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W Biel et al.

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Systems code studies on the optimisation of design parameters for a DEMO tokamak reactor

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In the European strategy towards fusion power, a demonstration tokamak fusion reactor (DEMO) is foreseen as the next single step between ITER and a power plant. The current baseline concept is a tokamak reactor with net electrical output power of P_{el} ~ 500 MW and plasma pulse duration of t_{pulse} ~ 2 hours. Systems codes are commonly used in the design process as numerical tools for optimization studies. The key performance data of the reactor such as electrical output power and plasma pulse duration are depending on a variety of design and plasma parameters. In the application of systems codes within this multi-dimensional parameter space, a clear quantitative understanding of the most suitable optimization criteria has to be developed. On the same time, various physics and technology limits should be obeyed in order to obtain meaningful results.

In this work we use a fusion reactor systems code to perform parameter variations for a pulsed DEMO tokamak reactor. Various output quantities are presented as a basis for the quantitative assessment of the numerical results, and different options for a further development of the current DEMO baseline design are proposed and briefly discussed.

Keywords: Tokamak, Fusion Reactor, Systems Code

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1. Introduction

The demonstration of reliable electricity production in the mid of the 21st century is the main goal of the European Roadmap to Fusion Energy [1]. On the way to developing a fusion demonstration power plant (DEMO), pre-conceptual studies are currently being performed to improve the understanding, to work out the most promising approaches and to compile and resolve remaining physics and technology gaps. Within 2015, a preliminary baseline design for a pulsed tokamak reactor ("DEMO 1") has been defined to serve as a working model [2]. Some key parameters of the current European (EU) DEMO 1 design are listed in table 1.

Parameter	Symbol	Value
Major radius	R_0	9.1 m
Minor radius	a	2.9 m
Aspect ratio	A	3.1
Elongation	κ_{95}	1.59
Triangularity	δ_{95}	0.33
Plasma volume	V	2500 m^3
Tor. magnetic field at R_0	B_0	5.7 T
Max. magn. field at TF coil	$B_{max,TF}$	12.3 T
Safety factor	q_{95}	3.25
Plasma current	Ip	19.6 MA
Greenwald density fraction	n/n_{GW}	1.2
Confinement qualifier	H	1.1
Auxiliary heating power	P_{ext}	50 MW
Net electric output power	P_{el}	500 MW
Plasma pulse duration	t_{pulse}	2 h

Table 1: Key parameters of the current EU DEMO 1 design.

This current baseline was developed by defining upfront the requirements for net electrical output power and plasma pulse duration, and adopting an aspect ratio of $A = R_0/a = 3.1$ for which the most wide database for larger tokamaks exists (including the ITER design). Most of the other key baseline parameters were then following from the goal of minimizing the tokamak dimensions while observing known limitations in physics and technology.

Within this paper, we aim to open the parameter space for a somewhat wider discussion and analysis of options for possible improvements towards the next revision of the baseline. For this purpose, a systems code is used to perform a number of two-dimensional scans of selected physics and design parameters, and a number of different output quantities are presented and discussed towards the suitability for design optimizations.

2. Systems code approach

The systems code used within this study comprises a physics model similar to more sophisticated codes [3, 4, 5], as well as a coarse treatment of radial build and overall costing. Within this short paper, only a brief summary of the main features can be presented.

For the plasma density and temperature, parabolic profiles with pedestal are assumed. While the pedestal density is limited to 80% of the Greenwald limit $n_{GW}[10^{20} m^{-3}] = I_p[MA]/\pi a^2 [m^2]$ in order to ensure a sufficient margin for controllability, the central density n_0 is defined such that the line averaged density remains at a value of $n_{dl} = 1.1 \times n_{GW}$, which results in a moderately peaked profile (profile peaking parameter $\alpha_n = 1$ used here). The pedestal temperature is assumed to amount 15% of the central value, a temperature peaking parameter of $\alpha_T = 1$ is assumed and the central temperature is derived from solving the equation $\tau_{E,scaling} = W_{plasma} / P_{loss}$. Here, W_{plasma} denotes the stored kinetic energy in the plasma, $\tau_{E,scaling}$ is the energy confinement time expressed according to the IPB98(y,2) scaling law [6]. The power loss of the core plasma by conduction and convection is approximated by

