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EU-DEMO design space exploration and design drivers

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ABSTRACT

Once the high-level requirements for a fusion power plant are set, the expected performance of plant systems, including the plasma, must be defined and then the available design space for the plant can be explored in order to converge on a final overall plant operating point, setting the major plant parameters such as number of toroidal field (TF) coils, tokamak major radius, plant power flows etc. Such design space explorations are conducted using systems codes which contain simplified models for plant systems, and attempt to capture the interactions between them in order to balance performance trade-offs and enforce overall consistency. This paper summarizes the work carried out to identify the EU-DEMO baseline operating point and the underpinning assumptions about technology and physics performance. The major design choices are described and the principle drivers for the direction of conceptual development, resulting in the operating space for EU-DEMO, are identified. The final output of the systems code forms the basis for more detailed engineering and physics evaluation and design work.

1. Introduction

The potential basic parameters for a fusion power plant cover a very wide range, from compact low aspect-ratio devices to advanced tokamaks based on high-performance plasma regimes. However, the plasma scenario is only one part of the complete power plant and, in order to balance competing demands on different systems while assuming relatively equal optimism in technology advances, the limitations and performance of all other relevant plant systems need to be taken into account when choosing the overall plant concept. Such limitations may arise from the engineering of the magnets, for example, positioning and sizing of the ports and components to allow effective remote handling and maintenance, and the power-handling capabilities of the materials and coolant of the plasma exhaust components. Targeting higher electrical output also means reducing the power needed to run the plant itself, in particular the plasma heating and current-drive power. The plasma itself must also, like all other power plant systems, be very reliable, which mean that potential higher-performance or high-bootstrap fraction scenarios which are more prone to disruptions or instabilities may be undesirable for those reasons.

The macroscopic behaviour of plant systems can be captured in parameterized models and these can then be combined into a *systems code* [1,2], which captures the interactions and trade-offs between the systems and can be told to optimize the plant parameters, subject to imposed engineering and physics limits, to maximize or minimize a given figure of merit – the plasma major radius R_0 in the case presented here. The design point is then passed to engineering and physics modelling teams who can evaluate the design and performance of each system in far more detail without themselves having to explore a near-infinite set of possible designs.

This paper outlines the decisions made regarding the major parameters for EU-DEMO and the explorations of potential design space around the nominal EU-DEMO design point aimed at giving confidence in the final operating point as a target for engineering design evaluation work.

2. High level requirements for demo

It is important to define the target output of a power plant. These high-level requirements can then be converted into a more detailed technical specification which places the actual performance targets on the specific systems.

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The high-level requirements for EU-DEMO are given as [3]:

- Produce substantial net electrical power (several hundred MWs) for substantial time (hours)
- Demonstrate tritium self-sufficiency
- Demonstrate the operation of supporting plant systems and materials capable of achieving commercial power plant operational availability
- Achievement of DEMO engineering design and evaluation on a timescale which provides continuity from ITER build and operation, i.e. operation in the 2050s

This has been interpreted to mean [4]:

- A target of 500MW net electrical power at flat-top. For the plant recirculating power assumptions and efficiencies of systems, this implies a plasma fusion power close to 2GW
- A pulse length of at least 2 h
- Sufficient room in the tokamak radial build for breeder blanket on both the inboard and outboard sides of the plasma
- A full remote maintenance scheme which can remove and replace in-vessel components from the vacuum vessel and move them to hot cell storage efficiently and in a way which will satisfy nuclear regulators of plant safety
- A closed fuel cycle which meets regulatory tritium release and inventory requirements, meaning exhaust treatment for isotopic separations as well as the installation of a breeder blanket
- A choice of materials, plasma scenario, and technologies which can be developed to sufficient maturity in the time available: this implies that they are available at least at lab-scale now (with technology readiness level, TRL$\geq$4), rather than relying on speculative technologies or effects demonstrated only at the microscale
- It is foreseen that DEMO will utilise a first set of blankets (called "starter") with a damage limit in the first-wall steel (EUROFER) of 20 dpa[1] and conservative design margins, and then switch to a second set of blankets aimed at a 50 dpa damage limit with an optimized design and, if available, improved structural materials that will need to be qualified in advance[2] [7]. An additional benefit of this "progressive" approach is the possibility of starting with a less optimized thermo-hydraulic or mechanical design (i.e. with larger safety margins) to cope with large uncertainties in the overall reactor loadings and performances
- DEMO is also planned to play the role of a "component test facility" for the breeding blanket. As such, its design must incorporate the ability and the flexibility to accommodate the testing of at least one type of advanced tritium breeding blanket concept with the potential to be deployed in a first-of-a-kind fusion power plant

These expansions of the high-level objectives have then been used to make the specific system choices described below.

In addition, there are factors which are hard to capture in systems codes but which, in further detailed engineering analysis, become important. These factors are also not yet well- or numerically-defined and further analysis is required.

- "Robustness" of the operating point: stability against failure to meet requirements for one or more systems; ability to recover overall performance in the case of underperformance of one or more system
- Engineering tolerances for manufacture and assembly or acceptable failure rates for components
- Cost drivers, and the potential trade-off between system performances in order to minimize overall costs. It should be kept in mind that economic competitiveness is not a target for DEMO, but rather demonstration of integrated operation of vital fusion technologies
- Controllability of operating scenario: i.e. robustness against fluctuations in output during operation.

