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Issues and strategies for DEMO in-vessel component integration

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In the frame of the EUROfusion Consortium activities were launched in 2014 to develop a concept of a DEMO reactor including a large R&D program and the integrated design of the tokamak systems. The integration of the in-vessel components (IVCs) must accommodate numerous constraints imposed by their operating environment, the requirements for precise alignment, high performance, reliability, and remote maintainability. This makes the development of any feasible design a major challenge. Although DEMO is defined to be a one-of-a-kind device there needs to be in addition to the development of the IVC design solutions a remarkable emphasis on the optimization of these solutions already at the conceptual level. Their design has a significant impact on the machine layout, complexity, and performance. This paper identifies design and technology limitations of IVCs, their consequences on the integration principles, and introduces strategies currently considered in the DEMO tokamak design approach.

Keywords: DEMO, tokamak, in-vessel components, breeding blanket, design integration

1. Introduction

The EU fusion roadmap Horizon 2020 [1] views a Demonstration Fusion Power Reactor (DEMO) to follow ITER as the remaining crucial step towards the exploitation of fusion power. It advocates a pragmatic approach considering a pulsed tokamak based on mature technologies and reliable regimes of operation, extrapolated as far as possible from the ITER experience. It foresees the development of a conceptual DEMO design by 2020. The three most important requirements for DEMO are defined in [1] as:

1. Achieve tritium self-sufficiency.
2. Demonstrate the production of net electricity.
3. Demonstrate all technologies required for the construction of a commercial fusion power plant, including an adequate level of availability.

Some rationales of these and consequences on the tokamak configuration are described in [2] and [3].

The conceptual design of DEMO must include amongst others a solution for (i) the vacuum vessel supporting the IVCs, (ii) the problem of the heat exhaust on the divertor and the first wall (FW), and (iii) for a breeding blanket achieving tritium self-sufficiency. For risk mitigation four blanket concepts based on different technologies are being developed in parallel throughout the conceptual phase. These are either based on ceramic pebble beds or liquid metal (LiPb) to breed tritium and are water-, helium-, or dual (helium-LiPb) cooled [4].

2. Design approach - vacuum vessel and shield

2.1 Vacuum vessel

a) Functions

The vacuum vessel (VV) confines the radioactive inventory, ensures vacuum, and supports the IVCs. Furthermore it shares two functions with the IVCs: It serves as neutron radiation shield for the protection of the superconducting coils and is the main contributor to the passive plasma stability.

b) Design and operating condition

The VV is a fully welded toroidally continuous double-wall structure made of a conventional austenitic stainless steel: 316L(N). As in ITER neutron shielding plates are stacked in the interspace between its inner and outer shells. The binuclear heating in the vessel is removed by water serving also as moderator. In order to avoid regular vessel baking cycles at 200°C (as required for the ITER VV operated at 70°C) and to reduce thermal expansion relative to the IVCs, the DEMO VV is operated at 200°C at a coolant pressure of 3.15 MPa. The neutron irradiation limit of the DEMO vessel is chosen based on the limit for the negligible irradiation damage applicable to the VV austenitic steel defined in RCC-MRx AFCEN Edition 2012 as 2.75 dpa.

c) Inboard wall design limitation

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In order to prevent damage to the superconductor in case a quench in a TF coil is detected the coil current is rapidly reduced using a dump resistor. Consequently a poloidal current is induced in the vessel, I_{pol}, the magnitude of which is inversely proportional to the TF coil discharge time constant, τ_{CQ}, since the current decay is slow with respect to the vessel time constant (~1s). I_{pol} reaches its peak in the initial phase of the TF coil fast discharge (TFCFD) when the toroidal field is still very strong. The consequent Lorentz forces B_{tor} x I_{pol} cause a pressure load on the vessel that is strongest on the inboard where the toroidal field is strongest. The options to increase the vessel strength are very much limited for the following two reasons: (i) when increasing the thickness of the vessel shells the current induced in the vessel I_{pol} increases roughly proportionally; hence the stress level remains unaffected. (ii) The pressure causes a hoop stress in the vessel inboard wall, a loading for which the (circular) vessel structure is already optimized. Consequently, a limit has been defined in DEMO for the minimum TF coil discharge time ($\tau_{CQ} \geq 28$s), which in turn required the conductor's copper fraction to be increased.

