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Critical Design Issues in DEMO and Solution Strategies

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The EU fusion roadmap defines as a goal the development of a DEMO, which achieves a long plasma operation time and demonstrates tritium self-sufficiency and net electricity output. Eight design issues have been identified as critical: (i) feasibility of wall protection limiters during plasma transients, (ii) integrated design of breeding blanket and ancillary systems, (iii) power exhaust taking advantage of advanced divertor configurations, (iv) tokamak architecture based on vertical blanket segments, (v) direct or indirect power conversion concept, (vi) configuration of plant systems in the tokamak building, (vii) feasibility of hydrogen separation in the torus vacuum pump and direct recirculation, and (viii) plasma scenario.

For each of these issues potential solutions have been identified and activity plans have been defined for the associated developments and assessments. Two of these affect in particular the integrated design of the DEMO machine, namely (i) and (iv). These are introduced and discussed in this article providing a summary of identified risks, requirements, and solution concepts. For the remaining six design issues references are provided only due to space limitations in this article.

Keywords: DEMO, tokamak, design integration, plasma-wall interaction, breeding blanket

1. Introduction

The DEMO design development is focused on eight key design integration issues. These were selected for their high impact on plant and tokamak design, safety, and maintainability and all affect in their resolution the design and possibly the technology of several tokamak and plant systems or even the DEMO architecture. A number of design variants are being considered as potential foreseeable solutions. Gate reviews are planned in 2020 to effectively assess the solutions and help evaluate and down-select among options.

The six DEMO key design integration issues not introduced here are described elsewhere:

- Breeding blanket (BB) design and ancillary systems: [1]
- Divertor configuration [2]
- Power conversion concept [3]
- Plant systems and tokamak complex [4]
- Hydrogen separation in the torus vacuum pump [5]
- Plasma scenario [6]

2. Wall protection

2.1 Requirements

Damages to plasma-facing components (PFCs) in DEMO caused by plasma-wall interaction requiring replacement are either caused by (i) damage to the armour, e.g. formation of W-droplets that could detach in future shots causing plasma disruptions, or (ii) damage to the armour <u>and</u> to the heat sink structure below causing a coolant leak. The frequency of required in-vessel component (IVC) replacements and the duration of the replacements must be minimized to achieve satisfactory plant availability and to protect the investment. The present considered concept foresees preventing the plasma to contact the BB first wall (FW) in events that are *expected* (see definition in [7]). Plasma limiters on the other hand where plasma-wall contact is *expected* must be designed for the corresponding heat loads, i.e. the heat sink structure must be designed according to relevant design rules. Damage to the armour requiring IVC replacement must be reduced.

In addition, to allow for nuclear licensing, the predicted number of in-vessel loss of coolant events (in-vessel LOCA) due to plasma-wall contact must be as low as possible and the impact area of leak accidents be kept as local as possible. We distinguish four types of events depending on the reaction of the vacuum vessel pressure suppression system (VVPSS), see Table 1. In a VV ICEIIa safety-credited isolation valves separate the leaking limiter or divertor from its PHTS. The amount of water spilled in the large VV volume causes a pressure below the set pressure of the bleed line valve. In a VV ICEIIb due to a coolant leak in any IVC the bleed line is opened and the VV pressure is suppressed sufficiently in the VVPSS tank avoiding bursting of the VVPSS rupture disks. In ITER the FW break size that will cause the VVPSS rupture disks to burst is defined as 200 cm² [8]. For the DEMO water-cooled blanket operated at higher pressure (155 bar) that break is quantified in the range 26-260cm² [9].

Table 1 In-vessel LOCA events in DEMO

Event	Max. # of events	Spilled water	Break size	Pressure suppression by
VV ICEIIa	85	<~2 tons	<~20 cm²	VV volume
VV ICEIIb	15	>~2 tons	<~20 cm²	Bleed line
VV ICEIII	1	>~2 tons	200 cm²	Rupture disk
VV ICEIV	-	>~2 tons	0.2 m²	Rupture disk

2.2 Plasma transients and consequent wall loads

The concept to protect the wall must consider the following three main causes of plasma-wall contact [10]: 1) Ramp-up/ramp-down, 2) loss of plasma position

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control leading to a disruption, and 3) unplanned H-L transition.

In the initial ramp-up phase and in the late ramp-down phase all particles are exhausted on the plasma-wall contact area before the diverted configuration is established. The energy of these particles is much lower as compared to the diverted configuration.