3. Context of EU-demo design

One of the lessons of power plant concept design integration emerging from the concept design phase (2014–20) is that integration of technologies into a coherent design is a central challenge for fusion, and should or cannot be postponed in the hope of unleashing innovation. The operation and placement of the rest of the plant systems impose stringent limits on specific system design and delaying integration considerations leads to a high risk that solutions developed cannot in practice be integrated. Furthermore, fusion is a nuclear technology and as such will receive scrutiny from Regulators under relevant principles such as ALARA (risks As Low As Reasonably Achievable). This means that licensing considerations related to shielding, safety, and remote handling must be taken into account from the outset and can play a significant role in the design. Ultimately assessments in this area can only be conducted once engineering design details have been developed, and thus require a certain maturity of design.

Fusion is currently a relatively small industry, and although there is a generation of engineers who have brought ITER to realization, if the DEMO Engineering Design Phase (to follow the concept design phase) starts too long after ITER is delivered this highly-skilled and experienced workforce will be lost to other industries due to lack of opportunity within fusion. Furthermore, ITER has worked to involve and scale-up industrial partnerships around the world in fusion materials and technology. This interest and expertise would also be lost in the event of a long delay between ITER and DEMO. There are political pressures too: to justify the continued use of public funds to develop nuclear fusion, there must be an emphasis on a solution that allows fast deployment of fusion energy. Furthermore, a critical input to fusion start-up, an external supply of tritium, is largely outside the control of the fusion community. In various modelled scenarios there is only sufficient tritium post-2060 for one large fusion device to enter operation [8]. D-D startup has been considered but is probably unfeasible.

These factors – the need for sufficient data for regulatory approval, the desire to maintain continuity of engineering and industrial expertise, and political considerations – lead to choices of materials, plasma scenario, and technologies as mentioned in the earlier list in Section 2. There is also a high degree of schedule dependency between ITER and DEMO. The 'success-orientated' approach of the EU fusion roadmap advocates concurrency between the exploitation of ITER and development of the DEMO design. That is, the DEMO design activity proceeds in parallel with the ITER exploitation, but relies on a progressive flow of input from ITER for design and physics basis validation, in particular burning plasma operation, prior to authorization of DEMO construction. This schedule aims to maintain the target of electricity production around the middle of this century, seeking the most pragmatic compromise between maintaining an ambitious schedule on one hand and reducing technical and project risks to an acceptable level on the

[1] The choice of 20 dpa as a limit is due to the fact that irradiation of structural materials of interest at this dose value can be simulated with sufficient accuracy in existing Material Test Reactors (MTRs), because the level of He production expected up to this fluence in a 14 MeV fusion spectrum is still relative modest (~300–500 appm, to significantly affect material properties). Fusion irradiation data to be provided in a DEMO orientated fusion neutron Source (DONeS) [5] foreseen to become operative by the end of the decade will be important to validate data collected in MTRs and extend irradiation data at higher fluences, relevant for the second set of breeding blankets.

[2] This type of progressive licencing approach has been used for the fuel cladding in fission reactors for many years; by limiting the maximum exposure level of the replaceable cladding to below the regulatory limit, while data for higher exposure operation is generated in test reactors or load test assemblies [6]. Licensing approval for operation up to moderate damage and activation could be obtained for the "starter" blanket, while high-dose engineering data for a more advanced materials blanket is being generated.

other.

The most critical and final major validation input from ITER, is the demonstration and operation of D-T burning plasma scenarios with $Q = 10$ (> 2035) and the results of the TBM programme (2037–2039). This provides direct operational experience with fusion plasmas and confidence in proceeding with DEMO procurement and construction starting in 2040 [9].

4. Choice of architecture for demo

Some fundamental design choices need to be made at the commencement of the systems code optimization and sensitivity studies. These include:

Number of TF coils – this is a trade-off between the size of the TF coils, set by the acceptable ripple at the plasma outboard mid-plane [10] and which would prefer as many TF coils as possible, and the acceptable access for remote handling which would prefer as few as possible to maximize the sizes of the ports. There is an additional consideration, that the segmentation of the in-vessel components allowing for efficient remote maintenance (RM) operations limits the number of segments per port, so fewer coils/fewer ports implies more massive components, up to a limit allowed by safe RM kinematics. Trade-off studies within the EUROfusion Programme have led to the conclusion that **16 TF** coils is the appropriate trade-off for a device of DEMO dimensions [11]. This compares to the 18 TF coils used in ITER.

Divertor configuration – materials available for the divertor to meet operational requirements mean that the divertor lifetime under full-power operation is approximately half that of the blanket [12]. Efficient maintenance operations mean that the divertor must be independently maintainable, and ideally that the necessary operations can be carried out simultaneously with blanket replacement operations. Further, the bulk of operational large tokamak experience (and anticipated ITER data) lies with lower single-null devices, making this a lower-risk configuration from a plasma perspective. Therefore the current DEMO baseline configuration is **lower single-null** (LSN), although alternatives are considered in the wider programme [13,14]. The choice of an ITER-like divertor also sets heat-load and erosion limits which influence the choice of plasma scenario (Section 4.4).

Aspect ratio, *A* – an aspect ratio study was carried out to investigate the impacts on plasma performance, remote maintenance, and plant engineering of aspect ratios from 2.6 to 4.1 (described in more detail in Section 5). Overall each operating point had benefits and drawbacks and there was no convincing reason or show-stopping element to enforce a particular choice. Therefore, in order to benefit from the bulk of available operating data [15], the ITER-like baseline aspect ratio of **3.1** was chosen. However there are good reasons for A to remain variable in future studies, particularly as further data on e.g. RM access and operation of devices at different A develop.