2.2 Shield

a) Elimination of the in-vessel shield

Previous international DEMO/fusion power plant studies, [6], [7], and [8], considered a semi-permanent in-vessel shield: a shell structure within the vessel providing support to the IVCs and neutron shielding to vessel and superconducting coils, actively cooled to a temperature similar to that of the IVCs, see Fig. 1. The impact on the tokamak design complexity of implementing a shield is significant since it must be a toroidally continuous structure in order to support the IVCs against the significant in- and out-of-plane EM loads. As it would be an in-vessel component its design would need to allow for removal and therefore separation into segments. The required mechanical joints would impose significant technological challenges and are currently considered impractical in DEMO.

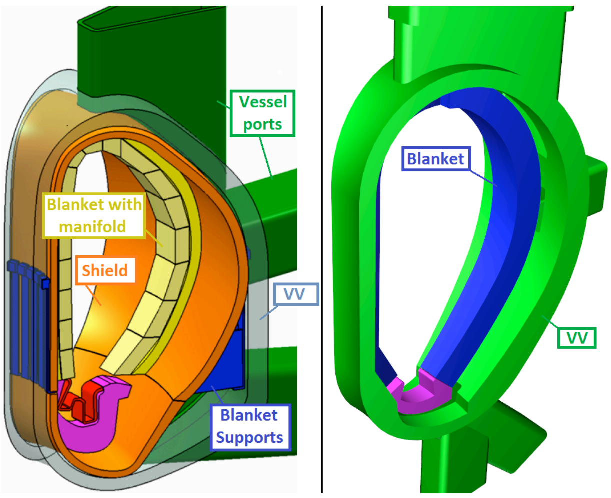

Fig. 1 2014 DEMO configuration with shield (left) and 2015 configuration without shield (right)

At the same time the vacuum vessel provides the required nuclear shielding of the superconducting coils and due to its robust and toroidally continuous design is well suited to support the IVCs. Hence no shield is integrated inside the DEMO plasma chamber and the IVCs are – as in ITER – attached directly to the vessel. The drawback is that the radiation loads to the vessel are mitigated only by the IVCs, see also section 3.3.

b) Neutron shielding of the vacuum vessel

The vessel is shielded by the divertor and the blanket. Presently the DEMO divertor cassette is a water-cooled steel box as in ITER, which efficiently shields the VV from neutron radiation. The breeding blankets show poorer shielding performance compared to the divertor as they are designed to minimize neutron absorption to allow for a high Tritium breeding ratio (TBR) [4] and do not contain sufficient efficient neutron moderators.

Neutron transport assessments [9] indeed found the nuclear heating of the vessel inner shell behind the inboard blanket about one order of magnitude higher than in ITER, where a shielding (not a breeding) blanket will be installed. A corresponding thermal-structural assessment found thermal stresses exceeding the allowable [10]. Consequently a reduction of the vessel inner shell thickness was recommended to reduce the (volumetric) nuclear heat load. This initial result indicates that during the design development of the DEMO vessel the hydraulic conditions providing efficient cooling of the inner shell will play a more important role as compared to the ITER vessel.

The dpa damage in the vessel inner shell is predicted for the different blanket concepts to be below ~0.2 dpa/full power year (fpy) [11]. Hence the dpa damage of the VV is close to the limit of 2.75 dpa at the end of the envisaged DEMO lifetime (~6 fpy).

Whereas a reduced activation steel (Eurofer) was selected as the structural material of the IVCs, the neutron induced radioactivity in the vessel material (SS-316) produces medium-long life radioactive nuclides, mainly due to the nickel, cobalt, molybdenum and niobium content. The contribution of the VV to the overall DEMO radioactive waste is therefore significant, in particular more than 100 years after the end of operation when the activity of Eurofer has decayed significantly [12].

All three issues associated with the nuclear shielding provided by the breeding blanket: (i) the nuclear heating, (ii) the material damage, and (iii) the activation are critical to the currently considered configuration without shield. The nuclear shielding performance of the breeding blanket must therefore be carefully monitored during the conceptual design phase.

3. Design integration of in-vessel components

3.1 In-vessel components functions

<u>Provide a plasma-facing surface:</u> The surface facing the plasma is composed of the divertor plasma-facing

components (PFCs) and the first wall (FW) of the blanket. This surface is exposed to heat loads due high neutron flux as well as radiation and particle influx. The latter applies in particular to the divertor targets that intersect the scrape-off layer. All PFCs therefore need to be actively cooled and designed for efficient heat removal, be made of irradiation resistant materials, and be armored (in DEMO with tungsten). In areas with particularly high heat loads or erosion rates plasma limiters might complement the FW.

