by the regulator. To this end, early engagement with a licensing consultant is needed to understand and tackle potential safety implications through design amelioration; (iii) the need for a traceable design process with a rigorous Systems Engineering approach; (iv) the technical solution should be based on maintaining proven design features to minimize technological risks [2].

2.2 ITER and DEMO schedule dependencies

The EU Fusion Roadmap emphasizes ITER as the crucial machine on which the validation of the DEMO physics and part of the technology basis depends. There is

therefore a high degree of schedule dependency between ITER and DEMO, although the ‘success-oriented’ approach outlined here advocates concurrency between the exploitation of ITER and development of the DEMO design. In this approach, the DEMO design activity proceeds in parallel with the ITER exploitation, but relies on a progressive flow of input from ITER for design and physics basis validation prior to authorization of DEMO construction. The DEMO design validation from ITER should not be seen as a single discrete event, but rather as an ongoing and progressive flow of information into the program – allowing continuous validation of specific aspects of the DEMO design, and if necessary, updates to the baseline.

Fig. 1 provides an overview of the analysis of dependencies identified between the revised DEMO and ITER schedules. The most critical and final major validation input, is the demonstration of D-T burning plasma scenarios in ITER that are scheduled to start around 2037 (with Q=10 short pulse in 2037 and long pulse in 2039).

In light of the above, the present DEMO development plan consists of the following three phases: (i) a Pre-Concept Design Phase to explore a number of DEMO plant concepts and develop system requirements up to 2020 (ii) a Concept Design Phase to mature and validate the baseline concept up to 2027¹; and (iii) an Engineering Design Phase beginning roughly around 2030 to develop the detailed design and prepare for the launch of major procurement activities around 2040’s, after ITER nuclear operation has confirmed the robustness of the underlying assumptions.

2.3 The Role of the ITER TBM Program

The design, R&D and testing of TBMs in ITER is viewed as an essential step to reduce the remaining technical risks and uncertainties associated with the demonstration of power extraction and tritium breeding technologies essential for a DEMO fusion power plant. This is required for: (i) developing and validating the scientific understanding and predictive capabilities; (ii) demonstrating the principles of tritium self-sufficiency in practical systems; (iii) developing and qualifying the breeding technologies to be used in next-step machines; (iv) providing the first integrated experimental results on safety, environmental impact, and efficiency of tritium extraction systems; and (v) providing initial components and operational reliability data for different ancillary systems (e.g, PbLi circuit, cooling systems, coolant purification systems and tritium extraction systems). The lesson to be learnt by the design and R&D of the ITER TBMs (both breeding boxes and ancillary systems) is viewed to be particularly valuable to aid the development and the down selection of the DEMO breeding blanket concept and will be discussed later in this paper.

¹ It should be noted that a transition phase of 2-3 years is expected for the concept design review consolidation and preparation of the Engineering Design Phase

However, to enable a consistent DEMO construction decision in time, the TBM Program must cover the best combination of design options that are considered to be the most promising candidate for the blanket to effectively minimize the main technical risks for DEMO. Thus, the results of the TBM Program, both during the phases of development, qualification and procurement and the following phase of testing, are expected to provide important input.

It is nevertheless clear that risks and gaps will remain after ITER and, therefore, a sound and complementary

R&D Program for DEMO to address long time performance at higher neutron fluence and high reliability is needed. In particular, vigorous materials irradiation in the limited number of existing fission material research reactors and ultimately in a DEMO-Oriented Neutron Source like IFMIF-DONES [5] is required together with the likely construction of a limited number of dedicated non-nuclear blanket test facilities (or upgrade of the existing ones) for testing integrated multi-effect blanket behaviour.

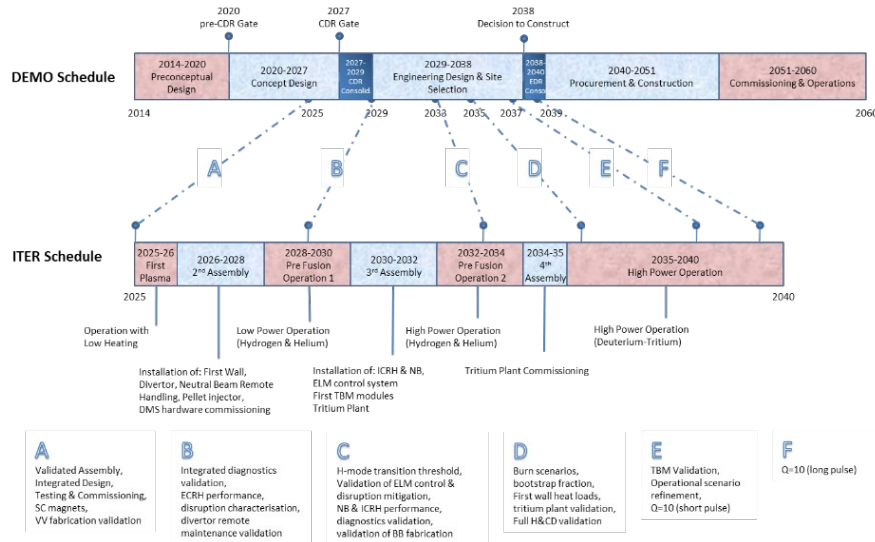


Fig. 1 - Overview of phasing and key technical inputs from ITER DEMO Schedule

3. DEMO Breeding Blanket Design Approach

3.1 DEMO design constraints and integration issues

In DEMO, the breeding blanket must perform a number of essential functions: (i) first, it must absorb the largest (~80%) part of the fusion energy transported by neutrons from the plasma and deposited volumetrically in the surrounding in-vessel structures. The remaining part (~20%) of the fusion power (fusion alpha particles) with the addition of the auxiliary heating power (~100MW) constitutes the so called “power exhaust”, and is deposited as surface heat onto the plasma-facing-components (PFCs), i.e., first wall (integrated in the front-side of the blanket), limiters and divertors. Taking into account the exothermal heat produced by nuclear reactions (about 1.2-1.3 energy multiplication factor depending on the neutron multiplier materials adopted in the breeding blanket), in a reactor of about 2 GW of fusion power, the blanket system has to remove about 1500 MW of nuclear power. Conversion of this energy at adequate thermodynamic efficiencies requires that the coolants are at high temperature and pressure. This has a strong influence on reactor engineering. (ii) Second, it must breed sufficient amount of tritium by capturing fusion neutrons in lithium-bearing materials (in solid or liquid form). Just as an

example, a 2 GW fusion power DEMO is expected to consume around 111 kg of tritium per full power year (fpy), and this clearly underscore the indispensable requirement for the breeding blanket to produce and enable extraction of the bred tritium to achieve tritium self-sufficiency (i.e., it must produce its own fuel). The implications of the tritium breeding requirements on the design and integration of the tokamak in-vessel components that compete for space usage that is needed for breeding (i.e., divertor, protection limiters, auxiliary heating systems, etc.) are briefly discussed below (see also [6]). In addition, (iii) Third, together with the vacuum vessel, the blanket must effectively contribute to shield various components from nuclear radiation (e.g., superconducting magnets and other equipment outside the reactor). Finally, (iv) the breeding blanket must be designed to enable efficient extraction of tritium and minimize losses of tritium. Considerations in this paper are limited to aspects of design and R&D of the breeding blanket. Further information on the tritium fuel cycle can be found elsewhere [7].

Figure 2 [2] shows: a vertical cross section of the current EU DEMO and the physical interfaces between the blanket and the other systems like vacuum vessel and superconducting coils.. The tritium breeding performance

competes with the shielding performance in space restricted regions such as the mid-section of the inboard region.

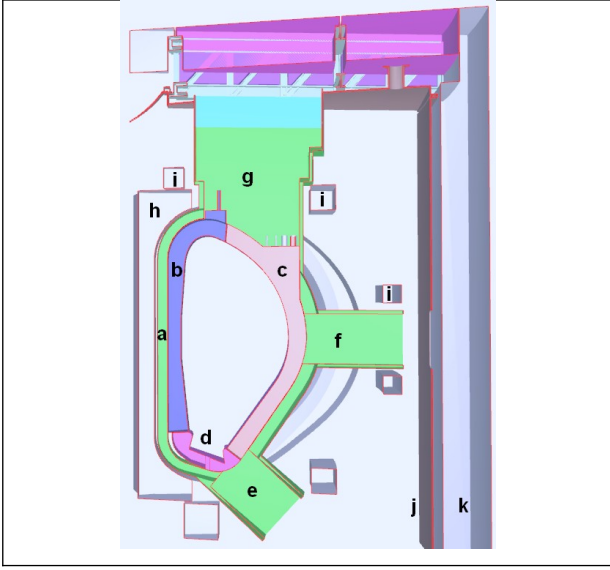


Fig. 2: Elevation view of the tokamak as generated by PROCESS; a) vacuum-vessel; breeding blanket (inboard); c) breeding blanket (outboard); d) divertor; e) lower port; f) (equatorial port); g) upper port; h) toroidal field coils; i) poloidal field coils; j) cryostat; k) bioshield.

The utilization of the space on the inner side of the torus represents a crucial design aspect in tokamak design and deserves some further considerations.