5. Specific plant systems

This section discusses the justifications for the basic choices of limitations in the systems code exploration for principle EU-DEMO systems.

5.1. Plasma

For the baseline plasma scenario, based on previous ASTRA modelling [16], the IPB98(y,2) ELMy H-mode energy confinement scaling was used with a maximum H-factor of 1.0 (1.1 with "radiation correction" [17] and core density peaking leading to a line-averaged density 1.2 times the Greenwald limit (n_G), this being justified by the low collisionality of the DEMO plasma [18]. The loss power assumed in the scaling law is the total heating power (P_{aux}+P_{alpha}) minus "core" radiation losses [17]. Xenon was used as a deliberately-seeded impurity added to radiate from within the pedestal and hence reduce the conducted power across the separatrix to the level that was consistent with the divertor protection assumptions (Section 4.4), while at the same time sufficiently high to allow the plasma to remain in H-mode [19]. It was also assumed that tungsten would be present at 0.005% atomic concentration due to sputtering from plasma-facing surfaces and transport into the plasma. The consequences of assuming L-mode confinement have been previously explored in [20]. An estimate is made of helium ash content and this dilution is also taken into account. For the design space of interest f_{He}~7% [21].

Safety factor at the plasma edge (q_{95}) is greater-than or equal to three to avoid plasma instabilities. This puts a limit on the maximum plasma current.

Elongation (κ_x) was set to 1.86 at the plasma X-point ($\kappa_{95} = 1.66$), and triangularity (δ_x) of 0.5 ($\delta_{95} = 0.33$). The κ limit was derived from vertical stability calculations [22], and the triangularity was inherited both from ITER and from previous power plant studies as a compromise between high-δ for plasma physics reasons and low shaping for improved maintenance access. For the former of these, it is well-known that plasma shaping has an impact on confinement [23,24], with higher triangularity leading to higher pressure at the pedestal top and higher density in the plasma core. However, high plasma shaping also implies a highly-shaped first wall which impacts on visibility of blanket segment attachment points through the upper port, and remote maintenance kinematics during segment removal. The overall sensitivity of device variables to plasma triangularity is not high [25] in the systems code studies, mainly as it is not represented in the energy confinement scaling laws used. This is however at least partly a consequence of the fact that the dependence of pedestal-top parameters on the triangularity is not taken into account in the simplified PROCESS models.

More recent explorations have coupled the PROCESS plasma to a core plasma transport model [26] including pedestal [27] for improved consistency with wider transport code modelling. In these cases, the H-factor is an output rather than input. Explorations of alternative (non-ELMy-H-mode) scenarios are described in [28]. These generally have an impact on the achievable power (or, conversely, would require an increase in device dimensions to meet the nominal performance targets).

The full assessment of the plasma scenario is summarized in [28].

5.2. Heating and current drive

For the baseline design, it was assumed that some heating power would be required for steady-state burn and scenario control. This was set to 50 MW to represent a time-average value (although more will be required to achieve burn; this also provides margin to stabilize against e.g. tungsten ingress [29]), and in the initial development of the baseline no current drive was assumed to contribute to increasing the burn time, although this was re-introduced in later iterations. This is the most pessimistic assumption for pulse length, and it allows focus on optimizing the electrical efficiency of the plasma heating systems (to reduce plant recirculating power) rather than the current drive efficiency. In the most recent baseline versions, the current-drive fraction from the auxiliary heating and current drive (H&CD) systems is 10–15% of the total plasma current, with the rest supplied by inductive or bootstrap currents.

The plasma model has a self-consistent bootstrap current model [1]. This contributes, for DEMO-like plasmas, in the region of 35–40% of the plasma current.

In order to minimize the recirculating plant power in the current drive systems, DEMO was chosen to be a pulsed device. Based on the physics assumptions outlined above, achieving sufficient current drive for steady-state in DEMO would require around 200 MW of coupled NBI and EC (The flexi-DEMO concept assumes some advances in physics [30] to reduce this). For the baseline case this increases the recirculating electrical H&CD power from around 125 MW to 500 MW, requiring around an additional GW of fusion power to support it.

As a consequence of choosing pulsed operation, there is a flux-swing requirement on the central solenoid. This depends mainly on plasma current and machine size, but is self-consistently calculated within PROCESS. For DEMO, this is around 320 Vs.

Additional localized CD power is required for stabilization of MHD modes. This is outside the scope of the design point, and is discussed in [31].

Since current drive efficiency is therefore not an overriding factor, ECRH is currently assumed as the principal H&CD system due to its ability to achieve high electrical efficiency and perform multiple roles.

5.3. Magnets

The TF magnets were assumed to be ITER-like: Nb_3Sn as the superconductor (SC), stabilized by Cu. The structural material (TF nose and SC conduit) (Tresca) stress was not permitted to exceed 660 MPa, being the lower of 2/3 the yield stress and ½ the ultimate tensile stress for the cryogenic steel of choice [32]. It is this stress, as a result of the Lorentz forces in the coils, that principally sets the size of the coils. In minimizing the size of the tokamak, there is a trade-off between the field achievable in the plasma and the thickness of the coils and hence the overall size of the machine. Overall the radial thickness of the TF coil inner leg is set by the peak magnetic field and the need to (a) resist large mechanical forces and (b) ensure the TF coil protection during possible superconductor current quench by limiting the temperature rise in such an event.