Breed tritium: All tritium required to fuel the plasma must be bred within the IVCs. Currently, tritium is bred only in the blanket. No breeding function is presently implemented in the DEMO divertor to allow for simpler design and integration solutions.

Exhaust heat to allow for efficient energy conversion: Almost all heat in a fusion reactor (>95%) is generated in the IVCs, see Fig. 6. In order to allow for efficient energy conversion into electricity the coolant temperature must be high, >300°C, and the outlet temperature well controlled. For improved efficiency the coolant temperature should be increased further, the cooling channel hydraulics be optimized to minimize the required pumping power, and the temperature rise in the IVC should be increased to reduce the required mass flow to remove the heat.

Provide neutron shielding: The IVCs must provide sufficient shielding of the VV, see section 2.2. Radiation protection of the superconducting coils is ensured by the VV.

Contribute to passive vertical stability: In their current configuration the IVCs contribute only little to the plasma vertical stability due to their toroidal segmentation, see also section 3.7.

3.2 Configuration to ensure tritium self-sufficiency

The blanket is the only system breeding tritium. The design of a breeding blanket aims at the arrangement of breeder and multiplier materials in proximity of the plasma in order to maximize tritium generation. Parasitic losses of neutrons being absorbed by structural materials need to be minimized. The FW protects the breeding zone from plasma heat and particle influx but also absorbs neutrons back-scattered from the breeding zone reducing the tritium breeding rate. The FW design has therefore been optimized towards a minimization of its content of neutron absorbing materials, i.e. steel and tungsten. The major fraction of parasitic neutron absorptions occurs in the steel structure of the breeder modules. In current designs ~11-15 vol% of the breeding zone are occupied by steel structures, which deteriorate the breeding performance.

In a dedicated study the Tritium breeding performance potential was investigated regarding the contribution of individual poloidal sections to the TBR (Fig. 2), [13]. This study has concluded in the DEMO divertor to be reduced to a minimum, eliminating the buffer region present in the ITER divertor, see Fig. 4. It shows the potential to achieve tritium self-sufficiency even in a double-null (DN) configuration with a 2nd divertor in the top of the machine. Studies regarding physics issues associated with a DN configuration are underway [14].

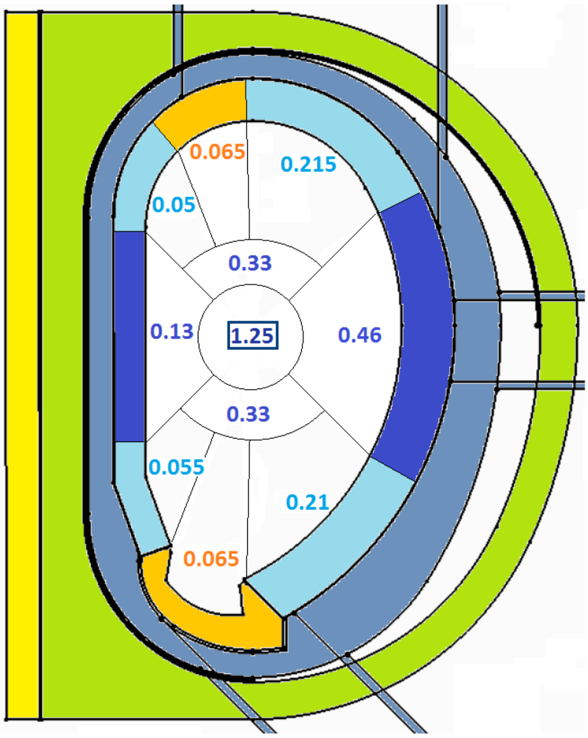

Fig. 2 Tritium breeding ratio potential in four 90° poloidal sections and subdivisions thereof based on DEMO-typical tritium-breeding IVCs and not considering any loss of breeding volume due to the integration of port plugs.

3.3 Size of in-vessel components

a) Blanket

It is in general desired to reduce the radial thickness of the blanket as this would have a number of advantages, particularly regarding remote maintenance, plasma stability (see section 3.6), tokamak size, EM loads, and investment cost. The size of the blanket is mainly due to:

(i) The size required for the breeding zone: The TBR must be larger than 1.10 to ensure tritium self-sufficiency for DEMO**Error! Reference source not found.**. The TBR is the ratio of tritons generated in the blanket and the neutrons generated in the plasma. The achievable TBR is dependent on the percentage of the plasma covered by the breeding blanket and the blanket tritium breeding performance. Depending on the blanket concept a breeding zone of typically 40-80cm thickness is needed to achieve the required TBR [13].