The power density in a tokamak can be written as

$$P_F \propto \beta_t^4 B_{t,o}^4 \quad (1)$$

where β_t (beta) is the plasma kinetic-to-magnetic pressure ratio and $B_{t,o}$ is the toroidal field strength at the centre of the plasma. By increasing $B_{t,o}$ and/ or β_t one clearly obtains a significant increase in power output. However, β_t is limited by plasma stability and $B_{t,o}$ is limited by technological constraints on the maximum practical magnetic field, $B_{t,m}$, at the magnet windings that is limited by technological constraints e.g., ~ 13 T for Nb₃Sn. The relationship between $B_{t,m}$ and $B_{t,o}$ (see Fig. 3) [8,9] is given by:

$$B_{t,o} = \left(1 - \frac{1}{A} - \frac{\Delta_{pw} + \Delta_{BS}^i}{R} \right) B_{t,m} \quad (2)$$

where A is the aspect ratio (R/a) (typically in DEMO $A \sim 2.5-3.5$), R is the major radius of the plasma, and $\Delta_{p,w}$ is the clearance between the plasma and the first wall (in DEMO ~ 0.2 m). The parameter Δ_{BS}^i is the thickness of the region occupied at the inboard by the breeding blanket and the vacuum vessel and includes also maintenance clearance and the thermal shield (i.e, the distance in mid-plane from the first wall to the TF coil windings). The cost of the TF coils, typically, increases as $B_{t,m}^2$ and the maximum practical value of $B_{t,m}$.

Eqs. (2) clearly shows that by reducing Δ_{BS}^i , for a given $B_{t,m}$, one can increase the value of the toroidal field

strength at the center of the plasma, and, thus, the reactor power, or for a given reactor power can reduce the machine size (i.e., R).

Similarly, the flux core radius, r_{OH} , for the OH coils is given by (see Fig. 3):

$$r_{OH} = R - (a + \Delta_{pw}) - \Delta_{BS}^i - \Delta_m^i - \Delta_{OH}^{\square} \quad (3)$$

where Δ_m^i is the thickness of the inner TF coil leg and its support structures Δ_{OH}^{\square} is the thickness of the OH support cylinder. For a given R , a , $\Delta_{p,w}$, $B_{t,o}$, and P_F , reducing Δ_{BS}^i reduces also $B_{t,m}$ and Δ_m^i and r_{OH} increases. Increasing r_{OH} reduces the ohmic heating field, B_{OH} ($B_{OH} \sim I/r_{OH}^2$). Besides the technological constraints on B_{OH} , the cost of the OH coils, and more importantly the cost of the power supply increases rapidly with B_{OH} .

All these factors, provide a strong incentive to reduce Δ_{BS}^i . However, satisfying the energy conversion and tritium breeding requirements in the blanket and providing the radiation attenuation in the blanket/shield necessary for magnet protection favours a relatively large Δ_{BS}^i .

In the current DEMO design the utilization of the space on the inner side of the torus and the required fractional coverage the breeding blanket to achieve tritium self sufficiency has been set on the basis extensive neutronics calculations [10,11,12] to define the basic geometric configuration, in particular, the radial reactor build. The main adopted design guidelines and criteria are described in Table 1 [13].

Because of the numerous penetrations (see Fig. 2) neutron streaming across penetrations on the outboard represents also a serious design issue. A biological shield is necessary to reduce the radiation biological dose outside the reactor to the maximum permissible dose for occupationally exposed individuals. It is conceived that the walls of the reactor building can serve the dual purpose of providing the necessary containment as well as biological shielding.

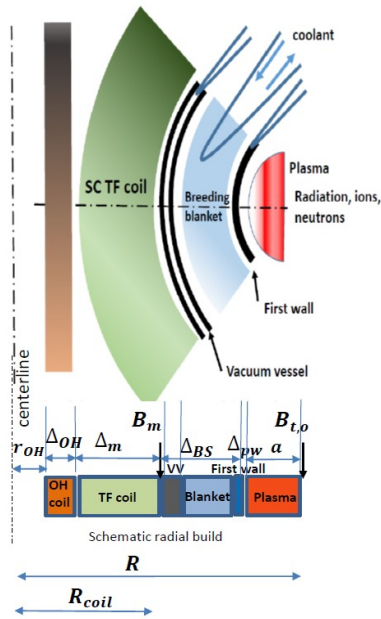


Fig. 3 – Schematic showing the radial build at the inboard

On the basis of the results of design effort conducted to date, it was found that about 1.40 m (60 cm vacuum vessel and ~80 cm inboard breeding blanket thickness was found to provide sufficient shielding, both in terms of material damage and vacuum vessel nuclear heating. To ensure tritium self-sufficiency, thin PFCs are required and, in addition, the number of penetrations must be minimised (about 85% of the plasma must be covered by the breeding blanket) and there are constraints on the space occupied by the divertor [14]. It should be noted that the definition of the effective radial build depends on the specific type of blanket design. For example, a blanket design with water as coolant (i.e., the water-cooled lithium lead concept (WCLL) is expected to shield better than the case of helium (moderation of water and reduced streaming in the manifolds/header) but to breed worse [15,16]. On the contrary, a design concept based on He cooling that use Be multiplier (i.e., the helium-cooled pebble bed (HCPB) concept) can achieve better TBR values and, therefore, can be thinner from the breeding efficiency point of view, but the n-shielding performance is worse. Analyses are in progress to calculate the radial build of a water-cooled and helium-cooled concepts aiming at determining the optimum thickness from the standpoint of tritium breeding, n-shielding and minimization of activation of the surrounding vacuum vessel [17, 17].

The modularity of the blankets is given by the magnet structures (i.e., the number of toroidal field coils), which, in the current design configuration, leaves 16 toroidal interspaces to give access to the blankets for remote maintenance purposes from the top of the machine [18]. Each of the 16 blanket sectors is divided into three segments at the outboard and two segments at the inboard. The central segment has to be removed before the lateral segments can be withdrawn out of the magnet bores. The

estimated average neutron wall load in DEMO is ~ 1 MW/m². Based on current operation considerations, up to 30,000 pulses (as in ITER) with a burn-time per cycle of 2 h each (much longer than ITER) are required to attain a total cumulative limiting fluence of 7 MW a/m² during the machine lifetime, which corresponds to 70 dpa in EUROFER steel components of the plasma near structures (table 2) [2].

Table 1 Breeding and shielding requirements and design targets used in the present design phase of DEMO.

Tritium breeding	<p>A TBR ≥ 1.05 requires:</p> <ul style="list-style-type: none"> • thin PFCs • limited penetrations, e.g., about 85% of the plasma must be covered by the breeding blanket • constraints on occupied divertor space
n-shielding	<ul style="list-style-type: none"> • Max displacement damage in VV (2.75 dpa) • Cutting/re-welding location in In-Vessel Component (IVC) cooling pipes helium production 1 appm
	<ul style="list-style-type: none"> • Total neutron fluence to epoxy insulator 10²²/m² • Fast neutron fluence to the Nb³Sn 10²²/m² • Neutron fluence to Cu stabilizer between TF coil warm ups 1–2·10²¹/m² • Volumetric nuclear heating in winding pack 50 W/m³
	<ul style="list-style-type: none"> • Port interspace: Shutdown dose rate 12 days after shutdown ~ 500 μSv/h (target) • Port cells (occasional access) 100 μSv/h (target) • In-cryostat area, 100 μSv/h (target) • Tokamak building areas beyond port cells requiring frequent access, Shutdown dose rate 1 day after shutdown 10 μSv/h (target)

Table 2 Current EU DEMO design assumptions

Main design assumptions
– P _{fus} ~ 2000 MW ~ 500 MW _e
– Pulses: 2 h
– Single-null water cooled divertor; PFC armour: W
– Low Temperature Super Conducting magnets Nb ₃ Sn (grading)
– 16 TF coils; B _{max} conductor ~ 12 T
– EUROFER for IVCS, AISI ITER-grade 316 for VV
– In-vessel RH: vertical (blanket)/ horizontal (divertor)
– DEMO plant lifetime (design) ~ 7-8 fpy
– Neutron wall loading (average) ~ 1 MW/m ²
– Thermal conversion efficiency > 30%
– Tritium fuel cycle: self sufficient
– Blanket lifetime
• Starter blanket: 20 dpa
• Second blanket: 50 dpa
– Reactor availability: a scenario is assumed in which the availability of a DEMO plant during its initial years of operation (starter blanket) is relatively low and increases (in stage II) to about 30% or more.

A more detailed description can be found in [2].

3.2 A progressive approach for blanket operation in DEMO

The current line of thinking is to adopt for the breeding blanket in DEMO a progressive licensing, where approval for operation up to moderate damage and activation could be obtained for the “starter” blanket, while high-dose engineering data for a more advanced materials blanket is being generated.

It is currently foreseen that DEMO will utilise a “starter” blanket with a 20 dpa damage limit in the first-wall steel (EUROFER) and conservative design margins and then switch to a second set of blankets with a 50 dpa

damage limit with an optimized design (i.e., with somewhat reduced design margins), and if available, improved structural materials that need to be qualified in advance. Because it is unfeasible to change the Balance of Plant (BoP), the same coolant must be used while switching from the first set to the second set of blankets.

This type of approach has been used for the fuel cladding in fission reactors for many years; by limiting the maximum exposure level of the replaceable cladding to below the regulatory limit, while data for higher exposure operation is generated in test reactors or load test assemblies [18], and in the case of fusion in IFMIF-DONES [6].