In the other two cases when the plasma contacts the wall a thermal quench (TQ) will occur releasing in ~4ms the plasma thermal energy to the contact surface (up to 1.3 GJ in H-mode [10], less in L-mode). During the subsequent plasma current quench (CQ) that lasts ~70-200ms [7] the plasma position can change significantly shifting the plasma-wall contact location. During the CQ the plasma magnetic energy (~0.9 GJ [11]) is released partly to the wall contact surface, partly via different mechanisms, e.g. halo currents [12].

Part of the wall protection strategy is the significant reduction of plasma-wall contact events due to H-L transitions by the plasma control system. Realistic controllers, diagnostics and power supply limitations are being considered to evaluate this transition phase. Nonetheless, since the control system is not safety-classified, also the inboard wall must be designed for plasma-wall contact.

2.3 Plasma-facing components for wall protection

In DEMO we have initially considered three types of plasma-facing components making up the FW:

1) A Eurofer or Cu-alloy heat sink cooled - similar to the divertor - by water at ~150°C mounted on an actively cooled Eurofer box structure. These shield blocks are attached to port plugs and installed inside the VV ports or – alternatively – are attached to the breeding blanket back-supporting structure, see Figure 1. In case damage to the cooling structure, the limiter PHTS (not the BB PHTS) would be affected. The amount of spilled coolant can therefore more effectively be reduced by isolation valves, see above. We expect that in most events when limiter PFCs are damaged no more than an estimated mass of ~2 tons of water would enter the VV.

2) The Eurofer-based FW structure that is an integral part of the breeding blanket cooled with a coolant at ~300-380°C [13]. The W-armour of this FW could in principle be increased in thickness [14]. In case of a damage of the cooling structure the BB primary heat transfer system (PHTS) would be affected.

3) A small component, e.g. a U-shape pipe, actively cooled by a gas, extruding the FW shape and acting as a fuse that would be damaged at plasma contact releasing its gas and causing a radiative collapse of the plasma before it can get in contact with the BB FW. This 3rd type of PFC is not considered as frequent replacements are expected to be required while at the same time the maintenance action is expected to be significant since the VV would need to be accessed and opened.

The impact of transient heat loads on different wall components in contact with the plasma is described in [10]. It was found that of existing PFC technologies divertor target-like PFCs could meet the requirement to protect the heat sink structure from failure during plasma-wall contact if the thickness of the tungsten armour was increased to ≥20 mm. The wall protection concept currently developed for DEMO is therefore based on installing limiters with this type of PFCs that extrude the BB FW and protect it from plasma contact.

Limiters, being additional in-vessel components and requiring special maintenance plans, add design complexity, reduce ports available for the installation of auxiliary systems, and reduce the plasma surface covered by the BB and hence the tritium breeding ratio (TBR). They offer however essential advantages: (i) good accessibility for replacement with exception of the inboard limiters, see below, (ii) individual alignment and hence small tolerances with respect to the toroidal field (TF), (iii) separate, non-BB PHTS [17], (iv) through implementation of isolation valves in the limiter port cell more effective limitation of coolant inventory spilled during in-vessel LOCA, and (v) their surface shape can be customized for plasma contact to well distribute the power, i.e. for a near scrape-off layer-like (SOL) power decay length ($\lambda_q \approx$1mm) [18], [19]. Although this might lead to peaked particle heat loads on the limiter surface during flat top, these can be tolerated due to the high heat flux capability of >2 MW/m².

a) Outboard wall protection

Ramp-up: The DEMO plasma is currently assumed to be ramped-up on the equatorial outboard limiters.

Loss of plasma position control: In case of an upward plasma movement, limiters installed in the upper ports will contact the plasma preventing it to contact the BB FW. In case of a downward plasma movement, an increased divertor outer baffle or a lower port limiter of corresponding size are currently considered. Both options are being analyzed to evaluate if it is possible to prevent the plasma to contact the BB FW. It might be necessary to implement additional limiter components between equatorial and lower ports.

All limiters on the outboard wall are being designed to allow their replacement without the need to handle or cut feeding pipes of divertor cassettes or BB segments.

a) Inboard wall protection

Inboard limiters might be:

(a) Replaceable limiter components attached to inboard segments as shown in Figure 3. These would require a front-side RH concept similarly to the ITER FW to cut and re-weld the cooling pipes and to release and re-engage the mechanical supports and electrical connections [20]. Most suitable access seems to be an opposite equatorial port whose limiter would be maintained during the same maintenance phase. Divertor or BB would not be affected.