Several options of winding pack (WP) designs are currently under consideration: the calculations to identify the operating point were carried out assuming a non-graded WP, the most pessimistic assumption. Quench protection measures were imposed with limits on the dump time, dump voltage, and peak temperature at the hot spot. These set the SC:Cu ratio in the winding pack. Details of the internal structure of the TF coils within the systems code is given in [2]. The Nb_3Sn current density parameterization used is the ITER model: this does not strictly limit the maximum field but above ~13T the critical current density becomes small.

The TF magnets, in the systems code, are assumed to follow the contour of the vacuum vessel (VV) (leaving space for a thermal shield) on the inboard side and around the top of the VV. (In transferring the systems code output to full 3D design, some alterations are made for a more realistic coil shape.) The position of the outer limb is set by the requirement to keep the magnetic field ripple at the outboard plasma midplane below 0.6%. It is assumed that the ferritic content of the structures (including inserts) between the magnets and the plasma will reduce this further in reality to an overall target of 0.3%. This ripple requirement has the result of pushing the outboard TF limb to larger major radius, so there is a substantial gap between the outboard side of the VV and the TF coils. This improves, to a minor extent, access to the vessel for e.g. RM systems.

There is an estimate made of the neutron dose received by the TF coils during operation. This is based on the amount of material in the blanket and VV between the plasma and the conductor/insulator in the coil. The heating from this dose is also used to estimate the power consumption of the plant cryogenics systems. The cumulative fluence limit for the DEMO TF magnets is set by the total neutron fluence to the epoxy insulator and is taken to be 10^{22} m^{-2}, equivalent to 10^7 Gray.

The PF coils are sized by estimating the currents required to achieve the equilibrium as described in [1]. Again, the basic assumed technology is ITER-like (NbTi superconductor). Limits on vertical forces in the coils are not imposed in PROCESS but have been calculated during the equilibrium refinement and assessment [33]. In addition, the power requirements for vertical stabilization of the plasma using the proposed scenario and PF coil set have been calculated [34]. While PROCESS assumes positions of these coils and uses these to estimate current etc., when the output is transferred to full 3D analysis the calculations are redone in higher fidelity for more realistic coil positions and plasma equilibria.

Fig. 1 demonstrates the consequences for the magnets of trying to

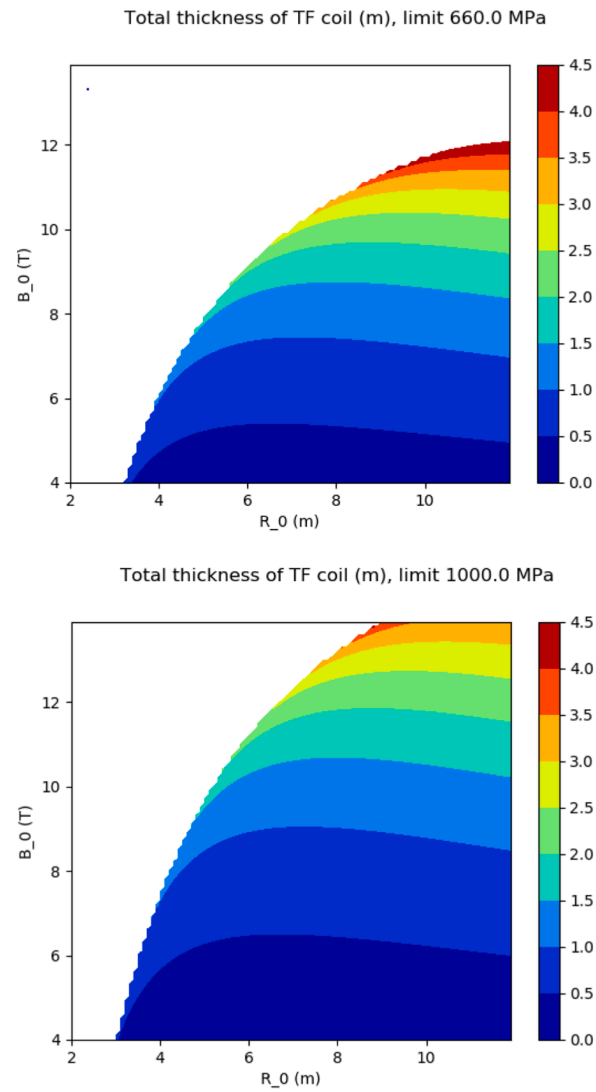

Fig. 1. Estimated TF coil inner limb thickness using DEMO design rules as a function of machine size (major radius, R_0) and magnetic field at the plasma centre (B_0), assuming an infinite current density superconductor. In the top figure the assumed stress limit is 660 MPa, below, 1000 MPa, which opens up some operational space. The blank space in the top left of each plot indicates an absence of solutions to the radial build: in these cases the TF coil fills the entirety of the space inside the VV to the magnetic axis of the machine, leaving no space at all for a central solenoid.

achieve high fields and small devices to target higher power density. The dependence of the superconductor characteristics on magnetic field have been ignored (i.e. this is assumed to use an infinite current-density superconductor, as a best possible case for future superconductor breakthroughs), although an estimate of copper stabilizer content required to protect against quench is used. These figures are plotted in the R_0-B_0 plane and show the required coil size for different structural material strengths. It can clearly be seen that the overall coil dimensions are strongly affected by the stresses imposed by the magnetic field, and this acts as a limit to achievable field regardless of magnetic effects on the superconductor.