It is also currently envisaged that DEMO act as a Component Test Facility for the breeding blanket, i.e., while operating with a near-full coverage “driver” blanket, which must be installed by day-1 to achieve tritium self-sufficiency and to extract the thermal power and convert this in electricity), it must be used to test and validate in a limited number of dedicated segments of more advanced breeding blanket concept(s) having the potential to be deployed in a future Foak FPP. The idea to test advanced blanket concepts in a reactor operating with a conservative breeding blanket design is not new. Early considerations on this aspect were given already in the 80’s (see for example [19, 20]).

Such flexibility and capabilities, however, have to be properly investigated early in the conceptual design phase and formalized as high level requirements, since they have major implications on the plant architecture, and systems requirements. This implies that adequate equipment external to the DEMO basic device (test loops) must also be installed at the beginning, or provision made for its later installation. The design features of the test elements will be as far as possible prototypical of a FPP with however the contingencies that the definition of key parameters will require engineering scaling to allow extrapolation between DEMO and FPP conditions, and that they should be compatible, reliable and safe enough not to jeopardize the operation of the DEMO Plant. The detailed design of the test elements will be done during the conceptual design phase.

The final decision on the type of “driver” blanket cannot be made today, because of the existing performance uncertainties and feasibility concerns even for the most mature design concepts. However, a down-selection is deemed possible by the middle of the next decade, by taking into account design and R&D input obtained not only in the area of blanket and TBM, but safety, materials, BoP, remote maintenance, etc. [21]. This will enable a DEMO plant concept to be coherently designed for a design review by 2027 (see Sect. 2).

3.3 Main breeding blanket design concepts in Europe

Breeder blanket systems are under development since the start of civil fusion investigation in the early 1950’s. In Europe, major design studies have been performed in

1990-1999 under the NET Program [20,21,22], in 2000-2004 under the Power Plant Conceptual Studies [23] and 2005-2007 under the DEMO studies [24]. Also, major comparative studies have been done in the US (see for example [25, 26]). An excellent review of the main technical issues in developing the blanket/first wall and the key R&D needs in non-fusion and fusion facilities on the path to DEMO can be found in [27].

Two breeding blanket concepts had been mainly investigated in Europe up to 2002: a HCPB concept and a WCLL concept; and mainly because of budget limitations, a decision was made in 2003 to narrow down the related R&D and to limit the work on two helium-cooled design concepts: the helium-cooled lithium-lead (HCLL) and the HCPB. In 2008, when the decision of the European TBM concepts to be tested in ITER had to be confirmed, and in the absence of a comprehensive DEMO design study, the choice was to consider for the TBM program the same type of breeding blanket concepts as developed up to that point (i.e. HCLL and HCPB), with the assumption that parallel advanced development in areas of BoP of Nuclear Systems and structural materials were to be expected from Fission Industry and in particular from the development of advanced fission systems (i.e., Gen. IV).

Because of the numerous remaining uncertainties and feasibility concerns, four blanket concepts were originally considered in the DEMO pre-conceptual design phase conducted by EUROfusion in the work package Breeding Blanket (WPBB) since 2014, covering all the possible technologies that are believed promising for a DEMO with a development time compatible with the EU fusion roadmap goal. These included (i) the two Helium cooled concepts (HCPB and HCLL) that were part of the EU ITER TBM Program and that make use of solid and liquid breeder respectively; (ii) a WCLL, which makes use of Lithium Lead as a breeder; and (iv) a dual coolant concept (DCLL) using helium and liquid breeder/coolant. Technical details of these concepts can be found elsewhere [28].

However, the awareness of the importance of the integration aspects, especially those related to the choice of the breeding blanket coolant, which affect the whole DEMO Plant [22], along with the perception of the technology gaps still to be overcome in some areas, have recently motivated a critical re-evaluation of the technical choices for the DEMO breeding blanket and the TBM concepts to be tested by Europe in ITER.

As we said, the choice of the breeding blanket coolant provides a clear example of a design issue that pervasively affects the overall design layout of the nuclear plant, and bear a strong impact on design integration, maintenance, safety because of his interfaces with all key nuclear systems. It is generally agreed that water should be considered as the divertor coolant for a near-term DEMO design as the divertor surface heat flux conditions prove to be beyond present helium power handling capabilities. However, the choice of the breeding blanket coolants is still open. Technical issues influencing the

choice include: (i) thermal power conversion efficiency; (ii) pumping power requirements; (iii) required power handling capabilities of the blanket first-wall; (iv) n-shielding requirements (e.g., reduce the blanket thickness that is critical at the inboard side); (v) achievable tritium breeding ratio; (vi) breeder tritium extraction; (vii) tritium permeation and tritium inventory control and purification; (viii) chemical reactivity, coolant leakages and chronic release; (ix) design integration and feasibility of BoP; and (x) design of safety system like the Vacuum vessel Pressure suppression System (VVPSS) that shall contain and confine the primary coolant in case of in-vessel LOCA keeping the Vacuum Vessel (VV) pressure below the limit presently set to 2 bar (as in ITER). Thus, the breeding blanket cannot be designed and developed as a stand-alone component, but it must incorporate the input and the constraints imposed by the plasma and all other systems of the nuclear plant interfacing the blanket (e.g., primary heat transfer systems, tritium recovery and purification systems, heat exchangers, energy storage and power conversion systems, VVPSS and other safety systems).

4. Revised Programmatic Strategy

4.1 Drivers for a new proposal

The rationale of the new strategy is based on the following main considerations:

- First, in the European Fusion Roadmap emphasis is given to the objective to develop DEMO design solutions that will allow fast deployment of fusion energy based as much as possible on mature technologies. Thus, the capability to develop breeding blanket solutions that can be delivered in the mid-term must be secured now, so as to decrease delays on the demonstration and deployment of fusion power. This intrinsically pushes for conservative solutions with high reliability and design margins.
- Second, the input from the DEMO pre-conceptual design activities, in particular, the importance of the design integration aspects, clearly shows that the selection of the breeding blanket for DEMO must not be solely based on performance criteria of the breeding blanket, but shall account for the interfacing systems, the tokamak integration and the safety approach. Investigating these fully, represents one of the major goals and drives the design and R&D efforts during the DEMO Pre-Conceptual Design Phase [29].
- Third, ITER will be the only opportunity to test relevant concepts for breeding blanket for DEMO. This implies that the TBM design must be carefully defined to confirm and validate the most promising concepts for the DEMO “driver” breeding blanket. Previous design options of the EU TBM (HCPB, and HCLL) were made in the absence of a comprehensive DEMO design study, and assuming that important parallel advanced development in areas of the BoP of nuclear systems and structural materials were to be expected from fission industry and in particular from the

development of advanced fission systems (i.e., Gen. IV), especially in the area of helium-cooling: unfortunately, this progress has proven to be much slower than expected.

- Fourth, as DEMO is foreseen to play the role of a “component test facility” for the breeding blanket [30, 31] its design must incorporate the ability and the flexibility to accommodate for testing at least one type of advanced tritium breeding blanket concept, with the potential to be deployed in a FoaK FPP..

In light of the above, a technical and programmatic assessment of the DEMO breeding blanket program and the EU TBM program, has been made in 2017 to study the feasibility and the technical coherence of a change of the EU TBM program and to identify: (i) the best and most cost-effective strategy including the necessary preparatory R&D activities and, taking into account ITER construction schedule slippage; (ii) the required preparatory work for the operation and scientific exploitation of the EU ITER TBM, including the required output for DEMO; (iii) the consolidation of the overall technical rationale and programmatic needs to validate a technically-coherent and financially-sound program, which harmonizes the DEMO breeding blanket and EU ITER TBM and the associated R&D program. The key recommendations are summarised below and are shown Fig 4. Only technical and scientific considerations are described in this paper. All the financial, administrative and governance aspects have been addressed and approved and are being implemented in a co-funded collaboration program by EUROfusion and its laboratories and the European Union's Joint Undertaking for ITER and the Development of Fusion Energy (F4E).

4.2 Key Recommendations

- Focus should be given from now, in the EUROfusion Work Package Breeding Blanket (WPBB), on the two most promising blanket concepts for DEMO, i.e., the HCPB and the WCLL. Nevertheless, a limited R&D activity should also be maintained on the other concepts, i.e., HCLL (e.g., if both water cooling and the solid breeder turn out unfeasible) and DCLL (as a potentially very attractive long-term option). Work in the latter concepts should be restricted to the aspects of R&D not already covered by HCPB and WCLL, and should not include design integration activities at least in the near term.
- Accordingly, two TBM concepts to be tested by Europe in ITER are a HCPB concept and a WCLL concept, the latter to replace a HCLL.. This strategy, will enable testing both coolants (helium and water) and both breeder materials (PbLi and ceramic/Be) and is perceived to be the best to minimize the technical risks and gaps to arrive, in the time frame foreseen by the EU Fusion Roadmap, to a consolidated design for the driver breeding blanket for DEMO. Because of the tight ITER schedule, which requires the completion of the conceptual design review for the WCLL TBM by 2020, the process to replace the HCLL TBM by a

WCLL TBM has been initiated at the time of writing of this paper.