$$P_{loss} = P_{fus,ion} + P_{ext} - P_{rad,core} \quad . \tag{1}$$

$P_{fus,ion}$ is the fraction of fusion power carried by ions and absorbed by the plasma. The core radiation power $P_{rad,core}$ is calculated as the sum of Bremsstrahlung, line radiation based on ADAS data [7], and synchrotron radiation following the model from Albajar et al. [8]. Numerical expressions for the most relevant fusion rate coefficients are taken from Bosch et al. [9] and from Slaughter [10]. In all calculations presented below, the core radiation is adjusted by adding Xenon as impurity in order to reduce the power entering into the divertor down to the value of the H mode power threshold, following the scaling proposed by Martin et al. [11] in the formulation

$$P_{LH}[MW] = 1.72 n_{20}^{0.78} B_0^{0.77} a^{0.98} R_0^{1.00} \quad . \tag{2}$$

For the purpose of this paper, we have assumed that the plasma elongation κ follows the relation proposed by Zohm et al. [12] $\kappa = 1.5 + 0.5/(A-1)$, and we estimate the triangularity as $\delta = 0.5 * (\kappa - 1)$. For the radial build of the tokamak, a constant value for the distance between plasma edge (high field side) and inner TF coil of b = 1.8 m has been assumed. To estimate the radial thickness of the TF coils, the space needed for the winding pack is calculated using the Biot-Savart law, assuming a mean current density equal to the value used for the ITER TF coils. The radial thickness of the steel part needed to carry the forces is derived using a model proposed by Freidberg [13]. Both contributions lead essentially to a quadratic increase of the radial TF coil thickness c_{TF} with the maximum field $B_{max,TF}$, as long as the dimensions a, b and R_0 are kept constant.

The remaining space in the tokamak centre $r_{CS} = R_0 - a - b - c_{TF}$ is then available for the central solenoid (CS) coil to provide the flux needed for plasma startup, current ramp-up and maintaining the main part of the plasma current during the flat-top phase. In the calculation of the duration of the flat-top phase of the discharge, we estimate the bootstrap fraction as $f_{BS} = 0.5\,A^{-0.5}\beta_{pol}$, where β_{pol} denotes the poloidal plasma beta, and the fraction of current driven by external heating is expressed by $f_{CD} = 0.011\,T_0 P_{ext}/(n_{20} R_0 I_p)$ with the central temperature T_0 in keV and all other quantities in the units as in table 1. Taking over some settings that were used when defining the baseline design, we assume for the recirculating electrical power $P_{recycl} = 288\,MW$, for the thermo-dynamic efficiency $\eta_{th} = 0.375$ and the wall-plug efficiency of the auxiliary heating system $\eta_{HCD} = 0.4$, respectively.

The optimization of a fusion reactor has to be based on quantitative criteria such as a cost/benefit ratio. For the purpose of this work, we estimate the "cost of electricity" *CoE* based on the total plant cost C_{total} accumulated over the assumed 40 years plant lifetime, divided by the total electrical energy available to the grid within that time

(3) $$CoE = \frac{C_{total}}{P_{el} \times f_{duty} \times 40\,years} \quad .$$

Here, f_{duty} denotes the duty cycle, i.e. the ratio of total burn time to the assumed 40 years of total plant lifetime.

The duty cycle is calculated taking into account a dwell time consisting of a constant of 10 minutes for pump-down and pulse preparation, plus the time for re-charging the CS coil when using an available charging power of 100 MW:

(4) $$t_{dwell} \quad 10\,min + 2\,W_{CS}/100\,MW \quad .$$

The second major contribution defining the dwell time is the time needed for the frequent blanket and divertor exchanges. For simplicity, we assume an equal lifetime of all major in-vessel components (IVC, blanket and divertor) equivalent to a neutron load of 10 MWa/m^2 accumulated at the equatorial level of the low field side, and estimate the total time for exchanging by 5 hours per surface area of one m^2. Depending on the size of the tokamak, this approach results in a 0.5 - 2 years duration for a complete IVC exchange.

