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Benefits and Challenges of Advanced Divertor Configurations in DEMO

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ABSTRACT

The divertor design and configuration define the power exhaust capabilities of DEMO, and act as a major design driver. They set a number of requirements on the tokamak layout, including port sizes, poloidal field coil positions, and size of toroidal field coils. It also requires a corresponding configuration of plasma-facing components (PFCs) and a remote handling scheme to be able to handle the cassettes and associated in-vessel components the configuration requires.

There is a risk that the baseline ITER-like single-null (SN) divertor configuration cannot meet the PFC technology limits regarding power exhaust while achieving the target plasma performance requirements of DEMO or a future fusion power plant. Alternative magnetic configurations (AMCs) – for example, double-null, snowflake, and super-X – exist and potentially offer mitigation solutions to these risks and a route to achievable power handling in DEMO. However, these options impose significant changes on machine architecture, increase the machine complexity and affect remote handling and plasma physics and so an integrated approach must be taken to assessing the feasibility of these options.

This paper describes work carried out to define a set of design limitations that any alternative divertor configuration for DEMO must consider, and assesses the feasibility of integration of a set of potential AMCs for DEMO.

1. Introduction

EU-DEMO is a proof of concept that aims to demonstrate integrated operation of fusion power plant technology in a representative environment, to show that net electricity production and attractive levels of availability are achievable through validated remote maintenance strategies [1]. Numerous studies into the economics of fusion power (e.g. [2]) have indicated the capital-intensive nature of fusion economics; that is, the initial investment in the plant is the largest driver of the final cost of electricity. One way of reducing this cost is to make the tokamak as small as possible for a target power output. Further significant economic improvements result from ensuring that plant availability is as high as possible. The current concept for DEMO is based around a set of technologies described elsewhere in this special issue, and any proposed structural design modifications to incorporate alternative technologies to expand the available operational space must not only consider the direct changes to the configuration that must be made, but also the impacts on the high-level targets.

One of the principle size-drivers in a tokamak power plant is the performance of the divertor in terms of the power which can be allowed to cross the separatrix, much of which must be radiated away in order to achieve detachment and thus avoid significant erosion rates of the divertor surface, which shorten component lifetime and hence reduce plant availability [3, 4]. The lowest-risk approach is to follow the path laid by ITER and make ITER-like assumptions for physics and technology performance, and this is the approach taken by the EUROfusion baseline design. However there is a risk that the baseline single-null (SN) divertor configuration cannot meet the plasma-facing component (PFC) technology limits regarding power exhaust and first wall protection while achieving the target plasma performance requirements of DEMO or a future fusion power plant or, alternatively, that attempting to do so pushes the plasma into an unacceptably unstable high-radiation scenario. Alternative magnetic configurations – for example double-null (DN), snowflake (SF), X- (XD), and super-X (SX) – exist and potentially offer solutions to these risks. DEMO-like design options have been developed using these configurations to explore the impact on global machine design and divertor performance [5, 6]. By understanding the impacts of incorporating such configurations early in the conceptual

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development process, resources can be focused on the most DEMO-relevant configurations and the overhead of integrating them later – if needed – is reduced. Consideration of divertor integration was one of the eight Key Design Integration Issues (KDIIs) to be studied in the Pre-Concept Design Phase, as outlined in [1].

There are three main issues to be addressed:

1. Do the alternatives create significant design and operational margin for the plasma?

2. Are the engineering and maintenance aspects manageable?

3. Can the plasma and configuration be adequately controlled to avoid damage to the plasma-facing components in the divertor and main chamber?

Much of the analytical work was carried out within the EUROfusion WPDTT1 work package 2014-2020 and has been previously reported [7, 5, 6, 8]. This paper reviews these previous results from the perspective of the DEMO engineering team.

2. DEMO baseline design

The 2017 DEMO baseline design used for this analysis is summarized in Fig. 1. It assumes modest advances on ITER physics and technology, with a target of 500 MW net electrical power and a minimum pulse length of two hours. The divertor challenge quantifier, $P_{sep}B\ /qAR_0$, is constructed by combining the Eich scaling [9] for scrape-off layer width with the tokamak geometry and conducted power loss and represents a measure of the power density on the divertor which is probably recoverable without significant damage should plasma detachment be lost [3]. It represents a measure of the conducted power density, parallel to the magnetic field, leaving the plasma which must be dealt with in the divertor. For baseline DEMO, this limit is scaled from ITER.

Characteristic	Value
R_0 / a (m)	8.9 / 2.9
κ_{95} / δ_{95}	1.65 / 0.33
Fusion power (MW)	2000
Burn time (s)	7200
$\beta_{N,tot}$	2.9
$P_{sep}B/qAR_0$ (MW T m^{-1})	9.2

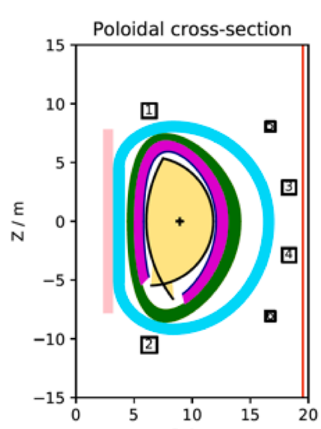

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Fig. 1. Key parameters from the 2017 EU DEMO baseline. The final value is the divertor challenge quantifier (see main text).

This tokamak configuration and plasma were used as the target for the studies on the alternative configurations. The aim was to investigate the additional integration overheads, and potential penalties or gains in device performance from using the alternative configurations.

3. Basic assumptions and requirements

A set of design rules were defined to provide guidance for the development of alternative configurations within the wider technology assumptions used for the baseline DEMO design.

3.1. Technology of plasma-facing components

The plasma-facing components were assumed to be constructed similarly to the DEMO baseline, using water-cooled high-heat flux (HHF) materials with tungsten armour. This implies a maximum steady-state heat load of 10-20 MWm^{-2}, with a maximum incident plasma temperature of 5 eV to prevent erosion [10, 11]. Consideration had to be given to loading on both the inner- and outer-limbs, with the assumption that the loading on the inner limbs should never be higher than that on the outer limbs.

3.2. Configuration of magnets

In order to avoid conflicting with other system requirements, the magnet contours must be set so that

- there is sufficient space for a mid-plane port and upper and lower ports
- the mid-plane port is a minimum of 3 m in poloidal extent, consistent with wider DEMO assumptions (e.g. [12])
- the upper port (for single-null) needs to be able to see all blanket segments for pipe connections and lift, and be large enough to extract blanket segments
- the upper port (for double-null) must be consistent with access requirements for proposed RM approaches, requiring interaction with wider remote maintenance and segmentation analysis [13]
- the lower port must provide access to divertor cassette for direct removal.

In the double-null (DN) configuration vertical removal of upper and lower divertors had to be considered (remote maintenance approaches are discussed in more detail elsewhere in this issue [13]), although alternative strategies could also be considered. The vacuum vessel (VV) ports (vertical and midplane) had to share the outboard poloidal space with the toroidal field (TF) intercoil structures and hence their sizes were limited, affecting the stress-handling of the TF coil structures, particularly for out-of-plane stresses. It should be noted that the adoption of a DN configuration provides a potential opportunity to consider a radial segmentation of the divertor cassette and develop a maintenance strategy to remove only the outboard targets that are much more exposed to power loads than inner targets.