- The selection of the “driver” breeding blanket and most promising advanced blankets is now impossible because of the existing uncertainties. However, in view of the DEMO schedule (Sect. 2) and interdependency with the ITER schedule, a decision of the DEMO driver breeding blanket is deemed to be possible and should be made at the latest by the first half of the next decade by taking into account design and R&D input obtained not only in the area of blanket, but safety, materials, design integration, Primary Heat Transfer System (PHTS), BoP, remote maintenance etc. This will enable a DEMO plant concept design review by 2027. It should be noted that the choice of the driver blanket now planned in 2024 will not further affect the TBM program.
- The timely and complete execution of the TBM R&D program, consisting of the portfolio of concepts with helium and water described above is mandatory for the validation of the choice of the “driver” blanket before starting DEMO construction. The RoX and technical information foreseen by the EU TBM R&D Program (see table 3) will play an important role in the down-selection of the driver blanket. In addition, the results of the TBM tests during the nuclear phase will allow to validate the interpretative/predictive codes required to confirm design choices for the DEMO breeding blanket. DEMO breeding blanket design will benefit by then also from the result of materials irradiation tests to be achieved on a well suited test bed such as DONES [6].

The expert panel also endorsed some additional important elements of the breeding blanket development strategy that were already adopted in the original roadmap. This includes:

- The role of DEMO as a Component Test Facility for the breeding blanket, as described above.
- The phased-operation strategy to utilise a “starter” blanket with a 20 dpa damage limit in the first-wall steel (EUROFER) and conservative design margins and then switch to a second set of blankets with a 50 dpa damage limit with an optimized design, and if available, improved structural materials.

- An early engagement with a licensing consultant is needed to understand and tackle potential safety implications through design amelioration.

Finally, the Expert Panel has recognized the criticality of the issue of tritium availability for operating fusion power plants after ITER. Based on the results of a study conducted in 2017 [32] and the forecasts of tritium production in Heavy Water Reactors (HWRs) of Canadian Deuterium Uranium (CANDU) type-reactors in countries where tritium extraction is carried out, or planned to be carried out, worst-case scenarios were identified where it would appear that there is insufficient tritium to satisfy the fusion demand after ITER.

Clearly there is a need to better understand and monitor the future availability of tritium and to understand the impact of limited resources on the timeline of DEMO. However, there is essentially very little that the fusion community can do to exert an effect on the supply side, as tritium is a by-product of the operation of these reactors, and not the primary economic incentive. Defence stockpiles of tritium are unlikely ever to be shared, and commercial CANDU operators will not alter their plans just to sell us more tritium. In the short-term is recommended to monitor the production of tritium in HWRs and estimate the commercially available supply. If, at some point in the future, it looks as though the demand for DEMO will exceed the supply from CANDUs, then action would have to be taken. It is likely that production of significant amounts of tritium from a dedicated source would be very expensive and take a long time. The “tritium window” as it was once defined by Paul Rutherford [33] is not open indefinitely. Based on current estimates, we believe it would open until around 2050, after which it closes quite rapidly, unless the future of the CANDU reactor program turns out much more favorably than could presently be expected. By far the most advantageous way to fit fusion development into the tritium window would be to timely construct DEMO after ITER on the presently proposed timetable in Europe. Any program strategy that delays substantially the DEMO step places fusion at risk, by allowing the unique and effectively irreplaceable tritium resource to decay to levels, which may prove insufficient to complete fusion’s technological development.

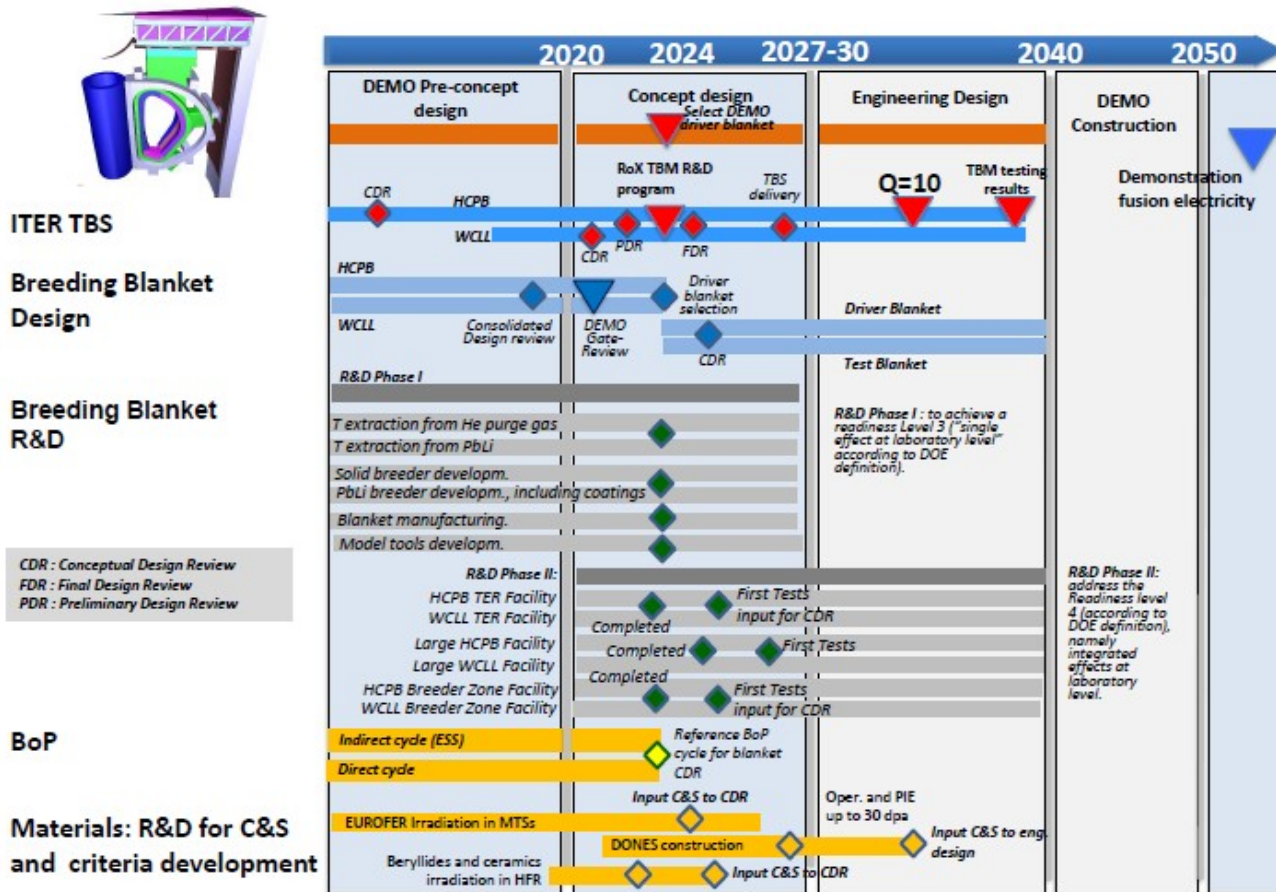


Fig. 4 New R&D strategy proposed to re-align the ITER TBM and DEMO breeding blanket

Table 3 Return of Experience (RoX) and technical information provided by the EU TBM R&D Program

<p>• EUROFER97 development & qualification</p> <ul style="list-style-type: none"> ◇ Database on base material and welded joints, including properties of irradiated EUROFER97 (low dose up to 2 dpa) ◇ Introduction in the RCC-MRx code (DMRx) → strategy adopted by the TBM program ◇ Development of design rules for specific behavior/properties of EUROFER97 (e.g. high-temperature design rules; considering neutron irradiation, creep, low-ductility, (hydrogen) embrittlement issues) ◇ Risk/hazard analysis on EUROFER97 material → Nuclear Particular Material Appraisal (NPMA) ◇ Identification of EUROFER97 material data needed for design and fabrication → gap analyses ◇ Small Specimen Testing Technology (SSTT) methodology development
<p>• Experience from the EUROFER97 finished products fabrication/procurement</p> <ul style="list-style-type: none"> ◇ Melting technology, including alloying elements and impurities control ◇ Quality heat treatment (normalization/quenching and tempering parameters) ◇ Products (plates, bars) acceptance testing → chemical composition, metallurgical and mechanical properties testing ◇ Exploration of various EUROFER97 products fabrication (e.g. plates thickness up to 50-60 mm, bars of various cross-section, tubes)
<p>• Development and qualification of functional materials (i.e., ceramic breeders, beryllium, Pb-16Li alloy)</p> <ul style="list-style-type: none"> ◇ fabrication routes for advanced ceramic breeders, Be/beryllide materials, Pb-16Li eutectic alloy ◇ Li-6 enrichment issue → regulatory & export control aspects, market availability ◇ FM characterization results, including neutron irradiation response, Be/air & steam interactions ◇ Data input for designers (e.g., Material Assessment Report (MAR) & Material Database Report (MDBR))
<p>• Development and qualification of permeation barriers²</p>
<p>• PbLi technology</p> <ul style="list-style-type: none"> ◇ Pb-16Li purification (experimental validation) ◇ Experiments on in-TBM LOCA → validation of thermal-hydraulics prediction (by RELAP5) ◇ Magneto-hydrodynamics (MHD) experimental validation of the predictions ◇ Safety aspects related to impurities limitation (e.g. Bi, Hg, Tl), polonium issue, Pb-16Li/water interaction (pressurization and H₂ generation)
<p>• Tritium technology</p> <ul style="list-style-type: none"> ◇ Tritium extraction from Pb-16Li → technologies experimental validation (gas/liquid contactor, PAV), efficiency

² It should be noted that the decision to implement permeation barriers in the TBM is still pending. They may be required for solving the issue of tritium contamination in the Port Cell & Vertical Shaft for WCLL concept.