(b) Inboard blanket segments with special PFC design. The integration of Cu-alloy-based PFCs in an inboard BB will require the regular replacement of these inboard BB segments even if no damage occurs due to the irradiation

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lifetime of Cu-alloy of about 10 dpa, [17]. The replacement of one inboard segment requires the prior removal of at least 2 divertor cassettes and 2 outboard BB segments. Since the corresponding RH sequence is substantial and would be required in intervals of ~1 full power year the relevancy of this concept for future fusion power plant could be questioned.

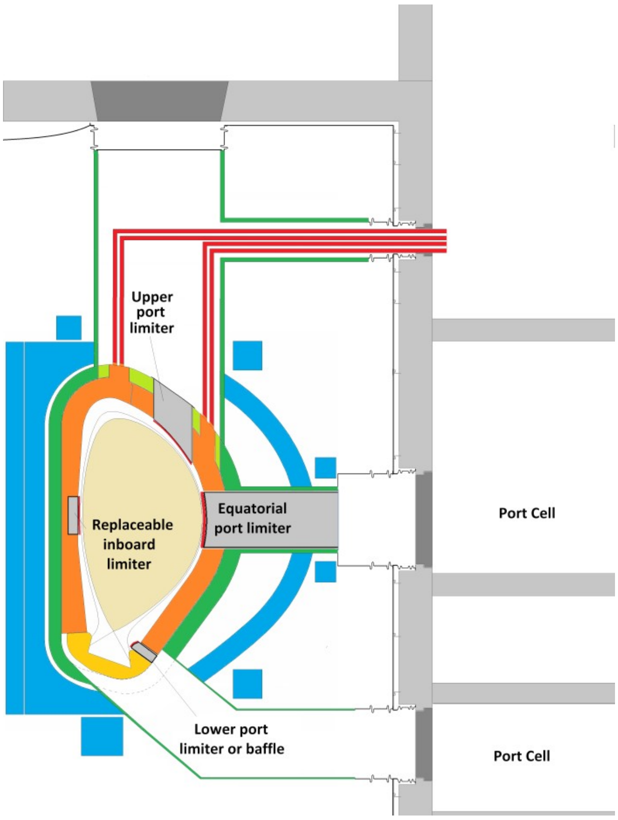

Figure 1 Plasma limiters considered in DEMO

4. Blanket vertical segment architecture

4.1 Issues and solution strategies

In DEMO the BB is segmented into large vertical segments as substantiated in [17]. The first main rationale of this concept is that the related RH tools operate within the VV upper and lower ports rather than inside the main VV chamber.

During the extraction of the BB segments the BB transporter must react some bending moments. These occur because the BB center of gravity (c.o.g.) cannot be directly below the BB handling interface and, also, some segments require during the extraction kinematics to be rotated by few degrees. Considering the large weight of the BB segments (~40-70 tons) these bending moments are moderate when compared to a horizontal extraction concept with cantilevered components since gravity acts vertically. The moderate bending moments at the BB handling interface are second main rationale of the BB vertical segment architecture.

To avoid complex recovery scenarios and in particular to prevent damage to the safety-classified VV the drop of a BB segment must be avoided. It is therefore also required for the BB transporter to sustain the loads occurring during seismic events. In seismic events in addition to vertical also horizontal accelerations occur. These are the most severe loads for the BB transporter. Three types of BB transporter concepts have been considered initially and one based on a hybrid kinematic mechanism has been assessed, [21]. Risks associated to this concept were identified recently and it was decided to study alternative concepts offering potential solutions for the BB handling interface:

1. Splitting the BB segments at the equatorial plane in half to reduce their weights.
2. A new design concept of the BB transporter for the full BB segments customized to the required kinematics.

The potential in this regard of a third solution has been recognized but it is not currently being investigated: lifting the BB segments by one or more cables using an overhead-crane approach. This approach would however require the lifting interface to be above the c.o.g. A counterweight would need to be attached to the inboard and lateral outboard segments during removal. Only vertical loads would be transferred between load and crane, hence the horizontal seismic excitation of the BB transporter would not be transferred to the BB segment. Attaching counter-weights to BB segments might however be found impractical given the space constraints within the upper port and the maintenance hall.

4.2 Split BB segment concept

For the alternative DEMO architecture to study the concept of split BB segments a double-null (DN) divertor configuration was chosen in order to allow the simultaneous study of integrating a DN divertor with a vertical rather than a horizontal port, see Figure 2.

The study of this architecture will assess its main potential caveats, i.e. risks [22]:

1) Lowering of half segments will require 16 lower vertical ports consuming most of the space below VV and magnets currently occupied by magnet feeders. An alternative configuration for the magnet feeders must be identified.

2) A maintenance hall below the machine will require significantly lowering the tokamak complex basemat, which is associated with large costs.