5.4. Divertor

Assuming equivalence to ITER divertor geometry, a limit of $(P_{SOL} B_t)/(q_{95} A R_0) < 9.2$ (MW, T, m) was imposed. 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 [35]. This represents a peak heat flux on the plasma-facing surfaces of about 10 MW m^{-2} [36].

The system code is able to vary the concentration of seeded impurities (as well as the other variables in the imposed limit) to try to achieve this, 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, with impurity concentrations in the core and SoL coupled through a "compression factor" to represent transport between the two. The average charge state as a function of temperature/minor radius is then taken into account in calculating plasma dilution, and radiative losses from Bremsstrahlung and line radiation are also calculated, with an impact on energy confinement time loss power and P_{SOL}. In total this results in around 90–95% of the charged particle power being radiated away, from within the plasma core and the scrape-off layer, before the plasma comes into contact with the divertor surfaces.

In addition, the divertor must operate in detached or near-detached mode, with the incident particle temperature <5 eV, to prevent excessive erosion from the plasma-facing surface.

Divertor performance is a critical design-driving element. This is demonstrated in Fig. 2, below. The curves in these figures indicate the acceptable boundary of operational space according to certain criteria: the L-H threshold; a limit on divertor loading, and the required impurity concentration to achieve detachment. These are plotted with respect to the power entering the separatrix scaled to the L-H threshold power (f_{LH}), and R and B (which are somewhat reciprocal in this model). The overall operating point solution for DEMO is a trade-off between meeting these requirements and other engineering limits – for example, the current density and stress limits in the TF coils. The impact of changing aspect ratio is also explored in [35]: at higher aspect ratio the Reinke ($c_{z,det}$<1.0) criterion severely constrains the available operating space. At lower aspect ratio, the issue becomes fitting in elements of the radial build.

Alternative divertor geometries (double-null (DN), super-X (SX), snowflake (SF)) are under assessment in the wider EUROfusion programme. These are discussed in [13]. Mitigation of transient heat loads during reattachment by sweeping have also been investigated [37].

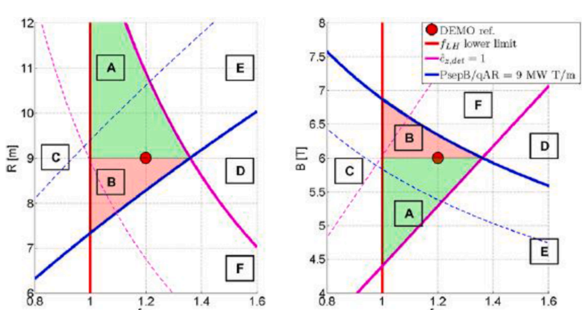

Fig. 2. Representation of constant $c_{z,det}$=1 (radiating impurity levels sufficient to cause detachment) and $(P_{SOL}\ B_t)/(q_{95}\ A\ R_0) < 9.0$ MW T m^{-1} in the f_{LH}-R plane (left) and f_{LH}-B plane (right). Fusion power is assumed constant at 2 GW, and the reference DEMO scenario is shown with a red point. The green areas (A) show the feasible DEMO solutions, with the red areas (B) potentially available with improved, less space-intensive magnet technology. Areas C are excluded by below the L-H threshold, E by requiring excessive impurity concentration to achieve detachment, F through exceeding heat loading on the divertor, and D by both these latter criteria. The dashed lines represent more conservative cases c_Z=0.7 (magenta) and $S = 0.5\lambda_q$ (blue, low spreading factor in the divertor). If these more restrictive criteria are assumed, no solution exists [35]. Overall, the size of a reactor is limited in terms of R by the impurity concentration to reach detachment and in terms of B by the heat flux on re-attachment.

5.5. Balance of plant

The target net electrical power production for DEMO is 500 MW. This is calculated taking into account the demands of all ancilliary plant systems as shown in [2]. A neutron energy multiplication factor in the blanket, from neutron multiplication and tritium breeding reactions, is taken to be 1.27 (this is applied to neutrons in the blanket only) [38].

Thermal power in the blankets is extracted through the primary coolant and converted to electrical power at an efficiency of 37.5% [39]. This figure applies only to the heat deposited in the blanket, and does not take into account the power required for coolant pumping, which is handled separately in the overall calculation. A fraction of the heat extracted from the divertor is also used to pre-heat the blanket coolant to improve energy recovery, although the temperatures of the divertor coolant are too low to be used as for primary heat. The energy multiplication, coolant pumping power, and blanket thickness is based on the helium-cooled pebble-bed blanket concept [40] for the purposes of the systems code evaluation.

Electrical power for other plant systems is calculated as described in [2]. This is dominated by coolant pumping power (~10% of total thermal power, including heat deposited in the divertor and energy multiplication in the blanket, if helium is the coolant, less for water) and the electrical power for the plasma heating systems (assumed to have a 40% electrical efficiency, so 125 MW electrical power is required to couple 50 MW of plasma heating). Overall with the other systems, this means ~450 MW of steady-state electrical power is required to run the fusion power station during a pulse. Additional variable power is required for MHD stabilization systems and plasma vertical stabilization as previously described, as well as all the ancillary systems such as the tritium plant, cryogenics, building services, etc.