(ii) The size required for the blanket manifold feeding the FW and the breeding units, which is similar to that required for the breeding zone.

Based on these studies the average radial thickness of the DEMO breeding blanket should therefore be chosen to be ~around 1m.

Since the toroidal magnetic field decreases radially with 1/R a configuration with a slimmer inboard blanket enhances the plasma performance as it allows the plasma to be shifted radially into the higher field region. The increase of the plasma power *density* would allow the major radius to be reduced. The beneficial effect is however compensated to some degree by the higher required plasma current and the consequent increase of the solenoid that provides the flux swing. The thickness of the blanket was set in DEMO to about 80 / 130 cm, inboard / outboard, respectively. This configuration was found to be suitable to achieve the required TBR [11]. It was accepted that the increased thickness of the outboard blanket compromises the plasma vertical stability, see section 3.6.

A further reduction of the DEMO inboard blanket thickness assuming no increase of the outboard blanket was studied regarding the favorable effect on the major radius and the unfavorable effect on the TBR, see Fig. 3. This modification could therefore be implemented only in case the TBR reduction could be compensated, e.g. by reducing the divertor and increasing the breeding blanket. Whereas the dimension of the FW was considered as a constant in this study, the blanket manifold was assumed to shrink proportional to the reduction of the breeding zone. The latter assumption might not be conservative and requires substantiation by design. For the current DEMO configuration it was found that a reduction of the blanket thickness Δt_{BLK} allows a reduction of the major radius $\Delta R \sim 1.35\ \Delta t_{BLK}$.

A further reduction of the inboard blanket thickness might require the integration of comparably small amounts of dedicated shielding material, e.g. tungsten carbide, in the back of the blanket.

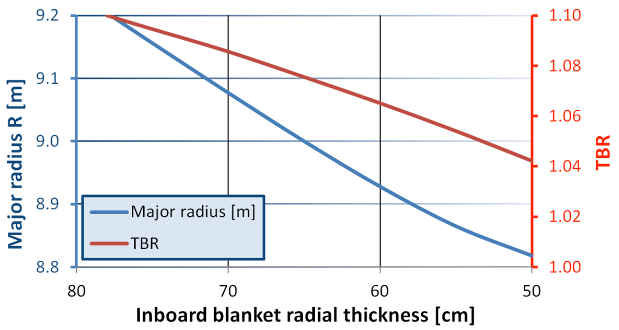

Fig. 3 Effect of the reduction of the inboard blanket radial thickness on the plasma major radius and the TBR assuming an IVC configuration achieving a TBR of 1.1 with an inboard blanket of 78cm thickness.

b) Divertor

The design of the divertor cassette aims mainly at providing the following three functions: (i) support inner and outer targets and the dome, (ii) neutron shielding to the VV, and (iii) facilitate vacuum pumping.

The divertor cassette being a water-cooled (and hence water–filled) steel box with internal ribs and possibly internal shielding plates efficiently shields neutrons. It can therefore be significantly thinner than the breeding blanket. To allow the flow of gases from the divertor into the lower port and towards the vacuum pump a cut-out in the cassette body is required, see Fig. 4. The divertor dome must be designed in order to reduce neutron streaming onto the vessel through this duct. As described in section 3.3 the poloidal extension of the DEMO divertor was reduced extending the blanket up to the vertical target plates. In order to further reduce the size of the divertor the following two approaches offer significant potential:

1. In contrast to ITER it is aimed in DEMO to avoid the integration of separate plasma-facing units onto the cassette body, see Fig. 4. Instead the possibility to join the vertical target PFCs directly to the cassette body is being investigated. This approach would simplify the design and save the space required for the plasma-facing units and their supports. The possibility to replace the PFCs in the hot cell and reuse the cassette body would not necessarily be compromised.

2. The vertical and poloidal size of the divertor is driven by the distance between the x-point and the strike points on the target plates. Currently 1.0 and 1.35m are considered for these distances to the inner and outer target, respectively. A reduction of these distances will however be limited as it impacts on the flux expansion in the scrape-off layer.