The poloidal field (PF) coils must be supported by the TF coils and the forces on them cannot be arbitrarily large. This means that the forces and stresses in these components are limited, and their positioning must be compatible with access for the removal and replacement of the in-vessel components. The following limits are taken to apply:

$<$ 400 MN force on single PF coils

$<$ 300 MN separation force in the CS

$<$ 660 MPa stress in the TF coil

There is a target flux swing from the central solenoid of 320 Vs.

During the final phase of the analysis, the shapes of the TF magnets considered throughout the earlier stages of the study were found to be unsuitable to withstand the EM forces. The contours were therefore adjusted for further analysis as shown in Fig. 2 below.

This had the following consequences:

- In the SN/DN cases the increased size of the TF coils moved PF coils 2 and 5 moderately further away from the plasma. Vertical plasma control with these coils will most-likely require more power and further analysis will be carried out. If in-vessel coils can be considered for vertical stability (VS) control this would not be a concern, although additional engineering considerations apply to the use of such coils. Increasing the number of TF coils may allow more-optimised (smaller) coil shapes, although with significant impacts on remote maintenance (RM) access through the reduced inter-coil ports.
- Snowflake/SX: These configurations require strong plasma shaping around the lower (active) X-point, meaning high currents in the surrounding PF coils. As there is a non-trivial movement of these coils the achievability and stability of these configurations needs to be recomputed with the new coil shapes. This assessment is still in progress. In this case, increasing the number of TF coils would not necessarily allow more-optimised (smaller) coil shapes, as the limit is imposed in this case by the divertor size.

3.3. Use of in-vessel coils

In-vessel coils, carrying a steady-state current, could be considered for plasma shape control, based on some advances in ITER technology. Up to 400 kA DC per coil (total, not per turn) was considered as an absolute upper limit purely to explore design space but the technical feasibility to integrate coils carrying such currents into DEMO has not been shown. ITER ELM coils, for comparison, have a DC current up to 15kA per turn limit with 6 turns (total 90kA, with zero average current over the duration of the pulse) [15] and are intended for plasma control rather than maintenance of the plasma equilibrium. The ITER coil turns are each 60 × 60mm arranged in a 120 × 180mm pancake – an equivalent 400kA DEMO coil would require ∼24 turns, perhaps in a 420 × 180mm pancake. These coils then must be shielded by the blanket or divertor and supported by the vacuum vessel. They must not obstruct ports or component removal paths. Overall, placement options are somewhat limited. In-vessel coils for equilibrium maintenance are therefore undesirable but it was considered that if they made the difference between an impossible configuration and a possible one they could be considered for the analysis to allow exploration of potential design space. Further specific analysis of the coil engineering requirements will be required if such configurations are identified.

In-vessel coils for divertor sweeping as a reconnection-mitigation measure are discussed in [16]. By designing for these from the outset (as opposed to retro-fit into the design, as with ITER) they can be well-shielded and integrated behind the blanket.

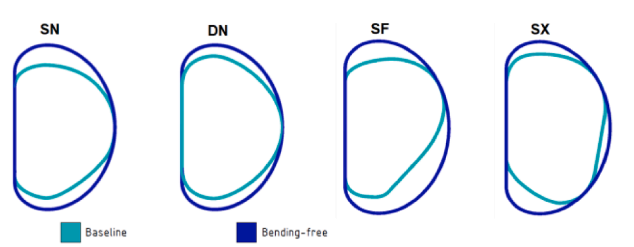

Fig. 2. Comparison of the “baseline” ADC TF coil shapes to symmetrical bending-free shapes. A series of intermediate TF coil shapes was investigated for each option to find the optimal compromise in which the peak stresses were within limits [14]. The “baseline” TF coils are about 18m in height; the “bending-free” are ∼22-24m in height, depending on the space required for the divertor configuration. See [14] for details.

3.4. Plasma control

The strike point positions, and power split between them, must be relatively stable under plasma perturbations to avoid excess power being delivered to places that are not designed to cope with them. This is true for all configurations, even though the number of strike points differs, but is particularly critical in the SF case with four strike points which exhibit different heat loading, but for which the peak heat transfer may rapidly switch between strike points if the configuration is perturbed. Defining accurate load specifications for the divertor components is critical for integration into the DEMO divertor engineering design.

In addition the vertical stability (VS) of the plasma configuration must be sufficient that the plasma centroid does not move excessively in the case of perturbations. For the SN baseline, this has been constrained to be $\leq$ 5cm; much greater than this may be a showstopper.

3.5. Disruptions

The main additional concern for disruptions arises from the risks of tungsten dust or flake generation falling into the plasma from above. This requires dust-generating effects, which are likely in a divertor but not elsewhere in normal operation, and so this is considered primarily a risk for the DN configuration. A consultation with plasma-facing materials experts was carried out to discuss this issue [17]. The conclusions were that in normal (ELM-free) operation one would expect prompt redeposition of W. Adherence is good and therefore low dust formation is expected. Experimental results show that significant dust formation is usually associated with transients, co-deposition with e.g. Be droplets, or disruption of W layers [18]. Therefore bulk tungsten in ELM-free, disruption-free plasmas should not form dust. If a low-disruption ELM-free plasma scenario can be assumed for DEMO [19], then dust generation and accumulation should not be a problem.

There are some caveats:

- Erosion and redeposition dynamics will be complicated by surface cracking which may be observed in the complex morphology of the DEMO divertor where edges may be an issue; similarly gap-filling redeposition may occur. There is only scarce experimental basis on which to draw conclusions.
- A cold, rarified plasma at a detached strike point may not cause any sputtered W to ionize and redeposit locally, but will also have much lower source rate. More detached plasma experiments are required.
- Impurity seeding may lead to interlayers and more loosely adherent W, but this is probably less of a problem with e.g. Ar than N.

Additionally, although dust generation in the divertor has been studied we know much less about W eroded from main chamber wall or, in DEMO, from the plasma limiters. Planned experiments in ASDEX-U may confirm morphology and deposition patterns. If concerns remain, further modelling may be required to check up-down asymmetry of deposition and the effects of a reversed field (if required for I-mode operation, for example).

The general problem of first wall protection is discussed in [20].

4. Divertor options

4.1. Baseline single-null (SN)

This is the standard “ITER-like” divertor; a lower single-null of the type achievable on a wide range of existing experimental machines (Fig. 3) [11]. For this work, it acts as the reference.

