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

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HIGHLIGHTS

- A tokamak reactor systems code is described, with a costing module considering blanket exchange.
- Parameter scans of selected physics and engineering parameters are performed for design optimization.
- For cases of 500 MW net electric output power, the cost of electricity only weakly depends on the tokamak size.
- Options for improvements of the current EU DEMO 1 baseline design are presented and discussed.

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ABSTRACT

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 $P_{el} \sim 500$ MW and plasma pulse duration of $t_{pulse} \sim 2$ h. Systems codes are commonly used in the design process as numerical tools for optimization studies. The key performance data of the reactor such as P_{el} and t_{pulse} 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, and various physics and technology limits should be obeyed 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 options for a further development of the current DEMO baseline design are proposed and briefly discussed.

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 baseline design are listed in Table 1.

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 widest 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-

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

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	$B_{\text{max,TF}}$	12.3 T
Safety factor	q_{95}	3.25
Plasma current	I_p	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

dimensional scans of selected physics and design parameters, and several output quantities are presented and discussed towards their suitability for design optimizations.

2. Systems code approach

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

For the plasma density and temperature, parabolic profiles with pedestal are assumed, using the formulation by Kovari et al. [3]. While the pedestal density is limited to 80% of the Greenwald density $n_{\text{GW}}[10^{20}\,\text{m}^{-3}] = I_p[\text{MA}]/\pi a^2[\text{m}^2]$ 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 $n_{\text{dl}} = 1.1 \times n_{\text{GW}}$, which results in a moderately peaked profile (profile peaking parameter $\alpha_n = 1$ used here; see also the arguments on density peaking in Ref. [12]). 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 $\tau_{E,\text{scaling}} = W_{\text{plasma}}/P_{\text{loss}}$. Here, W_{plasma} denotes the stored kinetic energy in the plasma and $\tau_{E,\text{scaling}}$ 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

$$P_{\text{loss}} = P_{\text{fusion}} + P_{\text{ext}} - P_{\text{rad,core}} \tag{1}$$

P_{fusion} is the fraction of fusion power carried by ions and absorbed by the plasma. The core radiation power $P_{\text{rad,core}}$ 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 flow P_{sep} by convection and conduction crossing the separatrix down to the value of the H mode power threshold, for which we use the scaling law proposed by Martin et al. [11]

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

We note that this H mode threshold defines the minimum power which flows towards the divertor under H mode conditions, such that the ratio P_{LH}/R_0 can serve to characterize the required heat load capability of the divertor.

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 used, assuming that this is sufficient to accommodate vacuum vessel, blanket and a gap to the plasma edge, such that tritium self-sufficiency (tritium breeding rate TBR > 1) can be achieved and first wall loads can be kept at acceptable levels. 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 fraction 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 at the inner leg $B_{\text{max,TF}}$, as long as the dimensions a, b and R_0 are kept constant.

The remaining space $r_{\text{CS}} = R_0 - a - b - c_{\text{TF}}$ in the tokamak centre 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 $f_{\text{BS}} = 0.5 A^{-0.5} \beta_{\text{pol}}$, where β_{pol} denotes the poloidal plasma beta, and the fraction of current driven by external heating is expressed by $f_{\text{CD}} = 0.011 T_0 P_{\text{ext}}/(n_{20} R_0 I_p)$ 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 $P_{\text{recirc}} = 288\,\text{MW} + P_{\text{ext}}/\eta_{\text{HCD}}$, for the thermodynamic efficiency $\eta_{\text{th}} = 0.375$ and the wall-plug efficiency of the auxiliary heating system $\eta_{\text{HCD}} = 0.4$, 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

$$\text{CoE} = \frac{C_{\text{total}}}{P_{\text{el}} \times f_{\text{duty}} \times 40\,\text{years}} \tag{3}$$

This approach would have a more stringent meaning in case of a commercial power plant, however, we apply it here to DEMO, since in the European strategy the demonstration reactor is supposed to show the economic viability of a later power plant. The assumed 40 years of plant lifetime are chosen as a typical value for power plants, and for simplicity we neglect interest and inflation. In Eq. (3), f_{duty} denotes the duty cycle, i.e. the ratio of total burn time to the assumed 40 years of total plant lifetime. For the calculation we consider the flat top duration and take into account a dwell time between pulses, estimating a constant of 10 min for pump-down and pulse preparation, and deriving the time for re-charging the CS coil assuming an available charging power of 100 MW (this value was chosen assuming that a level of 20% of the electrical output power will be regarded as acceptable for plant startup and control purposes):

$$t_{\text{dwell}} \sim 10\,\text{min} + \frac{2W_{\text{CS}}}{100\,\text{MW}} \tag{4}$$

A second major contribution entering into the duty cycle is the time needed for the blanket and divertor exchanges. For simplicity we assume an equal lifetime of all major in-vessel components ("IVC", comprising blanket and divertor) equivalent to a neutron load (fluence) of 10 MWy/m^2 accumulated at the equatorial level of the low field side, and estimate the total time for exchanging by 5 h per surface area of 1 m^2. Furthermore, we neglect any other possible reasons for down-times of the reactor, such that the derived duty cycle represents an upper limiting case.