5.6. Radial build elements

Many of the elements of the DEMO radial build are fixed: a 225 mm gap between the blanket first wall and plasma is assumed for diagnostics and control purposes; the blanket is 755 mm thick (inboard) and 982 mm thick (outboard); the vacuum vessel (VV) is assumed to be water-cooled, double-walled steel of 600 mm thickness; and there is a thermal shield (50 mm) between the VV and the TF magnets. Elements of the radial build are shown in Fig. 3. The reasons for these thicknesses are described in [41].

The available variables in the radial build are thus the machine bore and central solenoid (CS), the thickness of the TF coils, and the plasma minor radius (which, given that A is fixed, depends on the other elements). The minimum bore and CS size are dictated by the flux consumption requirements of the plasma scenario and pulse length, and the

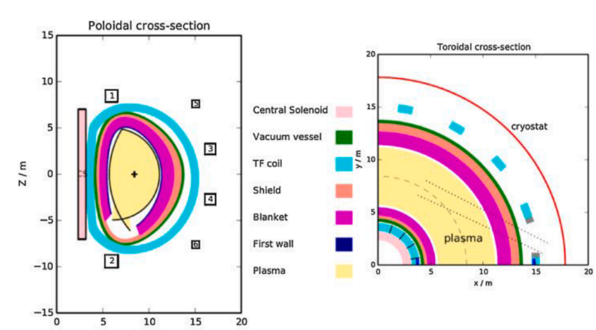

Fig. 3. Cross-sections of PROCESS model of a pulsed reactor. In one of the TF coils the winding pack is shown in blue, and the shielding for the neutral beam duct in grey. The thermal shielding which is needed to separate the cold superconducting coils from the hot reactor inside, and from the cryostat outside, is not included explicitly. The ports for diagnostics and remote handling are not shown because they are not modelled in PROCESS.

TF coil thickness by the demands of the toroidal field (roughly, the TF thickness varies with B_t^2). The interactions of these variables in setting the minimum overall machine size are shown in Fig. 4. As the field falls below the "optimal" value, the plasma size must increase to achieve the target fusion power. This provides more space for the bore and CS, meaning that longer pulses should be available in this regime. As the field is forced above the "optimal" value, the TF coil must grow to accommodate the increased forces and this pushes the machine size up. In this regime, higher fusion power than the target 2 GW should be available.

Vertically, additional space is allowed above the plasma to allow for some vertical displacement, and below the plasma to fit the divertor in. The overall radial build is shown in Fig. 5

6. Wider design space explorations

Elements of the DEMO operating point have evolved over time. There is a PROCESS runs database [42] of the exploratory work. PROCESS – and the DEMO operating point - has been continually updated with model and assumption updates through the period 2014–2020 [43–48]. There have also been various publications following the process [4,9,25,49–51].

A very significant factor that has emerged from this period of studies is the role played by thermal transients, both during normal (ramp-up/down) and off-normal events (such as divertor reattachment) [37]. The need to include specific robust sacrificial features and design margins overcomes narrow optimisations such as size-minimisation of the design point. These behaviours are difficult to capture in simple systems code models and are thus easy to overlook in design space explorations, yet play a strong role in the sizing and internal component specification of DEMO.

In particular, there was a sensitivity study carried out relating the impact of ±10% variations of some of the key inputs [25]. This investigated the sensitivity of net electrical power and burn time on selected parameter limits. The greatest impacts are from plasma elongation (κ), density, and H-factor. Plasma elongation has a high impact because increasing κ allows higher plasma current at the same edge safety factor (and hence better plasma energy confinement) as well as increasing the plasma volume: the overall effect means that fusion power $P_{fus}{\sim}\kappa^5$, all else being equal.

A study of the impact of changing the aspect ratio A was carried out in 2015. A range of operating points with aspect ratios varying from 2.6 to 4.1 was generated, and evaluated for plasma performance, remote handling impacts, and overall achievability [11,25,52,53]. A lower aspect ratio allows higher plasma elongation, but moves the TF coil further from the plasma centre and thus lowers B_t in the plasma. The plasma performance in terms of β limit increases, but overall the fusion

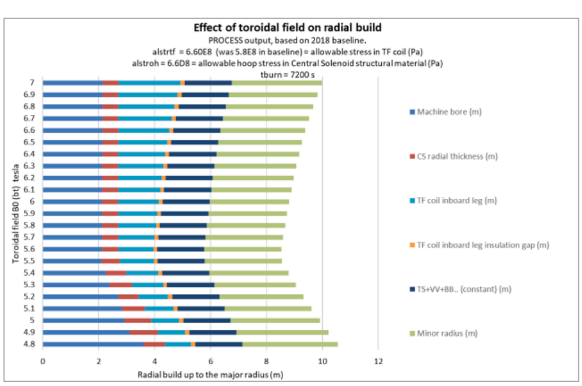

Fig. 4. Effect of forcing higher or lower toroidal fields on the PROCESS nominal solution. (TS+VV+BB indicates the total thickness of the thermal shield, vacuum vessel, and breeder blanket.) See main text for a description of the two regimes of behaviour above and below the "optimum" build.