4.1.1. Remote handling

Remote handling approaches for the baseline single null are well-advanced but are not yet validated (Fig. 4). Questions still remain to

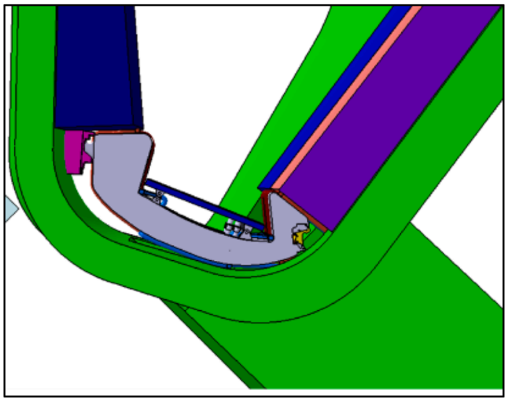

Fig. 3. Geometry of the baseline single-null DEMO divertor. Full description and load specifications are available in [21].

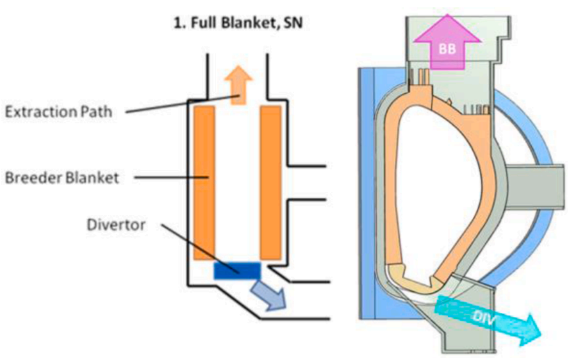

Fig. 4. Basic full-blanket-segment RM concept for single-null baseline [13]. Other options are explored in [13].

be answered regarding segmentation of in-vessel components, seismic stability and component kinematics. It is principally the trade-off between requirements for visibility of in-vessel components through ports, the sizes of the ports, and masses of components which has led to the current 16-TF-coil baseline. The ports are also well-occupied by pipework. It is not expected that any ADC will significantly ease such problems, except where placement of the PF coil set allows ports to be expanded.

4.1.2. Magnets

The baseline PF coil set is relatively well-studied. However the TF shaping poses issues: in particular, there are areas of stress concentration leading to peak stresses above the design limits [8]. A review panel for the DEMO magnets also noted that deviations from a smooth contour may complicate manufacture and increase cost for all winding pack options, and recommended that such deviations should be minimized or eliminated [22]. These can be eased by re-shaping of the coil towards a more so-called "Princeton-D" shape, taller than the vacuum-vessel-wrapping shape assumed up to now (Fig. 5). For the Baseline, this requires some movement of the PF coils but the impact can be assumed to be relatively small. Details of the equilibrium and PF coil layout optimisation process can be found in [6].

4.1.3. Disruptions

Electromagnetic and thermal loads caused by disruptions drive the design of several tokamak components [23]. Presently it is unknown whether massive material injection will be efficient in preventing or suppressing the runaway electrons generated during the current quench of a disruption. This implies that (1) DEMO plasma scenarios must be robust/controllable and disruptions must be very rare, (2) the possibility of protecting the PFCs with sacrificial limiters must be investigated [20] and (3) methods of disruption mitigation for DEMO and future large tokamak devices must be further studied.

While full analyses do not yet exist for the ADCs, alternative configurations from the baseline must not significantly increase disruption risk, nor expose in-vessel components to excessive forces or loads [24]. Of particular concern is the risk of tungsten dust generation and the ways in which it may enter the plasma.

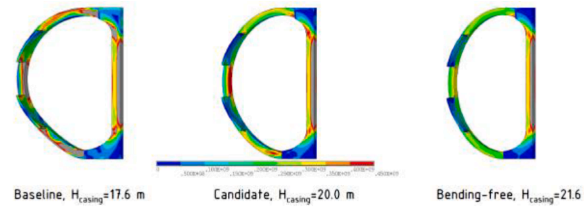

Fig. 5. The original "vacuum-vessel-wrapping" baseline TF shape (left), the bending-free Princeton-D shape (right), and the hybrid stress-minimising shape in the centre [14]. The hybrid coil is 2.4 m taller than the original coil shape. Bright red areas indicate the regions of maximum stress.

4.1.4. Diagnostics and detachment control

The diagnostic suite for DEMO is the subject of the Diagnostics and Control Project, and anticipates using a combination of techniques for the divertor diagnostics, including divertor thermo-currents, spectroscopy and other radiation measurements, and thermography [25]. Some of these elements require line of sight to the divertor strike points from the midplane port. A more complicated divertor geometry such as SX, or additional strike points, such as SF, may require a reconfiguration of the sight lines to ensure sufficient diagnostic elements for control or, potentially, additional vessel penetrations through the vacuum vessel and divertor cassette structure to diagnose conditions in otherwise inaccessible divertor regions. Account must also be taken of the potential movement of strike points, to ensure adequate coverage of all potentially-affected surface.

4.1.5. Plasma control

The plasma control concept is also reviewed in [25].

4.1.6. Divertor design heat loads

Assuming equivalence to ITER divertor geometry, a limit of $(P_{SOL}B_t)/(q_{95}AR_0) < 9.2$ (MW, T, m) was imposed for the baseline. This arises from assuming that the peak power loading on the divertor tiles is proportional to the conducted power entering the scrape-off layer (SOL) P_{SOL} divided by the wetted area, scaled by the machine size and ITER SOL width scaling [26]. In addition, the divertor must operate in detached mode, with the incident particle temperature <5 eV, to prevent excessive erosion from the plasma-facing surface. This is achieved by setting the concentration of seeded impurities, assuming Xe as the principal seeded impurity in the plasma core and Ar in the scrape-off layer. It is assumed that this impurity is present throughout the core plasma at constant concentration, and thus dilutes the plasma. Reducing this seeded impurity concentration by increasing the allowable P_{SOL} would expand the allowable operational space. This is the target of the ADCs, principally, in SF and SX configurations, by increasing the connection length along the magnetic field lines between the plasma and the strike points. For DN, the power is split between two divertors and the radiating volume between two X-points.

Divertor performance is a critical design-driving element as discussed (for ITER-like divertor) in [4]. The aim of investigating ADCs – provided they are feasible from an engineering and control perspective, is to provide a route to relaxing either or both of the divertor heat load or impurity radiation limit conditions, thus expanding the potential operating space for a fusion reactor.

4.1.7. Plasma scenario

The plasma scenario and equilibrium are described in [19]. Deliberate periodic movement of the divertor strike points (sweeping) of the X-point by external coils is being considered as a measure to distribute the heat loads over a larger surface area in the case of re-attachment until detachment is re-obtained [20]. Clearly, this strategy and the definition of the sweeping parameters (i.e., sweeping amplitude and frequency) depends on how large the flux on the target plate is in the case of divertor re-attachment and this depends on the value of the scrape-off width which is still uncertain. Studies have been carried out [27, 16] to determine relevant sweeping parameters and to determine the impact of thermal fatigue on the high-heat-flux components, AC losses, etc. [28, 29].

4.1.8. Modelled divertor heat loads and operational space

Modelling was conducted sweeping over a range of deuterium puffing and impurity (argon) puffing [8]. This modelling shows the outer target at a temperature considerably larger than the inner one, which constitutes the actual bottleneck for divertor operation, and there are few solutions which satisfy the criteria $T_{e,target} < 5$ eV and $n_{e,sep} < 0.6n_{GW}$.