- ◇ Tritium Accountancy
- ◇ Molecular sieves technology for tritium extraction from He flow (experimental validation)
- ◇ Validation of Tritium database (Sievert's constant, permeation, diffusivity, etc.)
- ◇ Water detritiation

- **Pressurized helium cooling technology**
- ◇ Thermal-hydraulics of high-pressure helium (prediction & validation)
- ◇ Helium circulator: design, performance R&D in existing EU facilities (e.g., HeFUS, HELOKA)

- **Pressurized water cooling technology**
- ◇ Water chemistry control → e.g. hydrogen dosing like in PWR (CVCS system); EUROFER97 & AISI 316LN steel
- ◇ Generation, transport and removal of activated corrosion products (ACPs)
- ◇ Structural material → EUROFER97 corrosion in PWR conditions; hydrogen embrittlement issue; e.g., stress corrosion cracking
- ◇ Electro-chemical potential control → structural material corrosion
- ◇ Water radiolysis under fusion environment → radiolytical products

- **Instrumentation / sensors development & qualification in fusion environment**
- ◇ Tritium concentration in He, water and Pb-16Li
- ◇ Sensors based on optical fibres, piezoelectric, fission micro-chambers, force sensors for EM loads assessment, sensors for eddy current in ferromagnetic structures
- ◇ Sensors miniaturization
- ◇ Sensors integration in TBM & TBS (e.g. n-activation system, sensors in breeding units, manifold area, PbLi loop, etc.)

- **Predictive tools development & validation**
- ◇ Tritium transport modelling; tritium inventories, tritium permeation, etc.
- ◇ Magneto-hydrodynamics (MHD)
- ◇ Pebble bed thermos-mechanical behaviour
- ◇ Electro-magnetics loads
- ◇ Neutronics assessments
- ◇ Pb-16Li liquid metal corrosion
- ◇ ACP generation, transport and removal (WCLL)
- ◇ Water radiolysis under fusion environment (WCLL)
- ◇ Coupling phenomena & integrated modelling

5. Example of Selected Achievements

Because of the limitation in space, only some brief technical information and main recent achievements related to the two breeding blanket concepts presently considered in Europe for DEMO are briefly summarized here. References to further relevant work published in these proceedings or elsewhere are provided.

5.1 HCPB design and R&D

The current DEMO HCPB design is based on the use of Li_4SiO_4 as tritium breeder material, Be as neutron multiplier and He (inlet 300°C, outlet 500°C, 8 MPa) as coolant. The HCPB blanket system is formed by 18 sectors with 5 segment each (3 outboard and 2 inboard), in which each of them features poloidally 7 blanket modules (see Fig. 5(a)). In particular, each blanket is a sandwich-like structure of parallel, actively cooled cooling plates (CPs) that subdivide the breeder zone in slices of the breeder and multiplier pebble beds (see Fig.5(b) and 5(c)) [34, 35].

A low pressure purge gas (0.2 MPa) of He (carrier gas) with an addition of 0.1% wt. H_2 (doping agent) sweeps both the ceramic breeder and the Be pebble beds, independently. The doping agent is of special importance, as it is the promoter of the isotopic exchange to form HT, which will be carried out of the blanket towards the Tritium Extraction and Removal (TER) system. The purge gas chemistry and flow control is currently an area of intense research. In particular, the possibility to use H_2O instead of H_2 as doping agent would allow the isotopic exchange of the tritium with H_2O to form tritiated water species, instead of HT. Such tritiated water species are non-permeating, which would significantly reduce the tritium permeation in the high-pressure He primary

coolant by orders of magnitude. However, careful analyses are needed to show that such change isn't simply trading one problem for a different problem: historically H_2O is excluded as doping agent because of its incompatibility with pure Be and the associated safety issues; in addition, the increased corrosion of Reduced-Activation Ferritic Martensitic (RAFM) such as EUROFER, that could be caused by the H_2O addition must also be addressed. As a matter of fact these steels contain only ~9%Cr, which is too low to provide robust oxidation resistance.

EUROFER97 is foreseen as blanket structural material, which is especially suited for the use of He coolant at the given temperatures. Due to the moderate heat transfer in the He coolant, the temperature difference between the fluid bulk temperature and the coolant walls is large enough to maintain the EUROFER97 temperature beyond its lower limit (~350 °C) to avoid the Ductile-to-Brittle-Transition-Temperature (DBTT) shift, especially at high dpa regions of the blanket like the first wall. Recent R&D points to the possibility to extend the high temperature operating window of EUROFER97 [36]. Albeit preliminary, these results if confirmed by the R&D program in place, would allow to push the outlet temperature of the coolant up to ~600--650 °C, improving the reactor net efficiency. Such temperature outlet will allow to increase the temperature difference along the reactor core, reducing the coolant mass flow and thus the circulating power, and also in the steam generators / intermediate heat exchangers, reducing the size of these components, thus minimizing the coolant inventory.

Emphasis on the HCPB design concept has been on improving the understanding of the nuclear performance (i.e., tritium breeding and nuclear shielding), as well as on

simplifying the design in order to minimize the amount of steel (i.e. reduced parasitic absorption for better neutron economy). Also, the simpler blanket internals based on a single component (CPs) have reduced the pressure drops from former designs, achieving a plant circulating power of ~130 MW. A comprehensive set of thermomechanical analyses show a correct overall behaviour of the blanket against normal and accidental conditions.

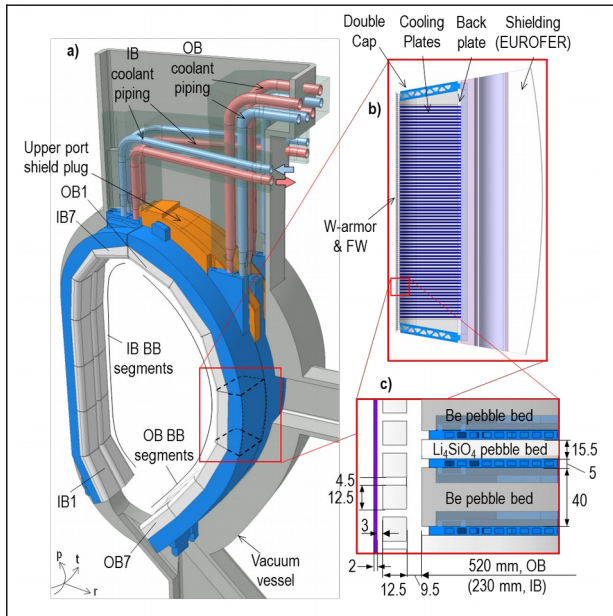


Fig. 5 DEMO HCPB Design: (a) elevation view showing 2 inboard and 3 outboard segments per sector; (b) blanket toroidal cross-section, showing the breeding (front) and shielding (back) regions; and (c) detail showing ceramic breeder and Be multiplier elements in the blanket breeding region.

Despite the aforementioned progress, some key issues still need to be solved. In particular, the current focus on the use of mature technologies for DEMO, especially for the BoP, poses a strict restriction on the upper limit for the plant circulating power for a He-cooled DEMO. On the other side, the reduced operational temperature of Be (<650°C in order to avoid excessive swelling and thus risking the pebble's integrity) poses a safety risk of excessive tritium inventory at the blanket End-of-Life due to the still large tritium retention in Be (40%) at 600 °C [35]. Also, a concept architecture based on cooling plates results in the presence of numerous cooling channels, making a further reduction of pressure drops difficult and negatively affecting in the blanket reliability.

For these reasons, alternative design concepts, which potentially could minimize these issues are explored. This research has led to an enhanced HCPB design [37, 38] based on single-module segments with a hexagonal arrangement of fuel-breeder pins. This enhanced HCPB concept uses Be₁₂Ti as neutron multiplier, which allows to work at significantly larger temperatures in this material, mitigating the tritium release issues. Moreover, the use of pins greatly simplifies the blanket internals and reduces the number of cooling channels about an order of magnitude, allowing a significant reduction of plant

circulating power (about the half of the current sandwich design), as well as improving the blanket reliability. Analyses show the advanced maturity of this design [39,40], fulfilling basic nuclear, thermohydraulic and thermomechanical requirements.

Due to some inherent issues of Be (toxicity, limited resources, industrialization difficulties and high production costs), additional research has been conducted to find back-up alternatives to Be/Be-alloys as neutron multiplier. Starting from the same fuel-breeder pin architecture, an alternative solid breeder blanket based on molten lead as neutron multiplier has been developed [39,40]. This alternative has shown to potentially fulfil the basic blanket requirements while eliminating the issues associated with the use of Be.

5.2 WCLL design and R&D

The current DEMO WCLL breeding blanket design is characterized, as the HCPB, by 16 sectors (dictated by the number of TF coils), each including two inboard and three outboard segments. This breeding blanket uses reduced activation ferritic-martensitic steel, EUROFER97, as structural material, Lithium-Lead (PbLi) as breeder, neutron multiplier and tritium carrier, and water as coolant at Pressurised-Water-Reactor (PWR) conditions: 295 – 328 °C @ 15.5 MPa [41,42].

The design is developed on the basis of the Single Module Segment (SMS) [42] approach (Fig. 6(a)). Each segment consists of an external box, composed by the first wall, the side walls, the bottom and top caps and the Back Plate (BP). Each segment is supported by a Back Supporting Structure (BSS), which connects the breeding blanket to the VV.