3) RH tools need to be developed which are capable of lowering half BB segments through the lower vertical port into the lower maintenance hall.

4) A concept needs to be developed to support split outboard BB segments not requiring access to all equatorial ports.

5) A RH concept needs to be develops to cut and re-join BB pipes at the equatorial level on the inboard.

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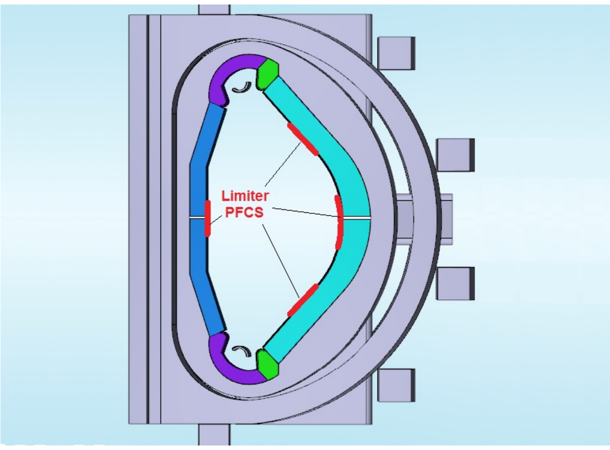

Figure 2 DEMO architecture with split BB segments and double-null configuration

4.3 Full BB segment concept

The full BB segment concept is being revisited aiming at simplifying the BB removal kinematics. In particular rotations of the BB segment will be minimized and the BB transporter concept will be adapted to these. This is expected to allow an increase in the stiffness of the BB transporter for it to better withstand horizontal acceleration of the BB segments.

At the same time the possibility to reduce seismic loads is being investigated. Since the DEMO site has not been identified it was decided to consider the ITER soil response spectra and – as in ITER – the DEMO tokamak complex to be supported on seismic dampers, [7], which strongly reduces the horizontal accelerations of all equipment inside the building. The potential of dampers at the tokamak support, see Figure 3, to further reduce the seismic acceleration of the VV is being studied.

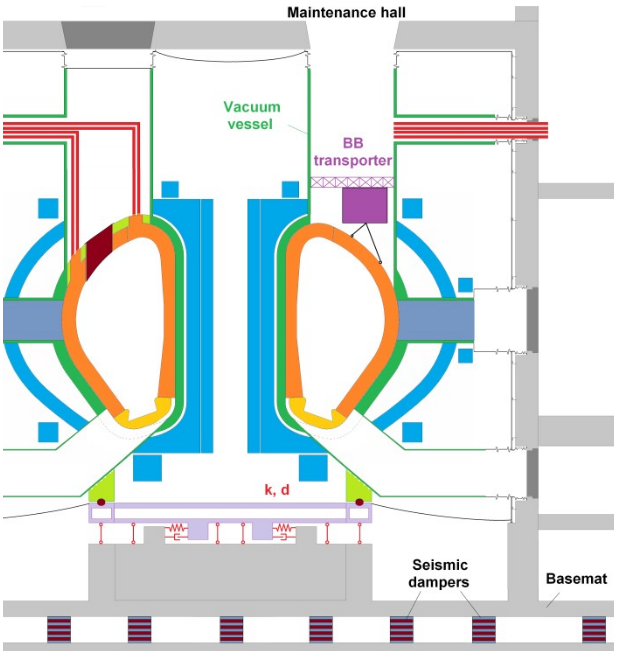

Figure 3 Schematic of structures supporting the BB transporter: VV supported on pedestal ring, possibly with horizontal dampers, and seismic dampers below tokamak complex

5. Conclusions

The eight main DEMO key design integration issues have been identified. The solution strategies and considered requirements of two of these have been described here, corresponding references have been provided for the others.

The wall protection strategy focusses on the integration of plasma limiters with high heat flux PFCs. The aim of the development is the demonstration of a reliable protection of the BB FW from contact with the plasma, the prevention of coolant leaks from the limiter PFCs, and a suitable RH strategy for the limiter components.

Feasibility issues with the blanket transporter were recently identified. An important problem is presented by the requirement to support the BB also during seismic events when horizontal accelerations occur. As part of the strategy horizontal seismic dampers below the tokamak are being investigated to reduce these loads. The strategy to resolve the blanket vertical segment architecture is however two-fold: (i) development of a tokamak architecture with BB segments split at the equatorial plane in half to reduce their weights. (ii) optimize the extraction kinematics of the BB segments and customize the BB transporter design to these aiming at an improved load bearing capability.

All solutions are planned to be evaluated at a Gate review in mid-2020.

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