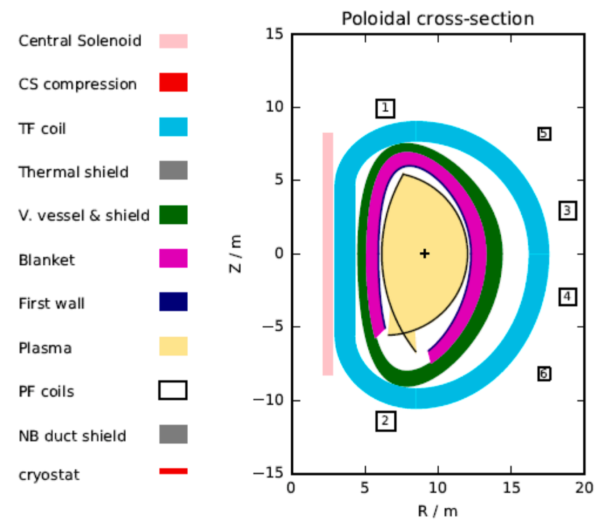

Fig. 5. DEMO 2018 cross-section showing elements of the radial and vertical build.

power density stays much the same [51]. Higher A eases divertor load limits and allows more room for a thicker TF coil (higher B_t) and larger CS (longer pulse), but decreased plasma performance. Overall, there is no clear advantage or disadvantage in changing A, so an ITER-like value was preferred. If steady-state is targeted, the higher f_{BS} enabled by a lower A may become of interest, although achieving all the elements of the radial build limits potential size reduction.

The consequences of using L-mode instead of H-mode (or H-mode-confinement-like modes) has also been investigated [20]. The machine size is much larger (R_0 ~17 m), with a plasma current in excess of 34 MA (with potential for very dangerous disruptions), and a very large TF coil (with manufacturing implications).

Alternative architectures and scenarios are in evaluation, in particular double-null [13,14], alternative divertor configurations [13], and operating plasma scenarios [28]. These are reviewed elsewhere in this Special Issue.

Analyses of the impact of uncertainty in system and component performance on overall DEMO performance has started and continues to be developed [54].

The PROCESS systems code has been benchmarked internationally, against SYCOMORE (CEA), numerous wider international codes, and extensively against TPC (Japan) under the Broader Approach collaboration [55,56].

7. DEMO baseline

Key parameters of the 2018 DEMO baseline are given in Table 1 and the following summary PROCESS output. The assumptions for the PROCESS run are given in [57]

8. Conclusions

The DEMO baseline operating point is the result of a comprehensive set of compromises between system requirements and limitations. A lot of study has been done on conceptual fusion power plants in the past. The distinctive features of our approach, absent in most of the other studies, can be summarized as follows: (i) Realistic physics and technology assumptions, (ii) divertor power handling limitations; (iii) systematic nuclear design integration considerations; (iv) impact of thermal transients arising from normal and off-normal operation. An attempt has been made to be equally optimistic (or realistic) across all systems, based on operation within known regimes based on technology which has been developed to at least lab scale, in order to have

Table 1
EU-DEMO Physics Baseline 2018 relevant machine parameters, produced by the systems code PROCESS. Two divertor challenge measures are given (see section 4.4 for an explanation of these).

Major and minor radius, R, a [m, m]	9.0, 2.9
Aspect ratio, A	3.1
Field on axis, B_0 [T]	5.86
Plasma safety factor, q_{95}	3.89
Triangularity, elongation, δ_{95}, κ_{95}	0.33, 1.65
Plasma current, I_P [MA]	17.75
Non-inductive current fraction, f_{NI}	0.39
Driven current fraction, f_{CD}	< 0.05
Fusion power, P_{fus} [MW]	2000
Power across separatrix, P_{sep} [MW]	170.4
LH threshold power, P_{LH} [MW]	120.8
Confinement H-factor, H_{98}	0.98
Electron density, $<n>/n_{GW}$	1.2
Average temperature, $\langle T \rangle$ [keV]	12.49
Normalised beta, β_N [% mT/MA]	2.5
Z_{eff}	2.12
$P_{sep}B/q_{95}AR$ [MW T/m]	9.2
P_{sep}/R [MW/m]	18.9
Pulse length [*sec*]	7200

EU-DEMO Physics Baseline 2018 relevant machine parameters, produced by the systems code PROCESS. Two divertor challenge measures are given (see Section 4.4 for an explanation of these).

confidence of achieving the target timescales laid out in the EUROfusion Roadmap at the lowest technical risk. Even with this choice, DEMO is still a huge technical challenge.

In general, the exploration of design space carried out using the tools described here has indicated that there are no single points of limitation: for example, if it were possible to produce very high-field magnets without an increase in their size (assuming high-temperature superconductor and very high-strength steel, say), this would allow higher power density in the plasma but this would manifest as more exhaust power transferred to the divertor, which is also operating at the limits outlined here. There needs to be a simultaneous improvement in materials and physics understanding to make overall system improvement possible. The role of transient power loads to components is also hard to capture in an effective multi-system plant optimization yet plays a significant performance-limiting role which is now much better understood.

At present there are many discussions about making fusion power producing devices smaller, cheaper, and faster, but there is no magic bullet to solve the integrated design problems. The present designs of EU-DEMO (either baseline DEMO or flexi-DEMO) are the logical consequence of the most mature knowledge in physics – i.e. the H-mode scaling and exhaust and technology, not an *a priori* desire to be big. These designs also provide a sound and detailed basis for investigating the engineering integration issues, which are considerable.

The size of DEMO is currently limited by the ability to handle the divertor exhaust power for a given machine size, as outlined in Section 4.4. A machine achieving the same fusion power with a higher toroidal field, and thereby smaller major radius, would effectively require a divertor solution capable of exceeding the present performance limit, or high radiative impurity levels in the plasma to reduce P_{sep}, probably impacting on plasma control and access to H-mode. At present there is no clear evidence that the SOL/divertor power handling capability in a standard divertor configuration can be significantly higher than assumed for ITER. In fact, there are big uncertainties on the plasma side due to the lack of real predictive capability. Investigating the effects on plant design of higher limits is straightforward but it is not reasonable to base a design on speculative extrapolations. Alternative divertor configurations are proposed but the plasma performance is unproven and there are considerable problems with integrating them into a practical power plant design, not least managing the remote handling access (see [13]).