The WCLL breeding blanket segments are equipped with internal stiffening plates (Fig. 6(b)), placed along the poloidal-radial and toroidal-radial planes, in order to guarantee that the performances be compliant with the structural design criteria for in-vessel components in normal operation (Level A) and in over-pressurization loading scenario (Level D). For the second condition the design pressure is 15.5MPa+20% [43,44]. The overall segment is formed of breeding units. Radial-toroidal baffle plates, placed between two horizontal stiffening plates, ensures the PbLi circulation in radial-poloidal direction.

The WCLL breeding blanket is cooled by two independent systems: the first-wall and the breeding zone cooling systems [41,45]. They are operated at the nominal pressure of 15.5 MPa and the temperatures 295-328 °C. The first wall cooling system is an integrated part of the blanket segment directly exposed to the plasma, constituted by an EUROFER U-shaped plate 25 mm thick, bended in radial direction. The first-wall plasma facing area is covered by a tungsten layer of 2 mm. The water flows in square channels (7×7 mm) in counter-current directions. The integrated system can safely remove the maximum average plasma heat flux (1.17 MW/m²) and

power nuclear heat depositions of DEMO, delivering coolant to the first-wall PHTS heat exchanger at the design temperature [45].

The breeding zone cooling system is in charge of removing the power deposited in the PbLi and EUROFER structures. It relies on a Double-Wall Tubes (DWT) technology to reduce the probability of “in-box-LOCA” occurrence [41,42]. The reference design 2017 foresees 4 U-shaped DWT, installed in one breeding unit (Fig. 6(b)). This configuration simplifies the manufacturing procedure and allows a temperature field in the breeding zone symmetric with respect to the poloidal plane passing for the center of the segment. Moreover, the thermal gradient in radial direction of the structure is limited, being the minimum and the maximum temperatures calculated with ANSYS CFX 18.1 equal to 300°C and 410°C, respectively.

The PbLi enters in the breeding unit from the bottom of the breeding cell (which is formed by the breeding units of the segment at the same poloidal elevation), flows in radial direction, from the back plate to the first-wall and, then, recirculates towards the back plate, as shown in Fig. 6(c), with an average velocity lower than 0.1 mm/s. The liquid metal hydraulic path in the breeding zone was studied considering the effects of the magnetic field that include, but are not limited to, flow pattern variation, turbulence suppression, and additional MHD-related pressure drops [43].

The water manifolds are integrated in the region between the BSS and the BP. The inlet and outlet of the breeding zone system are defined in a way that the symmetry is ensured and number of structures is minimized. The PbLi manifolds are constituted by two gaps of 40 mm, delimited by three plates of 30 mm. The PbLi is distributed (collected) in the six channels of the breeding units through holes (see Fig. 6.d).

The BSS is based on a plate 100mm thick with ribs welded in the convex part connecting the breeding zone back plate. The thermo-mechanical performances of the BSS and the SMS were assessed in normal operation conditions and in case of central major disruption loading scenarios, by means of the ABAQUS v. 6.14 commercial FEM code (steady state analysis). The modelling approach and the analyses addressed both the segment structure stand alone and the overall 20° sector including three outboard and two inboard segments and the VV attachment scheme. The preliminary verifications, according to ITER Structural Design Criteria for In-Vessel Components (SDC-IC), were satisfied [42,45].

Recent studies have been devoted to analyse a number of design alternatives to optimize the breeding blanket internals, the PbLi distribution, draining capability and minimize the associated MHD issues [46]: the goal is to explore a large number of options and minimize the technical risks of the current WCLL configuration. In parallel a number of key R&D activities are carried out. This includes primarily the study of accidental scenarios

with water/steam ingress into the PbLi [47] and the development of anti-corrosion/anti-permeation barriers, to minimize the corrosion of the EUROFER surface in contact with the flowing PbLi and minimize the tritium permeation through the structural material and consequent capture under the form of HTO in the primary coolant. Progress has been made in the production and characterization (also under irradiation) of Al₂O₃ coating developed by Pulsed Laser Deposition (PLD) and Atomic Layer Deposition (ALD) [48]. In addition, work is progressing on the development of TER technologies for the PbLi. The solution envisaged for DEMO is the Permatator Against Vacuum (PAV) and a dedicated R&D program is in place to qualify the components at a proper level of maturity (note that PAV is not used in ITER TBMs) [49].

Computational tools developed for studying the PbLi flow under magneto-hydrodynamic (MHD) flows [50] are being validated with properly designed experiments. In particular, a dedicated experiment is being conducted in the Magnetohydrodynamic PbLi Experiment (MAPLE) at the University of California Los Angeles (UCLA) [51] through a EU/ US collaboration. A team of experts from Europe and UCLA has recently completed the upgrade of the facility, which is equipped with a rotating magnet and will be dedicated to study the mixed-convection phenomena of volumetrically heated liquid metals flowing in variable magnetic fields [52]. First experiments on MHD mixed convection will be started soon.

The WCLL breeding blanket coolant system is integrated with the water coolant PHTS. Connecting pipelines are routed through the upper ports. The largest inlet/outlet coolant pipelines are DN-200 and are connected with the outer segment breeding zone system. The PbLi loops feed the breeding blanket from the lower ports and receive the breeder from the upper ports. Current pipes are DN-200, based on fluid velocity, but analyses are in progress to quantify the pressure drop induced by the magnetic field.



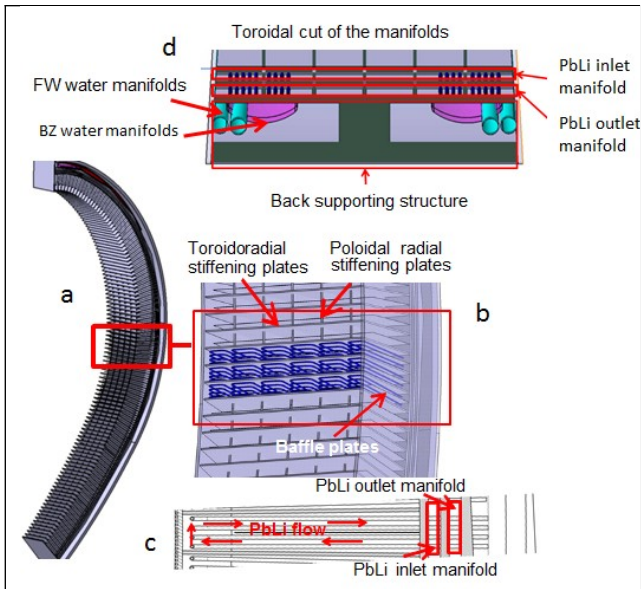


Fig. 6 WCLL BB SMS and BZ stiffening approach, cooling tubes, PbLi flow path and calculated EUROFER T field in outer central segment equatorial zone

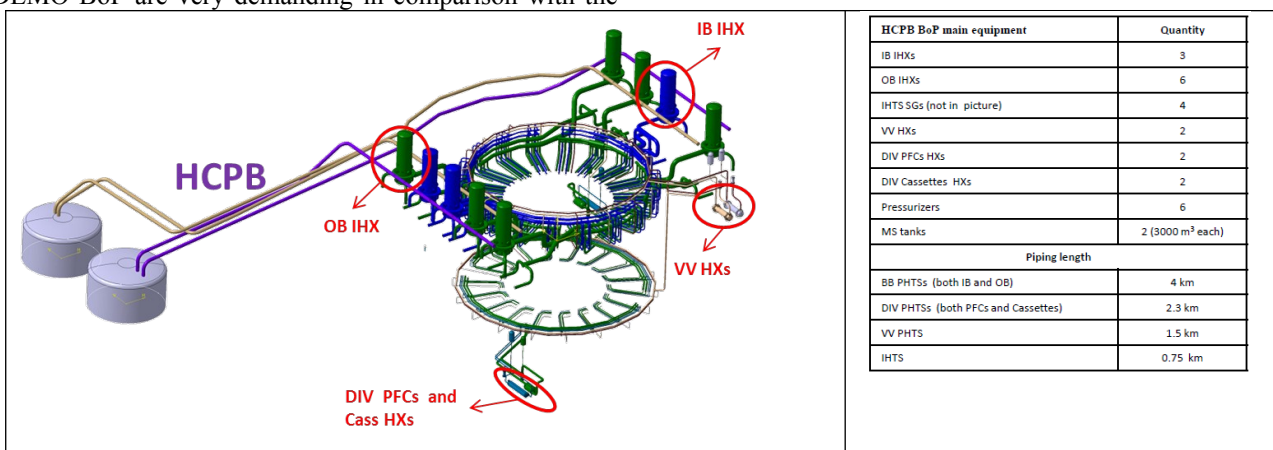
5.3 Balance of Plant

As it was mentioned above, considerations related to the characteristics of the BoP play an important role in the selection of the ‘driver’ breeding blanket. Emphasis at this early design phase has been on few important aspects of BoP, particularly the PHTS [53, 54] and the relevant PCS [55] (the complex system’s “chain” devoted to the extraction of the plasma generated pulsed thermal power and its conversion into electricity delivered to the grid) and the electrical power plant that requires early attention and a continuous reanalysis because of their technical complexity and strong impact on design integration, maintenance, safety [56] because of his interfaces with all key nuclear systems. The main issues for the blanket coolants have been described in [2] together with results of preliminary design work. The requirements of the DEMO BoP are very demanding in comparison with the

similar systems of a fission power plant (NPP). Different cooling fluids, different temperatures and pressures, pulsed operation, represent significant challenges to the design of the heat transfer and conversion system as well as the very huge and in part pulsed electrical power requested by the different electrical loads necessary for the fusion reactor (several times bigger of the electrical power requested in a nuclear or conventional power plant) [57]. Any effort to reduce the complexity of of the DEMO BoP, through simplification and a rationalization of the design and operation of the main reactor systems are expected to have a beneficial returns on the design of BoP systems, on the safety and on the operation of the plant and ultimately of the costs.