For the total cost we take into account the investment for the magnets C_{mag}, for the remainder of the tokamak core C_{tok}, the heat-

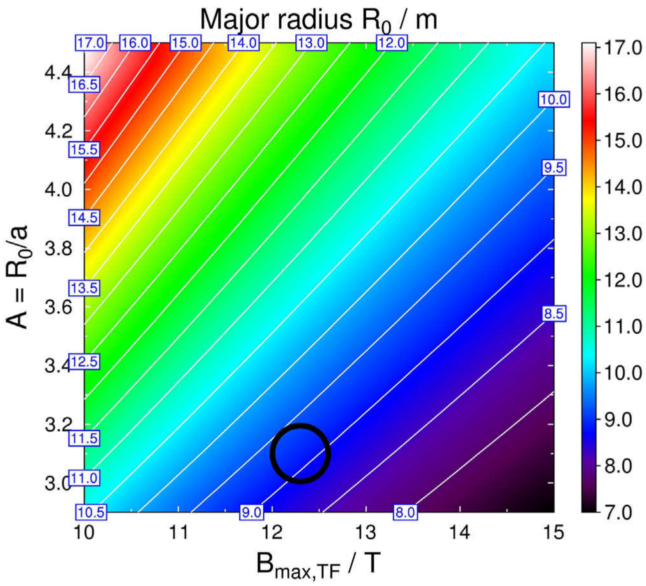

Fig. 1. Scan #1: Major radius (circle: current baseline).

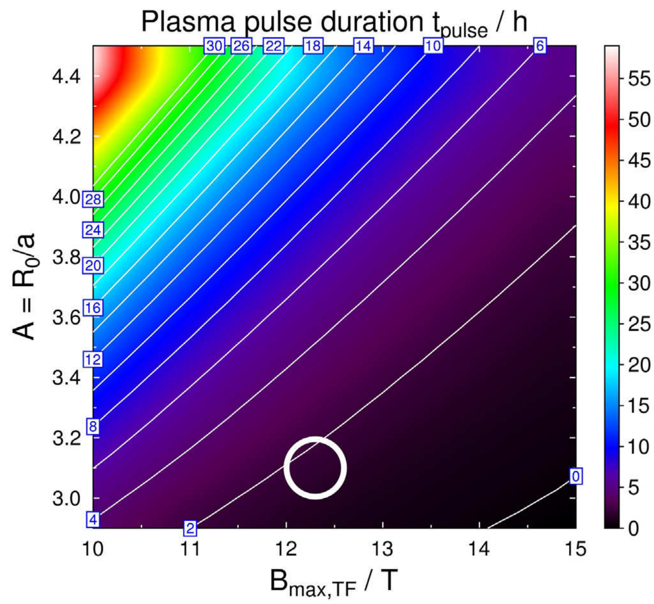

Fig. 2. Scan #1: Plasma pulse duration.

ing 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}

$$C_{total} = C_{mag} + C_{tok} + C_{HCD} + C_{build} + C_{periph} + C_{op} + C_{IVC} \tag{5}$$

For the purpose of this paper, these various cost contributions could only be roughly estimated, using some figures from the recent paper by Sheffield et al. [14] as well as ITER values as guideline. Specifically, we assume that the costs for the magnet and the tokamak are proportional to the components volume with $C_{mag}/V_{mag} = 2\,\text{M€}/\text{m}^2$ and $C_{tok}/V_{tok} = 1\,\text{M€}/\text{m}^2$, respectively, and estimate the volume of these components using a simple onion skin approach. The cost of the heating system is estimated 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 in relation to ITER 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 1 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 average annual cost would be around 1 B€.

3. Numerical results

Three two-dimensional scans of input parameters have been performed in order to search for interesting opportunities for improvements for a future baseline definition. In the first parameter scan, the aspect ratio was varied 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 $B_{max,CS} = \pm 13$ 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. 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.

At constant $B_{max,TF}$ the *H* mode power threshold is essentially not depending on the aspect ratio (Fig. 3). However, increasing the magnetic field increases the heat load towards the divertor. Thus any *H* mode based tokamak 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 of the divertor would become available.