It is this small operational window which is of concern for the single-null baseline, and the reason for the investigation of ADCs and their potential to widen it.

4.1.9. Particle exhaust

Modelling the particle exhaust is difficult and the aim at this stage is to investigate trends and strategies, rather than provide definitive quantitative results. A scenario for the SN baseline configuration has been developed covering varying degrees of detachment [30]. Initial results indicate that the pumping system may be severely stressed due to relatively low pressure in the divertor region. Further work and model refinement is required. Pumping concepts are reviewed in [31].

4.2. Double-null (DN)

An issue with the SN DEMO design is the power loading around the secondary X-point at the top of the machine, extending the requirements for limiters which need to be there for protection against vertical displacements and unforeseen transient events. Such limiters occlude sections of the breeder blanket [32], reducing tritium production and electricity generation. Given that this space may therefore already be lost to TBR, implementing a DN layout may avoid the power loading issues (particularly at the inner limb) and allows access to potentially better-performing physics regimes, but would introduce additional RM complications (Fig. 6).

4.2.1. Inner limb heat loads

In this case the inboard leg carries only a small fraction of the total divertor power [33] and the PFC could be incorporated into the inboard blanket segment for the assumed steady-state loads, with the effects of transients still to be investigated. However this means aligning the lifetimes of the divertor and blanket, which changes the available choices for materials and thus heat-removal capabilities. For example, a conventional (current baseline design) divertor using CuCrZr should be able to handle 10-20 MWm^{-2} but with Eurofer the limit should be assumed to be ~1-4 MWm^{-2} [10]. If this option is pursued further assessment of critical heat flux and the consequences of reattachment need to be considered (divertor sweeping is probably not a mitigation which is available in this case), as well as the plasma becoming SN during vertical displacement events.

This option is currently being investigated elsewhere within EUROfusion [34]. The maintenance could then be carried out by removal of a 'keystone' section containing the outer divertor section, which can be replaced independently. However, this requires additional RM operations, slowing component replacement procedures. An alternative configuration with midplane segmentation of the blankets takes advantage of the DN symmetry to reduce individual component mass; however this means that simultaneous divertor and blanket operations are no longer possible and requires very large spaces below the tokamak for access, further complicating building design and layout.

4.2.2. Remote handling

The remote maintenance of a double-null divertor presents additional segmentation and access problems to the baseline SN case. Coolant pipes for a divertor have to be added in the upper port, and maintenance logistics must adjust to also remove a divertor from the upper port prior to removing the blanket. Space for lifting attachments for the blanket segments is also reduced. This can partly be eased by incorporation of the divertor inner strike point into the inboard blanket segment (Fig. 7) (Fig. 8), however this means aligning the lifetimes of the divertor and blanket, which changes the available materials and thus heat-removal capabilities as discussed earlier.

4.2.3. Magnets

Stress analysis show areas of stress concentration leading to peak stresses above the design limits. These can be eased by re-shaping of the coil towards a more so-called "Princeton-D" shape, taller than the vacuum-vessel-wrapping shape assumed up to now (Fig. 9). As in the Baseline, this requires some movement of the PF coils but the impact can be assumed to be relatively small. Alternatives are also under

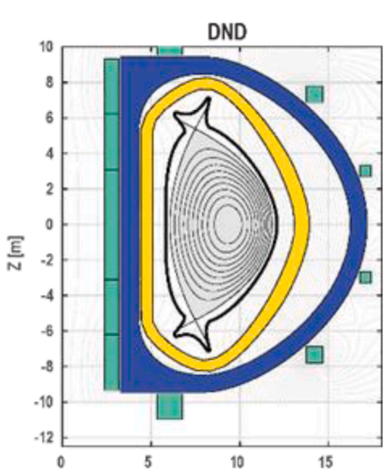

Fig. 6. Cross-section of conceptual double-null PF coilset and plasma [5].

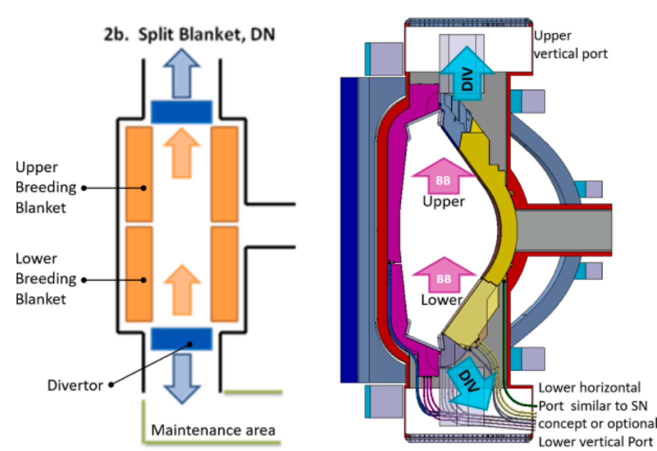

Fig. 7. Several variants for remote maintenance of a double-null configuration are explored in KDII#4: this is option 2b [13].

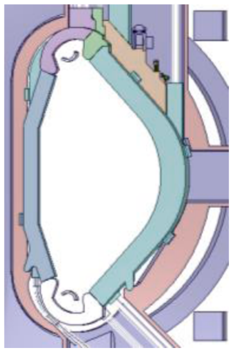

Fig. 8. An alternative segmentation for DN RM, using a "keystone" outer divertor cassette in the upper port.

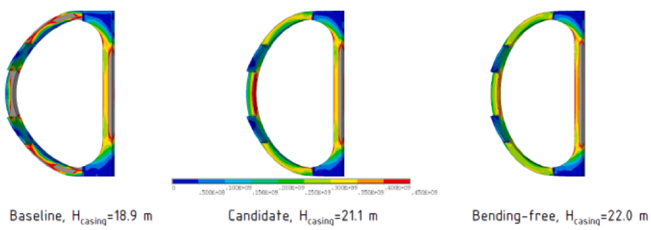

Fig. 9. The original "vacuum-vessel-wrapping" double-null TF shape (left), the bending-free Princeton-D shape (right), and the hybrid stress-minimising shape in the centre [14]. The hybrid coil is 2.2 m taller than the original coil shape.

investigation, such as a "box" intercoil structure as used on the Divertor Test Tokamak (DTT, [35]). Analysis of impacts on the coil stresses are underway; the manufacturability for a DEMO-scale device has not yet been investigated.

4.2.4. Plasma control

The major plasma control issue with a double-null configuration is obviously the balancing of heat loads between upper and lower divertor or, if the vertical movement of the plasma is too great, the transfer of heat load to the inner divertor strike points. The separation of the separatrices at the midplane, δr, needs to be smaller than the scrap-off layer width. Given the ~3m minor radius of a DEMO plasma compared to the <3mm predicted SoL width, to reach this level of precision is a significant potential control issue for the steady-state phases, and even more so in the case of transients.