Work is ongoing with a strong support of relevant industry, for both options of helium and water as coolants of the breeding blanket to advance the design of PHTSs, Intermediate Heat Transfer System (IHTS) and PCS and to assess the readiness of the technologies postulated for a plant that operate with an Energy Storage System (ESS) [58, 59, 60, 61, 62]. Fig. 7 shows the layout and summarises the main parameters for the case with helium and water, respectively. Such work is useful to: (i) assess dimensions of main components (e.g. HEX, circulators/ pumps, pipes, collectors); (ii) identify technical feasibility issues; (iii) understand commercial availability and R&D needs; and (iv) establish layout requirements and evaluate integration implications with other systems.

An attractive alternative design option is being investigated providing a more direct coupling of the PHTS to PCS with a small ESS. Main features of this new concept are described in [63]. Only about 10% of nominal flow is used by the steam turbine during the dwell period and amuch smaller storage of molten salt (HITEC) is required.



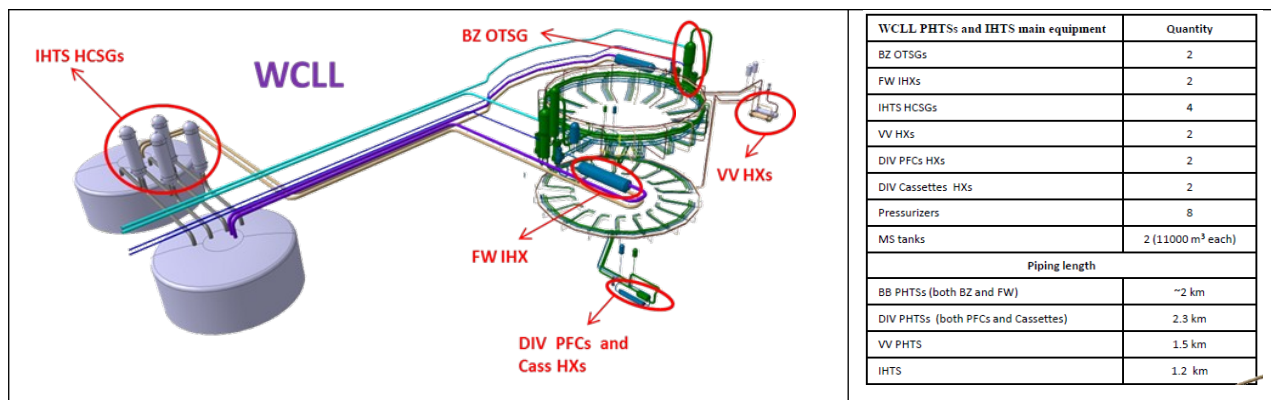


Fig. 7 Layout of PHTS and IHTS for HCPB and WCLL

6. Concluding remarks

The performance and reliability of breeding blanket systems represent foremost considerations in the successful development and deployment any future fusion devices using a DT fuel cycle after ITER. DEMO or any other future nuclear fusion device after ITER would need from the very beginning of operation a full coverage breeding blanket system able to produce and recover reliably its own fuel. Despite its criticality to the penetration of fusion in the electricity market, no breeding blanket has ever been built or tested. Hence, its integrated functions and reliability represent a very challenging development. This paper described the recently revised design and development strategy of the DEMO breeding blanket in Europe. This has been defined taking into account: 1) the ultimate objective of the EU Fusion Program to demonstrate, and subsequently deploy, fusion electricity to the grid as soon as is feasible, with a target date for demonstration around the middle of this century; 2) the input from the DEMO pre-conceptual design activities, which has shed light on the importance of integration aspects associated to the breeding blanket an the interfacing BoP systems, often neglected in the past; 3) the findings and recommendations of a thorough technical and programmatic assessment of the breeding blanket program and the EU ITER TBM program, recently conducted by an independent expert panel.

The main recommendations are discussed in this paper and consists of concentrating from now on HCPB and WCLL that represent the two design concepts to be developed for DEMO and to be tested in ITER, as they will cover most of the aspects to be explored. This implies the necessity for Europe to replace the HCLL TBM with a WCLL one. Nevertheless, as risk mitigation, a limited activity has to be maintained also on the other concepts, i.e., HCLL and DCLL, where work should be restricted to R&D aspects not already covered by HCPB and WCLL.

The testing of the TBM represents a critical step toward validating the principles and technologies of T self-sufficiency. However, even with a successful

exploitation of the TBM program in ITER, gaps are expected to remain to validate the DEMO blanket design, particularly due to the difference in the neutron loads, and a parallel vigorous R&D is required especially on neutron material irradiation. But without the testing experience and knowledge gained from the full deployment of the ITER TBM program, the risk of proceeding to a prototype device like DEMO requiring T breeding is unacceptably high. Thus, the maximum technological risk minimization on a consistent and timely decision on the DEMO construction is obtained if the TBM program covers a wide enough combination of coolants, breeding materials and technologies design options that are considered attractive breeding blanket design options for DEMO.

In addition, it is important to recognize the importance of the gradual increase of the involvement of industry in the design and monitoring process from the early stage to ensure that early attention is given to industrial feasibility, costs, nuclear safety and licensing aspects, and the strengthening of international collaboration to better exploit synergies and minimize duplications.

Acknowledgements

Most of the strategy aspects described in this paper are the result of an effort that has involved a large number of individuals. The authors wishes to thank and acknowledge, on behalf of the EUROfusion Consortium and Fusion for Energy, the outstanding effort of the member of the independent expert panel composed of M. Gasparotto (Chair), H. Ait Abderrahim, G. De Antoni, W. Korn, E. Salpietro, F. Wagner, S. Zinkle. We are also very thankful to A. Grosman (CEA), K. Hesch (KIT), A. Ibarra (CIEMAT), A. Pizzuto (ENEA), M. Shannon (CCFE), G. Veres (Wigner), L. Vala (CV Rez), D. Maisonnier (EU Commission DG RTD), L. Petrizzi (EU Commission DG ENER) for supporting the work of the expert panel and endorsing their recommendations, as well as for setting the foundation to address important management and governance aspects that support this proposal and ultimately will contribute to the efficient implementation of the program. The quality of their work

and the important recommendations that they have provided has played an important role in reshaping and strengthening the DEMO breeding blanket / TBM Programs in Europe.

This work has been carried out within the framework of the EUROfusion Consortium and has received funding from the EURATOM research and training programme 2014-2018 under grant agreement No 633053. The views and opinions expressed herein do not necessarily reflect those of the European Commission.

References

1 [F. Romanelli, Fusion Electricity, A roadmap to the realization of fusion energy, European Fusion Development Agreement, EFDA – Nov.
2012 - ISBN 978-3-00-040720-).

2 [G. Federici et al., DEMO Design Activity in Europe: Progress and Updates, Fus. Eng. Des. (2018) in press.

3 [L.M. Giancarli et al., Fus. Eng. Des. 87 (2012) 395-402.

4 [M. Zmitko; N. Thomas, M. Simon-Perret, A. Li Puma, L. Forest, Laurent3; et al., Current achievements and future perspectives in the
European ITER Test Blanket Modules fabrication technologies development, this conference

5 [A. Ibarra1, M. Pérez, F. Arbeiter, D. Bernardi, A. García, et al., The European approach to the fusion-like neutron source: The IFMIF-
DONES Project, 27th IAEA Fusion Energy Conference (FEC 2018), 22 – 27 Oct 2018. Ahmedabad, India. Event code: 54171 (CN-258).

6 [F. Cismondi, P. Agostinetti, G. Aiello, J. Aubert, C. Bachmann et al., Progress in EU Breeding Blanket design and integration, Fus. Eng. Des.
124 (2017) 562-566.

7 [C. Day, B. Butler, B. Ploekcl, T. Giegerich, A smart architecture for the DEMO fuel cycle, this conference.

8 [M.A. Abdou, Radiation considerations for superconducting fusion magnets, J. Nucl. Mat. 72 (1978) 147-167.

9 [M.A. Abdou, Important aspects of radiation shielding for fusion tokamak reactors, Proceedings of the 5th International Conference on
Reactor Shielding, Knoxville, Tennessee, USA. April 18-23, 1977, edited by R.W. Roussin, L.S. Abbott, D.E. Bartine, Radiation Shielding
Information Center, Oak Ridge National Laboratory, published by Science Press Princeton].

10 [U. Fischer et al, Neutronics design analyses of fusion power reactors based on a novel integral approach, Fus. Eng. Des. 84 (2009) 323-328.

11 [U. Fischer, C. Bachmann, J.C. Jaboulay, F. Moro, R. Villari, Neutronic performance issues of the breeding blanket options for the European
DEMO fusion power plant, Fus. Eng. Des. 109-111 (2016) 1458-1463.

12 [P. Pereslavtsev, C. Bachmann, U. Fischer, Neutronic analyses of design issues affecting the tritium breeding performance in different DEMO
blanket concepts, Fus. Eng. Des. 109–111 (2016) 1207–1211.

13 [C. Bachmann et al., Overview over DEMO design integration challenges and their impact on component design concepts, Fus. Eng. Des, in
press.