For the total cost, we take into account the investment for the magnets C_{mag}, for the remainder of the tokamak C_{tok}, the heating system C_{HCD}, the buildings C_{build}, the peripheral and supply systems C_{periph}, the operational cost C_{op} and the cost associated to the IVC exchange C_{IVC}

(5) $$C_{total} = C_{mag} + C_{tok} + C_{HCD} + C_{build} + C_{periph} + C_{op} + C_{IVC} \quad .$$

For the purpose of this paper, the various cost contributions could only be roughly estimated, taking some figures from the recent paper by Sheffield et al. [14] as input. Specifically, we assume that the costs for the magnet and the tokamak are proportional to the components volume with C_{mag} = 2 M€/m^2 and C_{tok} = 1 M€/m^2, respectively, and estimate the volume of components using a simple onion skin approach. The cost of the heating system is assumed as 20 M€ per installed MW of power. Throughout this paper, we have assumed that the installed heating power is equal to the H mode threshold power (see eq. 2). For the buildings, we take a total of 2 B€ which is scaled up with a factor R_0/6.2 to account for the size dependence. Concerning the periphery (supply systems, conventional power plant systems etc.), we estimate an amount of 1 M€ per MW of plant thermal power. The operational cost (including all maintenance and exchanges apart from IVC) is assumed as 200 M€ per year. Finally, the cost for each exchange of IVC is estimated as 1 M€ per surface area of m^2. For the cases investigated within this paper, each of the various cost contributions amounts to several B€, which results in a total cost over plant lifetime in the order of ~ 40 B€, meaning that the annual cost would be in the order of 1 B€.

3. Numerical results

The systems code has been used to perform three two-dimensional parameter scans in order to investigate a broader range of parameters in the multi-dimensional parameter space, with the aim to search for interesting opportunities for an improved set of parameters for a future baseline definition. In the first parameter scan, the aspect ratio was scanned together with the maximum field at the TF coil $B_{max,TF}$. In these calculations, the safety factor q_{95} = 3, the confinement quality H = 1.1, the relative line averaged plasma density n_{dl}/n_{GW} = 1.1, the net electrical output power P_{el} = 500 MW, the applied auxiliary heating power P_{ext} = 50 MW and the maximum field at the CS coil $B_{max,CS}$ = +/-13 T were kept constant.

Using these settings, the major radius (fig. 1) grows essentially linearly with the aspect ratio A, which means that the minor radius is almost independent from A. On the other hand, increasing the maximum magnetic field at the TF coil allows reducing the major radius almost inversely to the field.

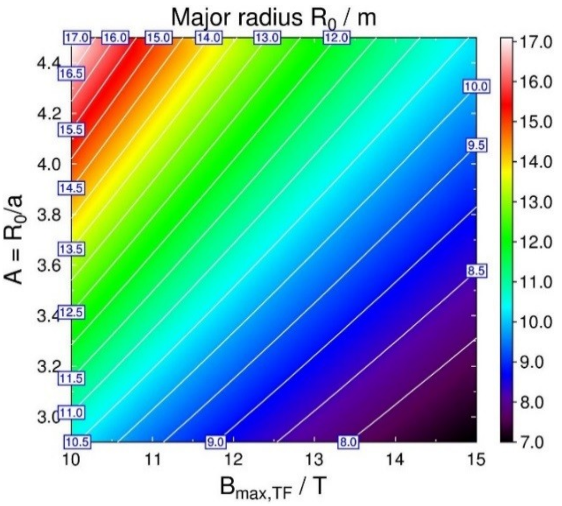

Fig. 1. Scan #1: Major radius

Fig. 2 shows the strong impact of $B_{max,TF}$ and A on the achievable plasma pulse duration. The smaller size of the tokamak as arising from higher field reduces the space available for the CS coils and thus leads to shorter pulses. Higher aspect ratio allows for installing a larger CS coil and hence leads to longer inductively driven pulse durations, which can exceed a full day.

The H mode power threshold, which is seen as a lower limit for the power entering into the divertor, is essentially not depending on the aspect ratio (fig. 3). Increasing the magnetic field, however, increases the heat load towards the divertor. Thus the reactor design could only take advantage from higher magnetic fields (if technically feasible at all), if on the same time an improved heat exhaust capability would become available.

The cost of electricity (fig. 4) remains fairly constant as long as we move along $B_{max,TF} \times A \;\; const$. However, interesting sub-structures (island of low CoE) are visible in the plots which are related to the discrete number of IVC exchanges which range from 2 (upper left corner) to 7 (lower right corner) for the cases shown here. Taking figs. 2 and 4 together, we find options to arrive on the same time at low CoE and reduced divertor load if moving towards smaller $B_{max,TF}$, which means a larger tokamak but reduced cost for IVC exchange over plant lifetime.