In addition, the presence of large symmetrical access ports, which remove toroidal continuity for the conduction of currents in the vacuum vessel at the 2- and 4 o'clock positions, can have an impact on the vertical stability of the plasma in DN. There are methods which may mitigate this, such as the use of electrical straps at the port plugs, but such options would add engineering complication and require 3D magnetic analysis to prove a benefit. It is important to recognize that, even if the DN has the worst passive stability parameters of the ADCs due to the increased distance between plasma and passive structures related to the presence of the two divertors, it is intrinsically more stable due to the up-down symmetry of the configuration and of the geometry. [8] shows that in the case of the DN the vertical instability has a growth rate of about 15 s^{-1} and a stability margin of 0.24. If internal coils can be used (under investigation for the baseline) this should also be a powerful mitigating option. [8] also shows that, in case of DN configurations with an up-down symmetric geometry, the vertical displacement in the case of plasma perturbations such as an ELM of minor disruption (mD) is ideally null. For this reason experiments are being carried out (i.e. in TCV), or proposed (e.g. MAST-U, or the international collaboration between DIII-D and EAST [36]) to determine whether the DN configuration can be operated using a smaller stability margin m_s in comparison with the value used for the design of the baseline SN configuration, $m_s \leq 0.3$ [37].

4.2.5. Divertor design heat loads

Modelling of heat loads for DN show an up-down asymmetry in the peak target temperatures, and the operational space is limited rather symmetrically (in-out) by the peak temperatures at the two lower targets [7]. The radiated power fraction varies within the operational space from 60% to 80%. In addition there are indications that a fully detached DN is likely to be unstable: if one leg experiences for some reason a temperature fluctuation, the hotter leg becomes hotter and the colder leg becomes colder. Additional modelling is required to identify if this can be controlled and what the requirements would be. See [7] for a complete description of this work.

4.3. Super-X (SX)

This configuration [38] aims to extend the strike point to high radius, and, in some versions, use a secondary null to spread the power over much larger areas (Fig. 10), although this is not considered for DEMO due to the requirements for additional coils.

However, some of the shortcomings of this approach in a real power plant rapidly become evident. The position of the PF coils mean that horizontal maintenance for the divertor segments is probably required – which makes aspects of the RM easier but means that blanket volume is lost to allow the access, reducing the tritium breeding performance (TBR) [39]. In addition, stress modelling indicates high out-of-plane loads acting on the TF coils challenging the design of the outer intercoil structures [7].

More problematically, from a plant-design perspective, the SX configuration only appears to protect the outer divertor limb, in the case of a detached plasma. In SN, around 33% of the conducted power ends up on the inner limb and so a SN-SX is limited in the benefits that it can provide. However, more recent modelling suggests this may not be true and the heat flux to the inner limb is also reduced: this requires experimental confirmation [7]. A DN-SX configuration would automatically protect the inner limb: however, without in-vessel coils, configurations providing the desired plasma geometry and RM access without hugely exceeding reasonable vertical force limits in the PF coils have proved elusive.

An alternative is reducing the outer limb length, and moving towards an X-divertor.

4.3.1. Remote handling

The proposed PF coil arrangement allows a port between the lower PF coils which may provide access to the divertor cassette in this

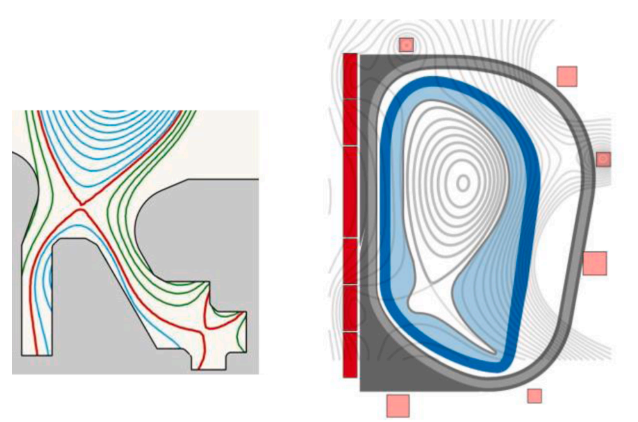

Fig. 10. The basic super-X configuration (left) showing divertor leg extension and flux spreading, and an initial attempt to incorporate it into a DEMO-scale plasma without in-vessel coils (right).

configuration (Fig. 11), but the current concept of the divertor cassette and blanket do not permit extraction kinematics [40]. Resegmentation of the components should be feasible, and another potentially viable option to be explored is the movement of the outer strike point onto the lower face of the cassette, permitting a reduction in the replaceable cassette volume. This affects the recycling of divertor neutrals in the private zone and so modelling is required on the impact of controllable detachment. The configuration is sensitive to PF coil position, particularly PF5, which limits the port position. The arrangement of piping in the port [41] also potentially requires modification to optimise the kinematics.

4.3.2. Magnets

As in the previously-described cases, peak stresses in the coils can be eased by re-shaping of the coil towards a more so-called “Princeton-D” shape, taller than the vacuum-vessel-wrapping shape assumed up to now (Fig. 12). In this case the lower PF coils, which control the divertor configuration, are substantially displaced. Impacts on the equilibrium, PF coil currents and forces, and plasma controllability are yet to be carried out.

4.3.3. Plasma control

The major plasma control issue with a SX configuration is the control of the strike point position, which can move dramatically in the event of an ELM or a minor disruption (mD) [8]. The control on this configuration is a challenging problem. Moreover, the shape sensitivity has an important effect on the power exhaust control for ADCs due to the variation of the power deposition on the divertor target.

In addition the vertical sensitivity of the plasma centroid to external disturbances (ELMs and mDs) makes the vertical controllability of SX, XD and SF very challenging due to the potentially high power request (>1GW) on the vertical stabilization system with external coils, and this may represent a show-stopper. At present these power requirements are speculative and as scenarios are developed, will be recalculated for realistic δβ, δli and current profile redistribution. The use of internal coils for the vertical stability of the plasma needs to be investigated for the alternative configurations (and also for the SN and DN). However, the movement of the plasma centroid is effectively instantaneous and it may not be possible to counteract it.

4.3.4. Divertor design heat loads

A fuelling and seeding scan was carried out [8], targeting at

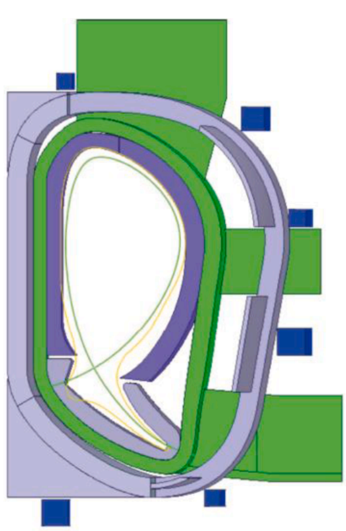

Fig. 11. Conceptual in-vessel segmentation for the SX divertor [8]. There is space for a port between the PF coils, although the cassette proposed here does not permit extraction and further development is required of the concept.