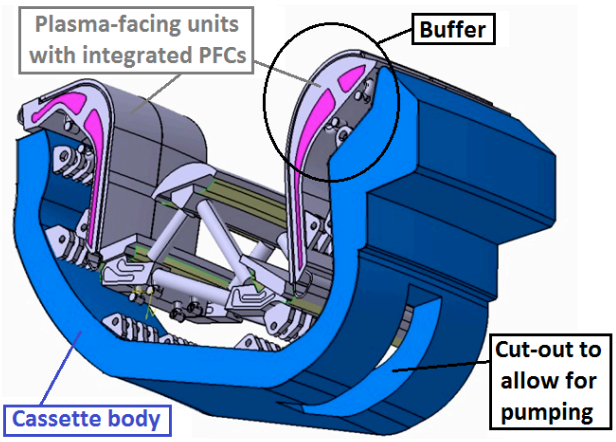

Fig. 4 Previous ITER-like configuration of the DEMO divertor including the buffer region

3.4 Configuration to allow maintenance

The risk of failure of any in-vessel component cannot be excluded due to the design complexity, the numerous and cyclic loads, and the risk of the plasma damaging plasma-facing components. From an investment protection point of view this requires all IVCs to be replaceable or in-situ repairable. The latter is considered impractical due to the high dose environment that not only prevents human access but also imposes limits in the choice and functionality of remote handling technologies.

a) Remote access to in-vessel components

Due to the high energy neutrons produced in D-T fusion plasmas and the high neutron fluence in DEMO helium production and material neutron damage is high in the IVCs. Reliable re-welding of pipes could therefore only be achieved at the backside of the IVCs. In the ITER blanket the feeding pipes are accessed from the front through the FW, which also requires space around the pipes to allow for orbital cutting and joining. Front-side access would be practical also to maintain diagnostic sensors, or to release IVC support structures or electrical straps at the back of IVCs. However, in DEMO the IVCs and the vessel become activated due to the high neutron fluence, and cause a dose rate in the plasma chamber that would severely decrease the lifetimes of viewing and weld inspection tools [15]. In addition front-side access would require access holes penetrating the FW panel. The FW channels in DEMO are however relatively small and their spacing was reduced to a few millimeters to avoid temperature peaks in the FW. The size of any penetrations that could be integrated between two adjacent cooling channels is therefore very limited. Solutions requiring pipe cutting and re-welding operations carried out at the back of the IVCs with front-side access are therefore considered impractical [16].

Instead all IVC pipework is designed for cutting and re-welding either inside the better shielded VV ports or at the back-side of the IVCs using in-bore tools operating in a lower dose environment.

b) Segmentation of in-vessel components

To allow extraction through the vessel ports all IVCs are divided into parts with individual feeding pipes. In order to reduce plant downtime when IVCs require exchange it is in general aimed at dividing the IVCs into large segments or cassettes. The maximum sizes of the ports are however defined by the cage formed by toroidal and poloidal field coils. Furthermore the back of all IVCs should be accessible through at least one port to facilitate cutting and rejoining operations of feeding pipes. These principles have led to the division of the blanket in three outboard and two inboard segments, see Fig. 5 and three divertor cassettes per TF coil. These are vertically handled through large upper ports [16]. This approach had also been adopted in 1988 in the next European Torus (NET), [17].

Due to the segmentation of IVCs disruptions typically cause eddy current loops within single IVCs. The consequent EM loads, e.g. the radial moments, could in principle be reacted amongst the IVCs. The required structural connections between IVCs would however obstruct the kinematics required for their removal during maintenance. Therefore the main supporting function of each IVC is individually provided from the back; and equal opposite forces acting on adjacent IVCs are – inefficiently – transferred through the vessel.

c) Space in the upper port

Much of the toroidal space in the area of the upper port is taken up by the TF coils. In addition a thermal shield must be integrated between the port and the coils with the necessary clearances. Independent of the poloidal location, the sizes of the TF coil, thermal shield, and port wall are constant and towards the inboard increasingly narrow down the remaining space inside the port. In ITER the vessel ports are on the outboard where the distance between the TF coils is large. The DEMO upper port though is rather narrow in particular on its inboard side and space is tight for the removal of the large DEMO blankets. Reducing their toroidal size of the TF coils is however not considered as it would require increasing the 2nd dimension of the winding pack, which on the inboard side is radial and would directly affect the radial build of the machine. In an attempt to enlarge the port internal space the feasibility of implementing a single plate of steel as toroidal port sidewall instead of the standard double-wall structure has been studied. Initial results indicate that a single 60mm steel plate might provide sufficient neutron shielding and be able to withstand EM loads. Without additional reinforcements however it cannot withstand a differential pressure between plasma chamber and cryostat greater than 0.3 bar. The loss of vacuum causing 1 bar in either of the two volumes can however not be avoided due to the possible occurrence of in-cryostat or in-vessel leak incidents, air ingress or during intentional venting. The VV being a confinement barrier the installation of a rupture disc to the cryostat volume is not practical as it would require the cryostat to become confinement barrier as well. It seems therefore likely that the port wall must remain a double-wall structure.