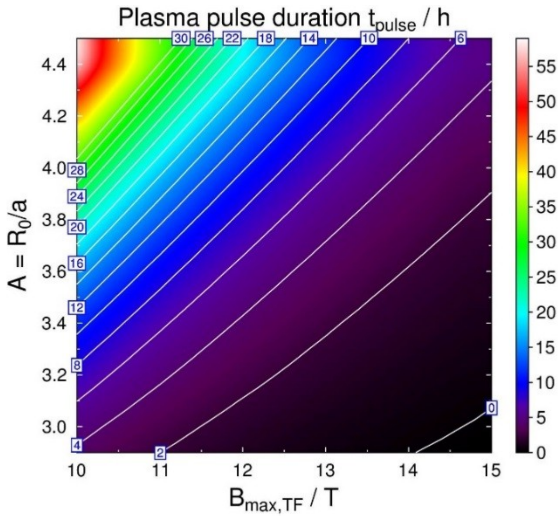

Fig. 2. Scan #1: Plasma pulse duration

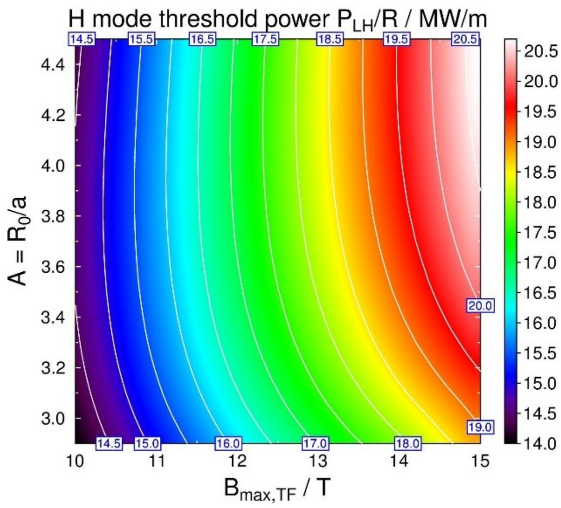

Fig. 3. Scan #1: H mode threshold power

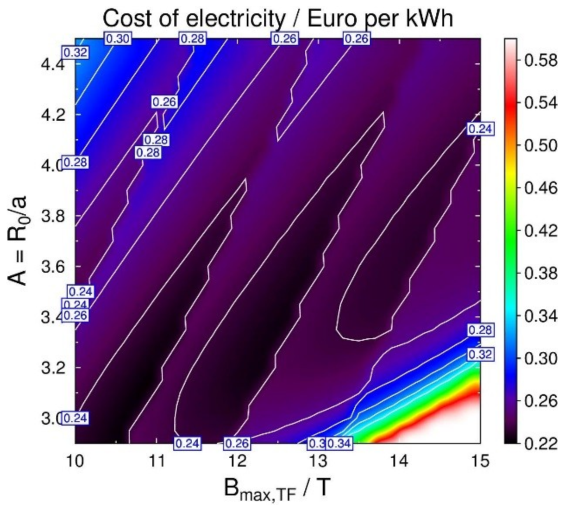

Fig. 4. Scan #1: Cost of electricity

Since the Greenwald limit scales with B/R, lower field is associated with lower density (fig. 5), which may be a disadvantage with regard to the goal of achieving detached divertor conditions. This question however goes beyond the possibilities of the current model.

Finally, we display the heat impact factor $\eta_{TQ}=W/F\,t_{TQ}^{0.5}$ of mitigated disruptions (fig. 6), where W is the energy deposited to a wall surface of area F within the thermal quench time $t_{TQ}^{0.5}$. In the calculation, we have assumed that half of the kinetic energy content of the plasma is deposited to the wall within a time of $t_{TQ}^{0.5}\ 0.5ms\times a[m]/2$, with a peaking factor a 3 accounting for local inhomogeneity. In all parameter ranges investigated here, the resulting heat impact factor significantly exceeds the crack limit for tungsten (~ 5 $MJ/m^2/s^{0.5}$), which means that no parameter window could be found where large area wall damage by mitigated disruptions could be avoided by design.