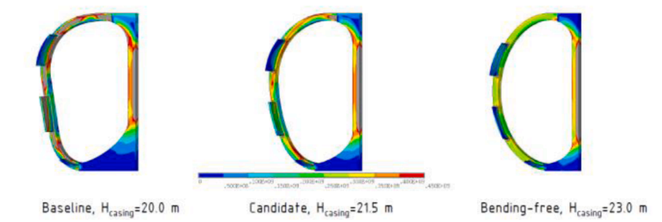

Fig. 12. The original “vacuum-vessel-wrapping” SX TF shape (left), the bending-free Princeton-D shape (right), and the hybrid stress-minimising shape in the centre [14]. The hybrid coil is 1.5 m taller than the original coil shape.

exploring the operating space of DEMO in this configuration. The overall target peak heat load (required to be below 10 MW m^{-2}) turns out to be a looser constraint in this case than $T_{t,max}$. with the inner target the critical limiting factor.

One striking recent result regarding the SX configuration is shown in further modelling indicating that the simple formula $q_{||,ODent}/q_{||,IDent} = L_{in}/L_{out}$, relating the outer:inner heat flux ratio to the inner:outer connection length ratio, should not be used to exclude a particular divertor geometry [8]. In this case, a higher energy flux than expected goes to the outer divertor. In some cases, there is a persistant significant departure of the total heat flux asymmetry away from L_{in}/L_{out}, suggesting a significant role of convective flows for these cases. This behaviour is still under investigation, and further analysis is required to confirm this. However, if confirmed, it suggests that the SX configuration may allow robust detachment at both outer and inner strike points at high P_{SoL}, greatly easing the divertor protection issue. It is strongly urged that this effect is investigated further to confirm that the result is robust.

4.3.5. Particle exhaust

By comparing the different divertor configurations it is deduced that for the same imposed separatrix pressure, the SX divertor appears to have better pumping capability compared to the SN [8]. This behavior is justified by the fact that the SX divertor has the largest separatrix length compared to X and SN, which results to higher incoming neutral flux and consequently higher neutral density in the private flux region. These results are discussed in detail in [7].

4.3.6. Further work and alternatives

There is scope for exploration of shorter super-X divertor limbs, to identify the point at which the benefit of the configuration becomes significant whilst simplifying the engineering. These “hybrid” scenarios are explored in [7].

4.4. Snowflake (SF)

The SF configuration [42] induces a second magnetic null very close to the first, generating 4 divertor limbs 60° apart (Fig. 13). However, although they have been demonstrated in a number of current machines, the physics of SF divertors remain underdeveloped and it is not clear in detail how the power is shared between the limbs. However, the area around the X-point where the connection length is long is large, allowing for high levels of X-point radiation which protect both inner and outer limbs. This permits a promising SF configuration which might allow for acceptable RM access to the divertor through a horizontal port (with the impact on TBR still to be assessed) and with reasonable forces in the PF coils. Previous work has shown that there is an impact on the flux swing supplied by the PF/central solenoid (CS) coilset [33], but the global impacts of this can potentially be mitigated by improved divertor performance allowing lower radiative impurity levels in the plasma, decreasing plasma resistivity and increasing fusion power density. An additional negative impact is that the increased X-point radiation places high EM radiation loads – up to 1 MW m^{-2} – on surfaces close to the

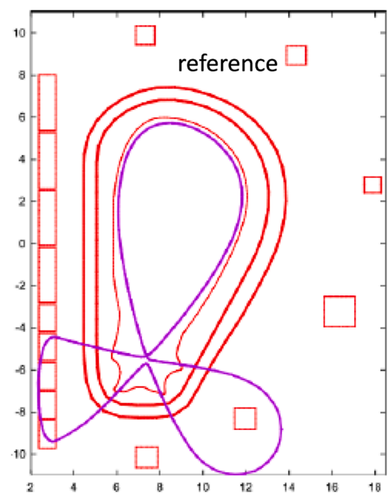

Fig. 13. A snowflake equilibrium, showing the extra limb-splitting in the divertor.

X-point.

A more substantial concern is the control of the divertor heat loading under foreseeable plasma movements. As with SX, small perturbations to the plasma position can shift the divertor limbs – and the heat sharing between them – considerably [8]. This may require defining the design heat load for each single target to be close to that for a standard SN configuration.

3D configurations are in preparation to investigate TF coil stresses, RM kinematics, and neutronics including TBR [7].

4.4.1. Remote handling

The proposed PF coil arrangement allows a horizontal port which may provide access to the divertor cassette in this configuration (Fig. 14), but the current iteration of the divertor cassette is too large for extraction [40]. This is hard to mitigate: expanding the port removes inter-coil structures and exacerbates the TF stress issue; shrinking the cassette reduces the plasma-facing surface area (see strike-point control, below). The configuration is sensitive to PF coil position, particularly PF5, which obstructs the port position. In addition, the additional strike points point to a potential need for additional coolant supply and removal pipes for the divertor. These occupy space in the port (which is at a premium [41]) and increase the maintenance time, with a corresponding reduction in availability.

No clear solution has yet been identified for divertor removal of this configuration.

Fig. 14. Conceptual in-vessel segmentation for the SF divertor [8]. There is space for a horizontal port between the PF coils, although insufficient clearance for the proposed cassette.

4.4.2. Magnets

A hybrid coil shape was developed to ease stress concentrations (Fig. 15). In this case the lower PF coils, which control the divertor configuration, are substantially displaced. Impacts on the equilibrium, PF coil currents and forces, and plasma controllability are yet to be carried out.

4.4.3. Plasma control

The SF strike points are particularly susceptible to perturbation through plasma movement [43]. For the SF this changes both the position of the strike point and the power split between the limbs. This is an intrinsic characteristic of the SF hexapolar geometry, which results in a disadvantage with respect to the other ADCs, will also likely generate more main chamber dust, and it is unclear if it can be mitigated. This lack of strike point position stability means a reduction in the area required for high-heat-flux surfaces, and hence the overall cassette size, is difficult.

As mentioned above, the vertical sensitivity of the plasma centroid to external disturbances (ELMs and mDs) makes the vertical controllability of SF very challenging due to the presently calculated high power request (>1GW) on the vertical stabilization system with external coils. The use of internal coils for vertical stability control remains to be investigated for alternative configurations. However, the movement of the plasma centroid is effectively instantaneous and it may not be possible to counteract it.

Presently, the possibility for safe control of a SF configuration on a DEMO size device is remote. This represents an absolute show-stopper for the use of this configuration on a power-plant scale device.

4.5. X-Divertor (XD)

At the time of the instigation of KDII#3, XD configurations were less developed than the alternatives and it was excluded from the KDII#3 option set. Since then, the WPDTT1 team have made excellent progress and XD emerges as a promising alternative for DTT and potentially for DEMO [44].

This configuration extends the divertor leg and creates a secondary null close to the strike point (Fig. 16). The result is a longer connection length and a poloidal flux expansion. The combination of these two effects potentially allows easier access to high-radiation and detached regimes, passive stabilization of the detachment front, and an increased wetted area, reducing the exhaust power density on the divertor surface.

A brief summary of the main results so far is given below. This work continues and its progress, and potential impact on DTT, are significant.