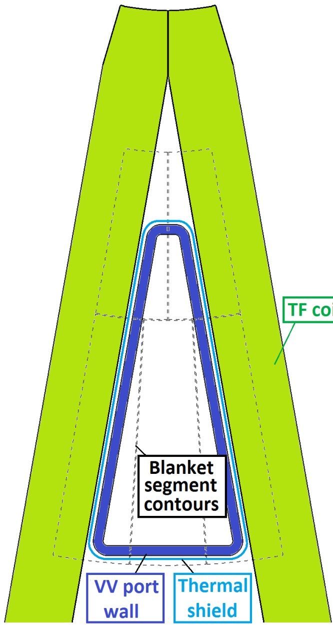

Fig. 5 Horizontal cut through one DEMO upper port with adjacent TF coils limiting the port size and contours of the 3 outboard and 2 inboard blanket segments.

3.5 Configuration for heat exhaust

a) Power distribution in DEMO

Exhausting the power generated in the plasma is the ultimate goal of operating a fusion machine. To allow for efficient conversion of the heat into electricity it is aimed at removing the heat with high temperature coolant. The technologies selected for the IVCs however impose upper limits on the coolant temperature:

Copper alloys soften at high temperatures. CuCrZr, used in ITER, softens at temperatures >300-350°C [18].

Eurofer instead maintains sufficient strength up to >500-550°C [20]. Since large temperature gradients occur in PFCs due to the high heat loads the temperature of their structural material may be ~100°C higher than the coolant, requiring the coolant temperature of copper-based PFCs to be below ~250°C. In contrast a Eurofer-based FW – the standard technology considered in DEMO – allows coolant temperatures of up to 380°C. In the breeding zone the energy density is much lower compared to the FW and the temperature gradients in the cooling channels are therefore low. The coolant of the breeding zone can therefore reach temperatures very close to the material's limit temperature, hence ~500°C.

In case liquid water is chosen as coolant it must be increasingly pressurized by raising temperatures to prevent boiling. This is possible up to 374°C. For the DEMO water-cooled blanket the same operating condition was chosen as in pressurized water reactors (PWR): up to 325°C at a pressure of 15.5 MPa. Gaseous coolants like helium naturally do not have an upper temperature limit but have much lower heat removal efficiency.

The challenging heat loads acting in particular on the PFCs require trade-offs being made between heat exhaust efficiency and heat load capacity. The choice of technologies for the different components is summarized below. These trade-offs must also consider the amount of heat removed by each individual component compared to the total heat generated. Fig. 6 shows one possible scenario for the distribution of the heat to the individual components. For some of the indicated distributions and in particular for the distribution of radiation and particle heat across the PFCs (red boxes) there are significant uncertainties and arbitrary choices have been made. The values provided in Fig. 6 should therefore not be mistaken for the heat load maxima to be considered in the design of the individual components. Despite the uncertainties the figure highlights the primary significance of the breeding zone, the secondary significance of the FW and the moderate importance of the divertor in the overall power balance. Taking into account the extreme challenge imposed by the power flux density at the strike points, [19], it is logical to make least compromises regarding the conversion efficiency of heat exhausted from the breeding zone and accept significant compromises in the cooling of the divertor.

b) Divertor plasma-facing components

In DEMO an ITER-like divertor target technology is being considered as a reference based on a water-cooled tungsten mono-block with a copper alloy heat sink. To enhance the performance and to comply with the DEMO conditions advances of this the target concept are studied making use of advanced materials [21].

This concept was chosen given its superior performance under high heat loads thanks to the high thermal capacity of liquid water and the small temperature gradient through the pipe wall due to the very high thermal conductivity of copper. These PFCs must be operated at a pressure significantly higher than the saturation pressure corresponding to the water temperature. This precaution prevents local water boiling and the resultant loss of heat removal in the event of heat load excursions, e.g. due to loss of detachment [22]. In order to realize a reasonable margin against this burn-out phenomenon and at the same time keep the operational pressure reasonably low the coolant temperature of the divertor PFCs was defined relatively low (150°C). This choice also prevents excessive corrosion [23] and material softening. Advanced composite Cu-alloys are expected not to soften up to higher temperatures. On the downside the relatively low coolant temperature does not allow for an efficient conversion of the removed heat into electricity. Given the relatively small amount of heat removed by the divertor PFCs (~5% of the total power generated in DEMO, see Fig. 6) the poor electricity conversion efficiency was accepted in favor of performance and technology readiness. The loss of ductility of CuCrZr irradiated a low temperature [24] is also a concern. The degraded material properties of Cu-alloys must therefore be considered in the design of the PFC cooling pipe.