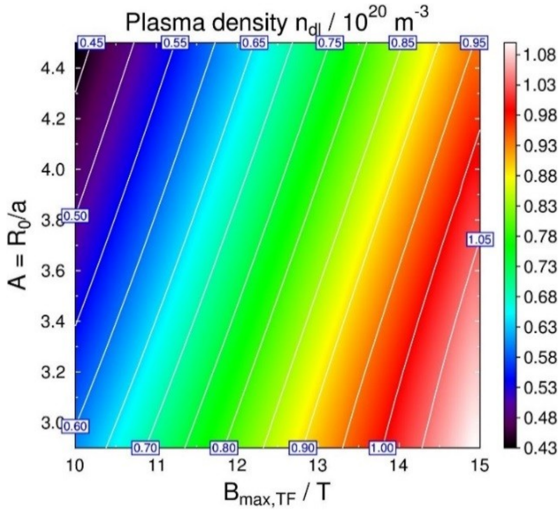

Fig. 5. Scan #1: Plasma density

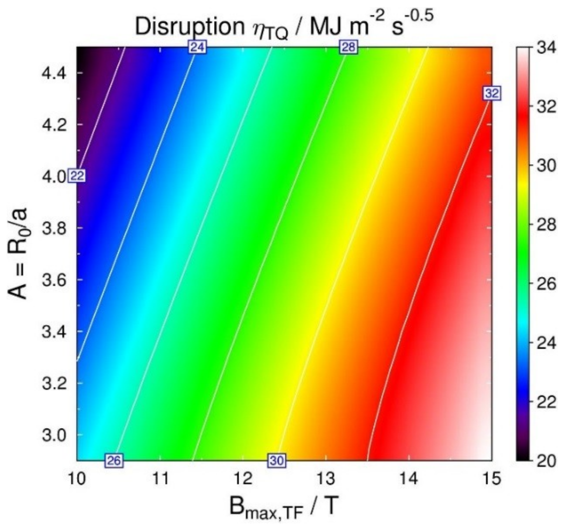

Fig. 6. Scan #1: Heat impact factor for mitigated disruptions

In a second scan, the aspect ratio was scanned along with the H factor in order to see which benefits could arise if a better plasma confinement could be achieved (figs. 7+8). In this scan, the maximum field at the TF coil held constant at $B_{max,TF}$ = 13 T and all other parameters were chosen as in scan #1.

A better plasma confinement allows for a reduction of the tokamak size for the same output power, and hence leads to a reduction of CoE, as long as the space available for the CS coils remains large enough to provide long plasma pulses, see fig. 7. On the same time, higher confinement at constant $B_{max,TF}$ leads to some reduction of the H mode threshold power, so that the power exhaust problem is slightly alleviated, see fig. 8.

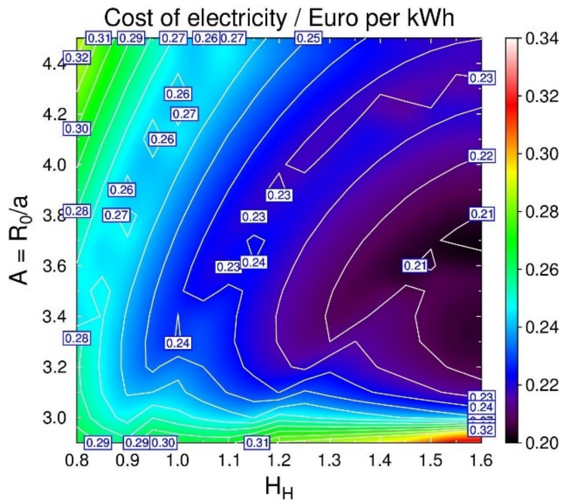

Fig. 7. Scan #2: Cost of electricity

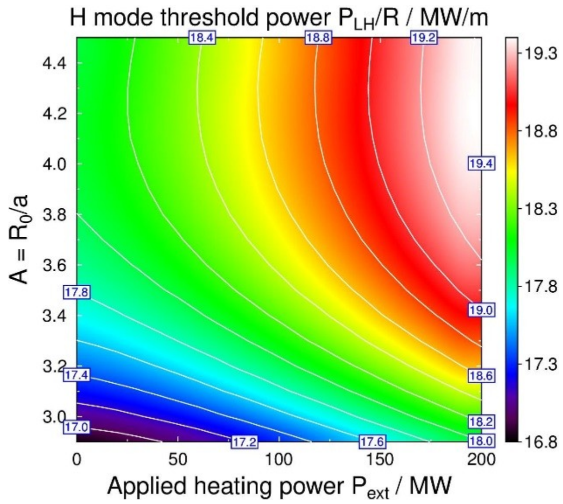

Fig. 8. Scan #2: H mode threshold power

In the third scan, the aspect ratio was scanned together with the applied heating power, in order to see the benefits of low heating power (low recirculating power) and to study which increase of pulse duration could be reasonably achieved under the assumed conservative assumptions for confinement, bootstrap current and current drive. In this scan, the maximum field at the TF coil was set to $B_{max,TF}$ = 13 T, the H factor H = 1.1, and all other parameters were chosen as in scan #1.