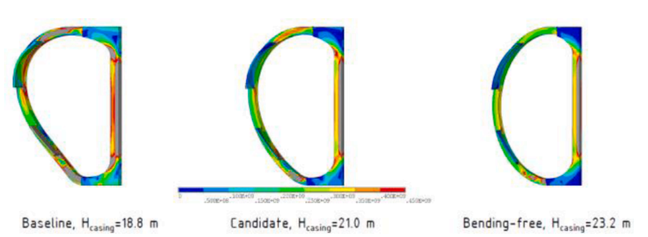

Fig. 15. The original "vacuum-vessel-wrapping" SF TF shape (left), the bending-free Princeton-D shape (right), and the hybrid stress-minimising shape in the centre [14]. The hybrid coil is 2.2 m taller than the original coil shape.

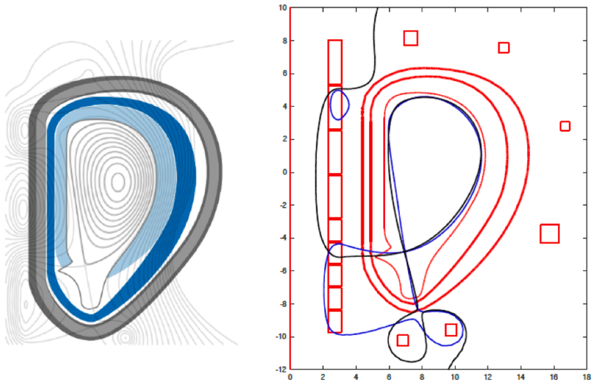

Fig. 16. X-divertor configuration.

4.5.1. Remote handling

The chief challenge posed to remote maintenance for the XD configuration is the placement of the upper coils PF1 and PF2, which restrict the size of the upper port. This means that potentially complex kinematics are required the extract the blanket segments (Fig. 17). This configuration was constructed to maximize port access (but see Magnets, below) and access to the divertor through the large lower port, while not straightforward, does not pose significant issues. Further iterations are required to converge on a solution which meets both maintenance access and magnetic requirements.

4.5.2. Magnets

Preliminary structural calculations for the XD were performed considering a minimal poloidal extension of the outer intercoil structured. This arrangement maximizes port size but diminishes rigidity. In this case, stresses in the outer limb of the TF coils systematically exceeds thresholds. However, this appears to be principally an issue with the current design rather than being intrinsic to the XD configuration: the equatorial and lower ports could be reduced in size to provide for stronger intercoil structures, and while the placement of PF1 and PF2 restrict the upper port sizing and make blanket handling difficult, there is potential for optimization in the placement of these coils as well.

4.5.3. Plasma Control

The XD plasma exhibits relatively large movements of the plasma centroid and X-point in response to changes in l_i and β_{pol}, similarly to the SX and SF configurations. Movements of the strike points, however, although larger than those in the SN and DN configurations, remain within the divertor target and the sweeping effect created may be beneficial under these circumstance.

Plasma control power requirements are increased over the baseline SN configuration, but could be improved through the use of passive stabilizing plates and/or in-vessel coils. These options are discussed in [44].

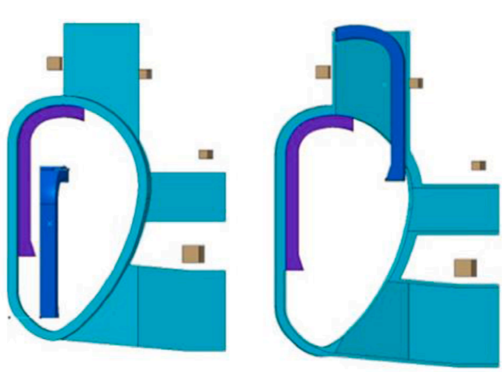

Fig. 17. possible kinematics for extraction of blanket in XD configuration.

4.5.4. Divertor heat loads

Multifluid simulations show an expansion of the operating space over SN, with greater margin in impurity seeding and fueling, meaning that acceptable operating conditions can be achieved with lower Ar seeding and a wider range of fueling levels. Overall this implies that the divertor configuration is less sensitive to external perturbations in these quantities (for example, failed pellets or temporary variations in seeding levels) which can be absorbed without losing detachment.

4.6. Magnet design

It is clear that these alternative configurations pose additional challenges in TF coil design and PF configurations. In general they require more space and therefore the TF coils are larger, with higher stored energy and stresses, and the PF coils are further from the plasma and often in conflict with one another, requiring higher currents than in the baseline SN scenario. With the exception of the DN configuration, the up-down asymmetry means the active power required to stabilize plasma perturbations is expected to be large [43]. The DEMO baseline TF coils are already borderline feasible from a manufacturing perspective, and developing larger or more intricately shaped coils increases DEMO project risks. These increased risks could be mitigated by the reduction of DEMO scope, or through reduction in overall device size which can be accomplished by more optimistic plasma physics assumptions: i.e. a risk transfer from technology to operating regimes with a reduced physics basis, for example higher performance regimes.

4.7. Remote handling

The PF coil layouts have been designed for the configurations outlined here with input from RM specialists, although kinematic studies and segmentation have not started yet for all configurations. Particular concerns relate to the size of divertor cassettes for the different configurations – particularly SX and SF – and the impact on other in-vessel components (IVC) from the horizontal access for these configurations, implying a reduction in TBR as well as reconfiguration and repositioning of the ex-vessel systems to provide access.

4.8. Physics

All the alternative divertor configurations have a reduced physics basis over the ITER-like divertor, and therefore increased overall performance risks. Particular unknowns cover the actual stable radiative performance of SF divertors and their controllability with respect to plasma perturbations. In addition, it is unlikely that the required lines of sight for divertor diagnostics are available from the midplane ports in e.g. SX and SF configurations; these can be achieved but at the cost of additional vessel ports and IVC penetrations in the divertor region.

5. Gaps in analysis

The results presented here are preliminary, and optimisation and investigation of further solutions is ongoing. In particular, convergence on final TF coil shapes which support the stresses generated is required. This is also an issue for the single-null baseline.

This means that some analyses are not available: for example, detailed vacuum pumping with final geometries and port layouts. The physics modelling for alternative configurations is also incomplete, although this applies to the baseline as well. Neutronics impacts (neutron leakage through reconfigured divertor cassettes and enlarged ports, impact on tritium breeding) have recently been published [45].

There is scope for exploration of shorter super-X divertor limbs, to identify the point at which the benefit of the configuration becomes significant whilst simplifying the engineering. The role of in-vessel coils in permitting e.g. X-divertor options and improved control are also under consideration. X-divertor options and their analysis are discussed in [7].

6. Discussion

The purpose of this KDII is to investigate the engineering feasibility of alternative divertor configurations, and identify areas where moving the baseline to one of them would have immediate impacts requiring rapid verification. To that end, potential show-stopping features have been identified.

However, it is clear from the analysis that all the alternative configurations present significant engineering challenges above the baseline SN configuration, and it is not recommended that any of them are pursued on the basis of simplifying the engineering of DEMO, but rather because they may allow divertor-protecting solutions in the event that the baseline fails.