c) First wall

The ITER FW is designed for heat loads that are significantly higher than the average nominal heat load during normal plasma operation. The reasons are (i) incentive to retain sufficient operation flexibility being ITER an experimental device, (ii) penalty factors to account for misalignment and non-uniform heat load distribution due to shaping of the FW, and (iii) sufficient margins to accommodate for off-normal thermal transients. As loss of plasma control or internal disturbances can cause the plasma to approach and even touch the FW, leading to high localized thermal transients. Unfortunately, large uncertainties exist on the magnitude of the thermal loads on the FW in ITER and even more in DEMO. In addition, the engineering limits of the DEMO FW based on Eurofer might require the implementation of Cu-alloy based technologies in some areas [25]. The DEMO FW must hence be configured aiming at distributing the heat loads as uniformly as possible across the FW and along each single FW channel to reduce local peaks. The FW should therefore be precisely aligned and its shape follow the plasma as smoothly as possible. Compared to ITER the DEMO FW must fulfill the additional function to efficiently exhaust the heat for electricity production. This requires on one hand the cooling condition at every point of the FW channels to be close to the optimum and therefore - again - a uniform heat load distribution. Moreover possible temporary heat load excursions must be minimized since these would require raising the nominal coolant velocity to increase the heat load the FW can tolerate. This would increase the required pumping and hence recirculating power significantly. Furthermore the nominal FW outlet temperature would be below what is aimed at to enhance electricity conversion efficiency. The distance between FW and plasma was, compared to ITER, increased to 22.5cm to minimize the heat loads in such transients.

Two blanket cooling scheme principles are currently considered, one of FW and breeding zone, see Fig. 7. Their thermohydraulic performances are studied in [26]. In the in-series cooling scheme the mass flow is equal in FW and breeding zone, which makes the thermohydraulic optimization difficult given that the heat loads generated in the breeding zone are about three times higher compared to the FW, see Fig. 6. On the other hand, if the FW is cooled by a separate cooling loop the outlet temperature of that loop is limited, see section 3.5 a). This limitation mainly affects the helium-cooled blankets that aim at outlet temperatures of ~500°C. The temperature of water as a coolant is by default limited to 325°C due to the choice of the PWR operating conditions.

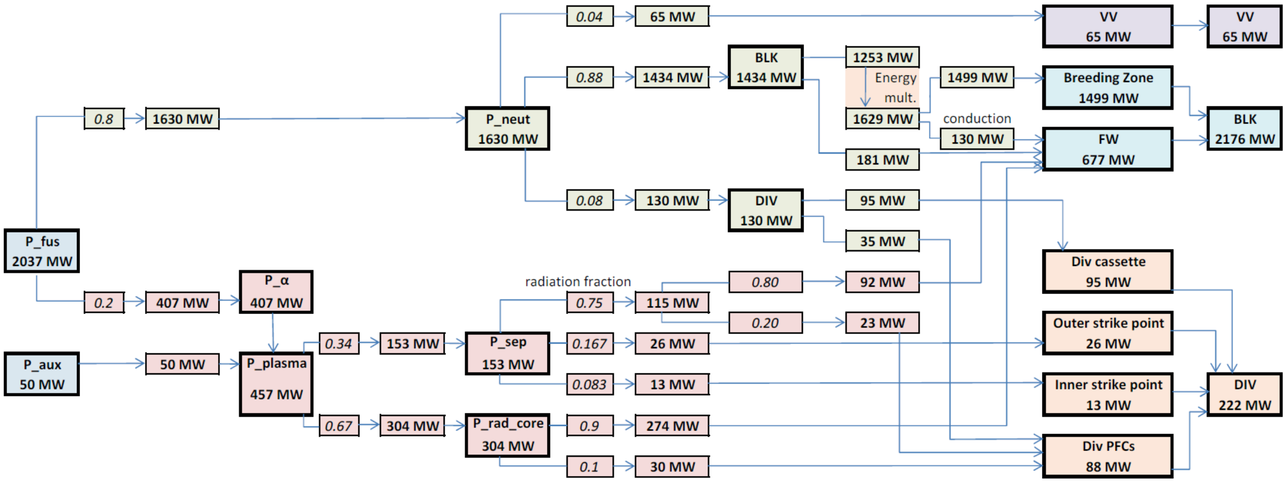

Fig. 6 Power generation during flat-top operation in the DEMO plasma and its division into the three forms by which energy leaves the plasma: as neutrons (green boxes), particles (P_sep), and radiation (P_rad_core). Furthermore: conversion of particle energy into radiation in the divertor (radiation fraction), energy multiplication due to interaction of neutrons with multiplier materials, and possible energy distribution to VV, blanket and FW, divertor cassette and divertor PFCs.