The cost of electricity (fig. 9) shows a distinct minimum for low applied heating power for an aspect ratio of A ~ 3.5 where the space available for the CS coil is still large enough to provide long pulse duration (high duty cycle). This shows that operation at high energy amplification $Q=P_{fus}/P_{ext}$ clearly provides an advantage by reducing the recirculating power and hence allowing for a reduction of the tokamak size. Increasing the applied heating power up to 200 MW, we however do not yet reach the region where steady state operation would come in sight, see fig. 10.

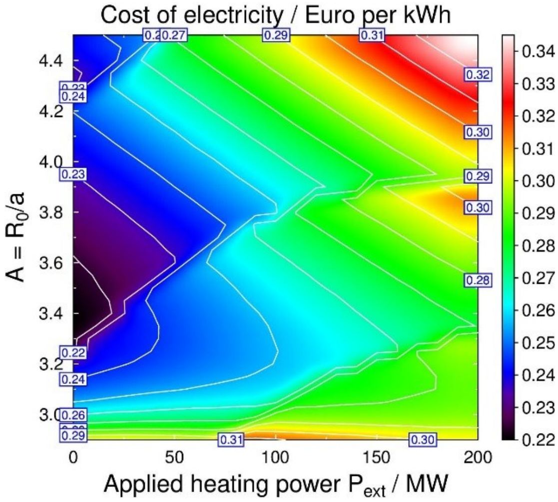

Fig. 9. Scan #3: Cost of electricity

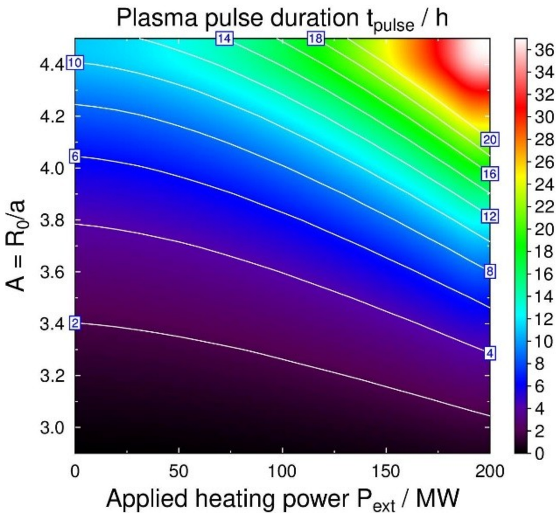

Fig. 10. Scan #3: Plasma pulse duration

4. Conclusions

A wider parameter space around the parameters of the EU DEMO 1 baseline has been investigated to see whether there is room for design improvements. As compared to the reference case, we find that the use of TF coils with somewhat lower field $B_{max,TF}$ would result in an increase of the tokamak major radius, but would allow to reduce the divertor load and obtain longer plasma pulses. Too high magnetic field at low aspect ratio leads to a low duty cycle and hence to unfavorable cost of electricity. Better plasma confinement than the standard H mode (H factor >> 1) at constant $B_{max,TF}$, if achievable, would allow reducing the size of the tokamak as well as the divertor load. In this case, the “cost of electricity” is also reduced, as long as the pulse duration remains long enough to provide a high duty cycle. Operation at low applied heating power reduces the recirculating power and allows for a size and hence cost reduction.

Assuming that the exchange of in-vessel components represents a significant cost figure over the plant lifetime, larger tokamak dimensions (low power density version) provide an interesting route for overall cost minimization. The discrete number of 2…7 blanket exchanges over the 40 years of assumed plant lifetime should be carefully observed when choosing the final design parameters.

Within the parameter range investigated, thermal loads of mitigated disruptions are unfortunately always significantly above the crack limit of tungsten, such that any mitigated disruption during high power phases of DEMO would cause surface damage on major parts of the first wall.

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