A second significant limit on the size of EU-DEMO is the magnet performance (Section 4.3). In the models used, the field available is principally limited by the stresses reached in the coils, rather than the superconductor performance. These forces vary with B^2, and since the coil cannot expand toroidally it must become radially larger rapidly limiting how small the machine can become. With an aspect ratio of 3.1, space for a breeding blanket, and stress limits of <700 MPa in the structural coil materials, targeting a field of 5 T in the plasma leads to a device with R_0>7 m without considering other limitations. A growth in the coil allowing higher fields representative of high-temperature superconductors (HTS) without a corresponding increase in the stress limit results in a larger machine (albeit, one with improved plasma confinement). To an extent this can be overcome by, for example, excluding or reducing tritium breeding from the inboard side to reduce the plasma-magnet distance, but this runs the risk of compromising the ability to breed fuel. Higher fields also reduce the operating space available between protecting the divertor and remaining in H-mode. The option of operation in I-mode is under investigation to explore the consequences of its use, but the current related physics basis (e.g. extrapolation of the LI-threshold) is so weak that it is not consistent with the aim of equally conservative assumptions across all systems.

Allowing a variation in aspect ratio may appear to overcome some of these limits. As the aspect ratio falls elongation can be increased and higher β_N is achievable; however the increased minor radius means that the field in the plasma is lower and the actual plasma pressure does not change much [51]. Overall, for the same achievable field at the TF coil, there is no significant change in power density, although lower aspect ratio designs can deliver higher absolute power due to increased plasma volume (but must still respect power exhaust constraints). This increased power comes at the cost of much bigger in size but thinner TF coils to accommodate the increased plasma volume.

Taking all these elements into account using more detailed models, and allowing for some conservatism as described above, leads to the DEMO baseline of R_0~9.0 m. To significantly reduce the size would require confidence in advances in plasma physics (particularly control and diagnostics in a fusion environment, plasma scenarios that reduce the power density to the divertor target, and highly reliable techniques to mitigate the effects of ELMs or plasma scenarios without ELMs); materials and design solutions to handle higher power densities in multiple parts of the machine during steady-state operation and transients; remote handling approaches that maintain high availability with restricted access; and improved magnets capable of generating higher fields and handling the resulting structural stresses. All of this must be achieved using systems capable of reliable and safe performance in a fusion environment, which can be remotely maintained. In general assuming improved performance in only one system results in a transfer of loads to other systems and only a minor reduction in overall size. In order to have confidence in achieving the high-level goals in the given timescales, such alternative speculative solutions are excluded. This does not mean that EU-DEMO is low-risk, but the approach is chosen to minimize the risk in extrapolation.

If the tolerance for risk is increased, there are potential approaches allowing design changes, which may ultimately reduce the size of DEMO. The first is a reduction in conservatism – a more complete scientific and technical basis allows a reduction in safety margins on extrapolation and increased confidence in plasma control at high radiative fraction, plasma-facing-component (PFC) surface erosion rates, or higher β_N. This may be offset by the need to operate in e.g. ELM-free regimes. It is anticipated that the ITER and DEMO research programmes will naturally improve matters here over time before the DEMO design point is finalised. There is a wider discussion of these issues in [9].

If the high-level goals of DEMO are relaxed (e.g. through a reduction in target electricity production or tritium self-sufficiency, or less targeted technology transfer to a fusion power plant) then size savings can be achieved. Pulse length could also be shortened (to save solenoid space) or lower aspect ratio explored (lower A can generally achieve higher bootstrap current fraction, supporting longer pulse length without additional auxiliary current drive). In the first case, the DEMO mission is compromised and in the second, the design proceeds on a reduced scientific basis.

At first glance, EU-DEMO can look similar to fusion power plant concepts from the past. It is certainly true that once the aspect ratio and magnet technology are fixed, R_0 is also more or less fixed, but from the intensive studies during FP8 we now have a better grasp of the role of divertor performance in limiting power density; plasma modelling is much better, allowing definitions of loads to components, particularly including transients; and materials development much advanced: better lifetiming, expectations of waste handling, etc.

There has also been development of integration models (which will continue in FP9) to enable faster evaluation of impact of system changes on overall plant output or requirements on other plant systems to recover performance.

The DEMO concept emerging from these exercises is a result of breaking down the high-level performance requirements, and timescales available, to define specific technology and operational limits to be assumed in constructing the overall plant operating point. In the wider EUROfusion programme of turning this operating point into a more fully engineered design, we have learned a huge amount about the integration issues arising, and these have progressively fed back into the operating point which has evolved in turn. As engineering and experimentation proceeds, it will continue to evolve.

It is important to note that EU-DEMO is still in a convergence loop, not yet an optimisation loop, and additional concepts are in consideration. The breadth of alternative concepts and their potential for incorporation can be seen elsewhere in this Special Issue.

CRediT authorship contribution statement

R. Kembleton: Conceptualization, Methodology, Investigation, Writing – original draft. **J. Morris:** Methodology, Investigation. **M. Siccinio:** Methodology, Investigation. **F. Maviglia:** Methodology, Investigation. **PROCESS team:** Software, Validation.

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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