In short: In order to successfully integrate the FW the primary aim must be to achieve a FW surface following the field lines as smoothly as possible with minimized FW alignment tolerances in order to uniformly distribute the heat load. Secondly heat load uncertainties need to be reduced in order to allow reducing the design heat load. This would enable a more efficient operation with higher outlet temperatures and decreased pumping power.

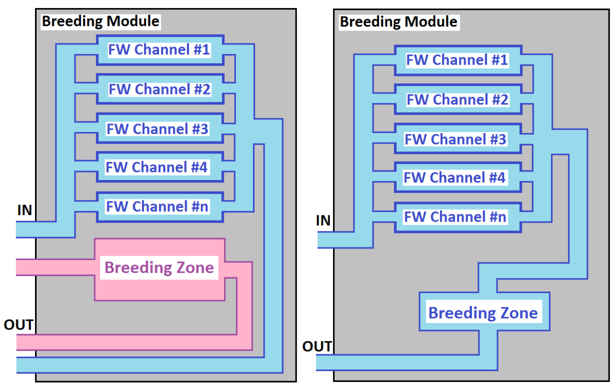

Fig. 7 Breeding blanket cooling scheme with two separate cooling loops for breeding zone and FW (left), and a single cooling loop (right)

3.6 Configuration for passive plasma vertical stability

Good passive stability of the plasma in general terms is achieved by providing a toroidally continuous conductive shell in vicinity of the plasma. Most effective are those parts of such a shell where the radial field provided by the PF coils is large, i.e. on the outboard at both top and bottom. A good passive stability relaxes the requirements of the active control system regarding both the necessary power and the reaction time. A conductive shell is naturally provided by the vessel inner shell, even if locally discontinuous due to the presence of the vessel ports. In a tritium breeding reactor the vessel inner shell is relatively far from the plasma due to the large radial size of the breeding blanket. The consequent poor vertical stability [27] needs to be compensated by one or more of the following options:

i. Reduction of plasma elongation,
ii. Reduction of aspect ratio,
iii. Implementation of toroidally continuous vertical stability coils in the plasma chamber,
iv. Integration of active or passive saddle coils in IVCs,
v. Poloidally continuous first wall panel of the outboard blanket segments, or
vi. Toroidal electrical connections between adjacent IVCs.

It is also recognized but not yet quantified that a double-null configuration might have a much improved vertical stability. The first two options have a significant impact on the tokamak design. In the current single-null configuration the plasma elongation at the surface of 95% poloidal flux, k95, was in fact reduced to 1.59 and the aspect ratio to 3.1 to improve the plasma vertical stability. Both changes however cause a reduction of plasma power density that can only be compensated by an increase of the plasma volume, which require an increase of the tokamak size and cost. The integration of coils in the plasma chamber or IVCs is possible but increases the design complexity.

Toroidal electrical connections between adjacent IVCs have a significant potential as they would enable toroidal currents to be induced in vicinity of the plasma. Toroidal electrical connectors between IVCs are however difficult to realize. One option could be bolted connectors. These would however require disconnection prior to remote maintenance. Access from the front in the narrow gaps between IVCs is considered impractical also due to the high dose rate in the plasma chamber. Access at the back is too limited and unfeasible in areas neither within nor in vicinity of the VV ports. Relying on preloaded springs is difficult since candidate metals seem to undergo severe relaxation under neutron irradiation [28]. In addition such flexible parts could not withstand the high EM loads due to the significant currents running through the connectors. A third option recently considered are connectors expanding hydraulically or pneumatically after installation of the IVCs but a corresponding feasibility study has not yet been undertaken.

4. Summary and outlook

Whereas ITER technologies and design solutions can often be adopted in DEMO, e.g. for vessel or magnet systems, the additional requirements in DEMO to breed tritium and produce electricity require different solutions for IVCs. In addition the high neutron fluence imposes severe limitations on remote maintenance operations where numerous constraints exist due to technology limitations and the design space compliant with the tokamak architecture is narrowed significantly. In many cases the design therefore drives the system requirements.

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