14 [C. Bachmann et al., Issues and strategies for DEMO in-vessel component integration, Fus. Eng. Des. 112 (2016) 527–534.

15 [M. Abdou, the choice of coolant in commercial tokamak power plants, Paper presented at the Fourth ANS Topical Meeting on the
Technology of Controlled Nuclear Fusion, King of Prussia, Pennsylvania, October 14-17, 1980.

16 [M. Gilbert, Waste assessment of the EU DEMO design, Fus. Eng. Des. (2018) in press.

17 [G. Federici, W. Biel, M.R. Gilbert, R. Kemp, N. Taylor and R. Wenninger, European DEMO design strategy and consequences for materials,
Nucl. Fusion 57 (2017) 092002 (26pp).

18 [M. Coleman, N. Sykes, D. Cooper, D. Iglesias, R. Bastow, A. Loving, J. Harman, Concept for a vertical maintenance remote handling system
for multi module blanket segments in DEMO, Fus. Eng. Des. (2014) 2347-2351.

19 [M. Chazalon, W. Daenner, and B. Libin, Blanket Testing in NET, Fusion Engineering and Design 11 (1989) 115-123.

20 [J.E. Vetter, Breeding Blanket Development for NET, Fus. Eng. Des. 11 (1989) 101-114

21 [F. Cismondi, P. Agostinetti, G. Aiello, J. Aubert, C. Bachmann et al., Progress in EU Breeding Blanket design and integration, Fus. Eng. Des.
Fus. Eng. Des., in press, <https://doi.org/10.1016/j.fusengdes.2018.04.009>

22 [E. Proust, L. Anzidei, G. Casini, M. Dalle Donne, L. Giancarli, S. Malang, Breeding blanket for DEMO, Fus. Eng. 22 (1993) 19-33.

23 [D. Maisonnier, D. Campbell, I. Cook, et al., Power plant conceptual studies in Europe, Nucl. Fusion 47 (2007) 1524–1532.

24 [S. Hermsmeyer, L.V. Boccaccini, U. Fischer, C. Köhly, J. Rey, D. Ward, Reactor integration of the helium cooled pebble bed blanket for
DEMO, Fus. Eng. Des. 75-79 (2005) 779-783.

25 [M. Abdou, et al., Blanket Comparison and Selection Study, ANL/FPP/83-1, Argonne National Laboratory, 1983.

26 [D.L. Smith, C. Baker, D. Sze, G. Morgan, M. Abdou, S. Piet, S. Schultz, R. Moir, J. Gordon, Overview of the blanket comparison and
selection study, Fus. Technol. 8 (1985) 10–113.

27 [M. Abdou, N. B. Morley, S. Smolentsev, A. Ying, S. Malang, A. Rowcliffe, M. Ulrickson, Blanket/first wall challenges and required R&D
on the pathway to DEMO, Fus. Eng. Des. 100 (2015) 2–43.

28 [L.V. Boccaccini, et al., Objectives and status of EUROfusion DEMO blanket studies, Fusion Eng. Des. 109–111 (Pt. B) (2016) 1199–1206.

29 [C. Bachmann et al., Critical Design Issues in DEMO and Solution Strategies, to appear in Fus. Eng. Des.

30 [G. Federici, et al., Overview of EU DEMO design and R&D activities, Fus. Eng. Des. 89 (2014) 882.

31 [D. Stork, P. Agostini, J-L. Boutard, D. Buckthorpe, E. Diegele et al. , Developing Structural, High-heat flux and Plasma Facing Materials for
a near-term DEMO Fusion Power Plant: the EU Assessment, J. Nucl. Mat. 455 (2014) 277–291.

32 [M. Kovari, M. Coleman, I. Cristescu, R. Smith, Tritium resources available for fusion reactors in the long term, 2018 Nucl. Fusion 58
026010.

33 [P. Rutherford, The Tritium Window, informal PPPL report (1999).

34 [F. A. Hernández et al., A new HCPB breeding blanket for the EU DEMO: Evolution, rationale and preliminary performances, Fus. Eng. Des.
124 (2017) 882-886.

35 [F. A. Hernández et al., Overview of the HCPB Research Activities in EUROfusion, IEEE T. Plasma Sci. 46(6) (2018) 2247-2261

36 [J. Hoffmann et al., Improvement of reduced activation 9%Cr steels by ausforming, Nucl. Mater. Energy, 6 (2016) 12-17.

37 [F. A. Hernández et al., An enhanced, near-term HCPB configuration as driver blanket for the EU-DEMO, this conf.

38 [G. Zhou, et al., Design of two experimental mock-ups as proof-of-concept and validation test rigs for the enhanced EU DEMO HCPB
blanket, , this conf.

39 [G. Zhou, et al., Progress on thermo-hydraulic and thermo-mechanical performances of a Helium Cooled Molten Lead Ceramic Breeder
blanket as a near-term alternative for EU DEMO, , this conf.

40 [P. Pereslavtsev et al., Nuclear analyses of solid breeder blanket options for DEMO, this conf.

41 [A. Del Nevo, E. Martelli, et. al., WCLL breeding blanket design and integration for DEMO 2015: status and perspectives, Fusion Eng. Des.,
124 (2017), pp. 682-686.

42 [E. Martelli, A. Del Nevo, et. al., Advancements in DEMO WCLL breeding blanket design and integration, Int. J. of Energy Research, 42(1),
2018, pp. 27-52.

43 [A. Tassone, A. Del Nevo, et al., Recent Progress in the WCLL Breeding Blanket Design for the DEMO Fusion Reactor, IEEE Transactions
on Plasma Science, 46 (5), (2018) 1446-1457.

- 44 []P.A. Di Maio, et. al., Structural analysis of the back supporting structure of the DEMO WCLL outboard blanket, Fusion Eng. Des., 124 (2017) 944-947.
- 45 []P.A. Di Maio, et. al., On the thermo-mechanical behaviour of DEMO water-cooled lithium lead equatorial outboard blanket module, Fusion Eng. Des., 124 (2017) 725-729.
- 46 []A. Del Nevo, et al., Recent progress in developing a feasible and integrated conceptual design of the WCLL BB in EUROfusion Project, , this conf.
- 47 []M. Eboli, N. Forgiione, A. Del Nevo, "Implementation of the chemical PbLi/water reaction in the SIMMER code", Fusion Eng. Des. (2016), doi:10.1016/j.fusengdes.2016.02.080.
- 48 []S.-E. Wulf, W. Krauss, J. Konys, Long-term corrosion behavior of Al-based coatings in flowing Pb-15.7Li, produced by electrochemical ECX process, Nuclear Materials and Energy (2018), submitted.
- 49 []M. Utili et al. "Design of a multipurpose laboratory scale apparatus for the investigation of hydrogen isotopes in PbLi and permeation technologies, Fus. Eng. Des. 2012, FUSION-6307
- 50 []L. Buehler and C. Mistrangelo. MHD ow and heat transfer in model geometries for WCLL blankets. Fus. Eng. Des. 122:919-923, 2017.
- 51 []C. Courtessole, S. Smolentsev, T. Sketchley, M. Abdou, MHD PbLi experiments in MaPLE loop at UCLA, Fus. Eng. Des. 109–111, 2016, Pages 1016-1021
- 52 []Experiments and Facilities to Explore Multiple Effects in Liquid Metal Blankets in the US/UCLA- EUROfusion Collaboration, M. Abdou et al., Presentation at the 5th IAEA DEMO Workshop, Daejeon, S. Korea, May 7-11, 2018.
- 53 []E. Martelli et al., Study of EU DEMO WCLL Breeding Blanket and Primary Heat Transfer System Integration, Fusion Eng Des, 2018, in press, <https://doi.org/10.1016/j.fusengdes.2018.04.016>
- 54 []I. Moscato, et al., Preliminary Design of EU DEMO Helium-Cooled Breeding Blanket Primary Heat Transfer System, Fusion Eng Des, 2018, in press, <https://doi.org/10.1016/j.fusengdes.2018.05.058>
- 55 []L. Barucca et al, Status of EU DEMO heat transport and power conversion systems, Fusion Eng Des, 2018, in press, <https://doi.org/10.1016/j.fusengdes.2018.05.057>
- 56 []C. Gliss et al., Initial layout of DEMO buildings and configuration of the main plant systems, Fusion Eng Des, 2018, in press, <https://doi.org/10.1016/j.fusengdes.2018.02.101>
- 57 []S. Ciattaglia et al., The European DEMO fusion reactor: Design status and challenges from balance of plant point of view, Proceedings of 17th IEEE International Conference on Environment and Electrical Engineering and 1st IEEE Industrial and Commercial Power Systems Europe, 2017, [10.1109/EEEIC.2017.7977853](https://doi.org/10.1109/EEEIC.2017.7977853).
- 58 []L. Barucca et al., EU DEMO Heat Transport and Power Conversion System design: options and status, this conference.
- 59 []S. Ciattaglia et al., EU DEMO Safety and Balance of Plant design and operating requirements. Issues and possible solutions, this conference.
- 60 []A. Tarallo et al., Preliminary implementation of EU-DEMO primary heat transfer systems for HCPB breeding blanket option, this conference.
- 61 []E. Martelli et al., Thermal-hydraulic modeling and analyses of the water-cooled EU DEMO using RELAP5 system code, this conference.
- 62 []I. Moscato et al., Progress in the design development of EU DEMO Helium-Cooled Pebble Bed Primary Heat Transfer System, this conference.