The currently-available physics basis for ADCs is sparse, and models are not strongly predictive. Results are thus in general too preliminary to demonstrate that solutions to DEMO at significantly different aspect ratio or major radius should be pursued, particularly given the other limits on DEMO operating scenario and radial build. Therefore the investigations concentrate on baseline-like scenarios, rather than attempting to construct a wholly-new operating point optimised around these results.

Due to these limitations the ADCs should not be assessed as direct alternatives to the baseline but as risk-mitigation options, some of which may be unfeasible for engineering or plasma control reasons. In order to demonstrate that they expand the potential operational space – and offer a solution where the ITER-like single-null baseline may not – further physics modelling and experimental investigation needs to continue.

A systematic comparison of the divertor plasma performance range for three of the four configurations has been conducted using a state-of-the-art divertor plasma modelling code (SOLPS-ITER). This is probably the most comprehensive study of its type to date. Despite SOLPS-ITER being a reference model, it has a number of simplifications, approximations and free parameters, but the results are still highly illuminating. The SN configuration has been analysed as the reference and the study confirms that the operating range is indeed rather small. The study for SX has shown that with high enough density of neutral deuterium and tritium, and argon (for extra radiation) SX plasmas show a much wider operational range, and indeed can apparently, with even stronger gas injection, dissipate ~300MW, a large fraction of the ~500MW to be exhausted from a 2GW fusion plasmas. This needs more substantiation but is very encouraging.

The DN configuration also has an enhanced operating space, but there appear likely to be significant issues with simultaneous control of the plasma in the four legs, and there are significant up-down asymmetries in the plasma for a symmetric magnetic configuration. An early hope had been that the DN configuration would greatly reduce the power to the inboard legs in nominal conditions, even to the extent of allowing the plasma facing components (PFCs) to have a very long life and be combined into the blanket rather than separate cassettes. However the difficulty of controlling the power sharing, and the need to plan for foreseen and unforeseen transients, suggests this may not be an easy path.

At this current Conceptual Design Phase the direct costs of design changes are relatively small, except in programme delays as the alterations cascade through the integration process requiring analyses to be repeated and other systems to be modified. However, some of these advanced configurations – in particular the Super-X – increase the execution risks of various systems in order to ease physics issues. While DEMO and any subsequent fusion power plant must consist of an integrated solution, possibly not provided by the ITER-like divertor, it is clear that there are no 'easy wins' offered by these configurations.

Currently the focus has mainly been on magnets and RM access: further work, once full 3D configurations are generated, will cover port configurations and IVC attachment and kinematics; exhaust pumping simulations in complex geometries; impacts on breeder blanket design including TBR; and assessment of plasma control issues. Finally capital cost and waste variations will be investigated.

Engineering of coils and intercoil structures consistent with RM access constraints is non-trivial and it seems likely that intercoil structures using single 250mm plates are not sufficient.

Pumping ports have not yet been dimensioned, as they require stable physics scenarios: these may pose additional integration issues.

Control of the SF configuration is extremely challenging. In combination with the issues regarding impacts on available Vs, TF coil design, and remote handling access, this is regarded as a show-stopper for the SF configuration. It is also regarded that X-point configurations, vertical stability, and strike-point position control will remain extremely challenging at reduced elongation.

To summarise the main features of each configuration investigated:

SN: Although the divertor physics is challenging, the engineering is well-studied and there are no as-yet identified show-stoppers to force the choice of an alternative configuration – maintain as baseline.

DN: So far the divertor physics appear attractive, although there is an intrinsically-worse performance for the passive vertical stability (albeit still in the controllable range), but better active stability performance (due to better decoupling of vertically unstable modes with respect to plasma perturbations), than for all the other ADCs. There are significant RM issues. This is the first choice alternative; some issues remain to be resolved, especially regarding unforeseen transient phases, and divertor physics to be clarified. The impact on achieving ELM-free regimes also requires confirmation.

SF: Control of strike points is projected to be extremely difficult, and it has so far not been possible to eliminate high stresses in the TF coils and inter-coil structures. The VS vertical displacement due to plasma perturbations, using external PF coils, represents a current show-stopper. RM access is also difficult. Therefore it should be ruled out of main DEMO work stream.

SX: Preliminary modelling shows physics benefits, but large cassette, and TF stresses require novel intercoil structures – potentially of interest if shown to provide better detachment control. Intermediate options with shorter outer limb should be developed to explore whether potential physics benefits can be realized in a more achievable configuration. Recommend excluded from main DEMO configurations, with potential for re-engagement in DEMO design subject to demonstration of physics performance and wider engineering limitations being met. The VS vertical displacement due to plasma perturbations, using external PF coils, represents a current show stopper if it cannot be resolved through modification of the configuration, for example through shortening of the limb and use of in-vessel control coils.

XD: This configuration shows potential gains in heat handling capacity and improved controllability over other ADCs, but the magnet design and remote maintenance approaches require further convergence before it is clear that the benefits can be realized in DEMO.

7. Conclusions

All the alternative configurations present significant engineering challenges above the baseline SN configuration, and it is not recommended that any of them are pursued on the basis of simplifying the engineering of DEMO, but rather because they may allow divertor-protecting solutions in the event that the baseline fails. That is, they are risk-mitigation options.

Overall, the double-null (DN) configuration is probably the most achievable alternative configuration at this stage, provided RM and transient issues can be resolved. It is recommended that this is maintained in the DEMO programme as the main alternative to the baseline SN variant as a risk-mitigation option in case the SN divertor loading issues cannot be resolved. It is also recommended that work on SX continues and is reviewed for re-inclusion following the completion of remaining tasks.

The future strategy proposed is to focus on a continuum of single null configurations with longer legs (the SX scenarios are essentially longer leg SNs, not true SXs). This continuum approach allows the plasma improvements to be systematically balanced with the increased engineering challenge to clarify the trade-offs required, and thus help the overall optimisation. In addition, the development of XD configurations will continue, not least for DTT..

There is also a target to develop, as early as possible, concepts for fuelling and pumping that combine high neutral density for high dissipation with the constraints on gas throughput from fueling and vacuum systems, since the present simulated throughputs seem to greatly exceed the foreseen pump capacity.

The work developed in the exploration of these alternative configurations, and the direction it provides for future development, provides confidence in the potential for flexible thinking about divertor geometries to provide substantial benefits. The wide design space in this field remains underexplored, partly due to limited existing experimental facilities, but improved modelling and the commissioning of new devices such as MAST-U [46] and DTT aimed at exploring divertor physics expands the scope for new geometries to be developed and their performance verified. As divertor performance remains a significant limiting factor and size driver for DEMO, these studies remain an active area of interest.

Work is continuing to develop a set of performance indicators/metrics, including required end-point performances, to guide the programme and provide measures of progress, and account of technical and schedule consistency with the wider PEX programme, especially the use of new facilities and modelling will be taken.

Richard Kembleton: Conceptualization, Writing- Original draft preparation. Mattia Siccinio: Writing - Review & Editing Francesco Maviglia: Writing - Review & Editing, Investigation Fulvio Militello: Methodology, Investigation

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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