The cost of electricity (Fig. 4) remains fairly constant as long as we move along lines of $B_{max,TF}/A \sim$ const. However, interesting

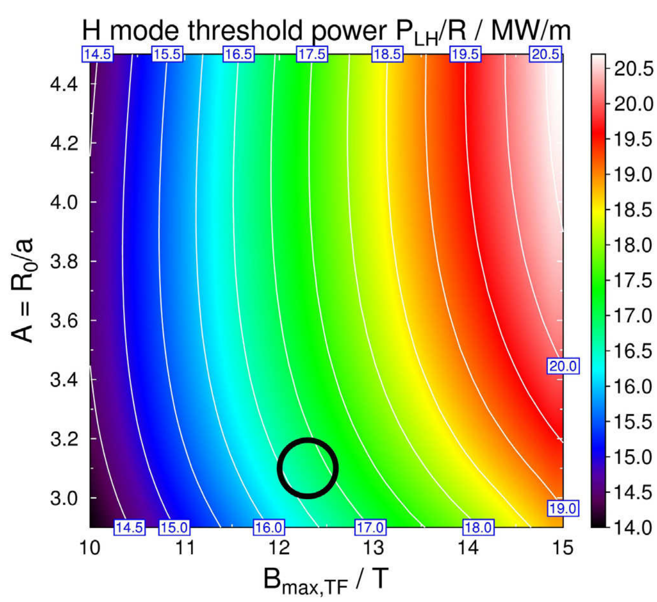

Fig. 3. Scan #1: *H* mode threshold power.

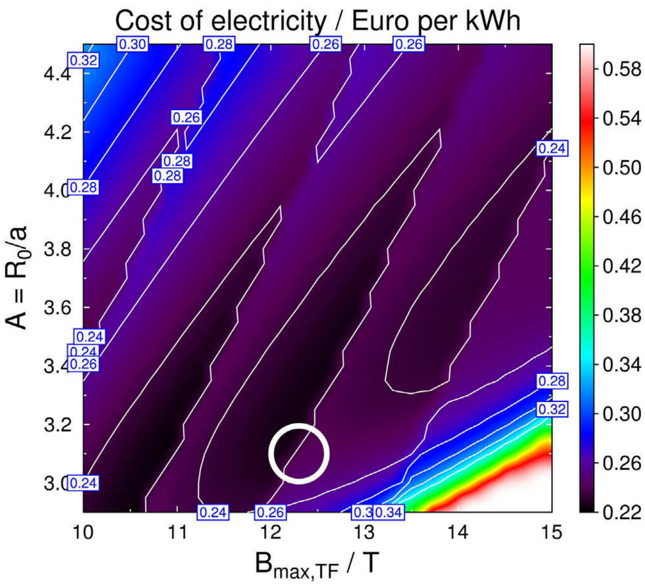

Fig. 4. Scan #1: Cost of electricity.

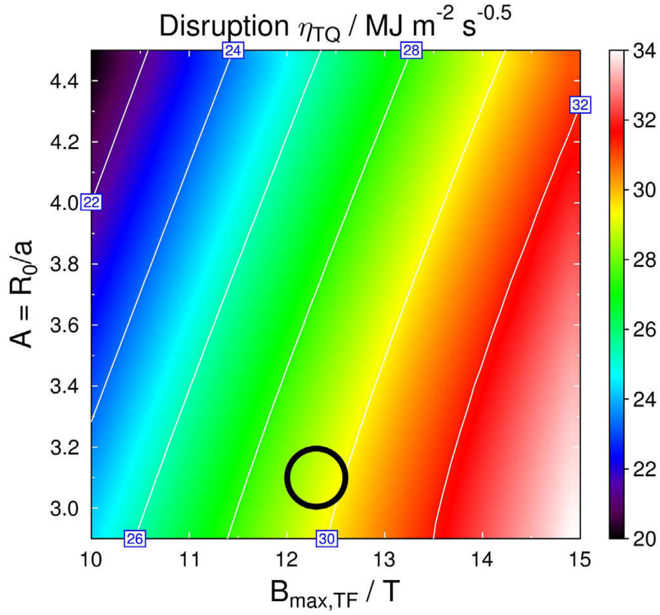

Fig. 6. Scan #1: Heat impact factor for mitigated disruptions.

sub-structures (island of low CoE) are visible in the plot 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 $B_{\mathrm{max,TF}}$, which means a larger tokamak but higher availability and reduced cost for IVC exchanges over plant lifetime.

Since the Greenwald limit scales with B_0/R_0, lower field is associated with lower absolute 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 $\eta_{\mathrm{TQ}} = W/Ft_{\mathrm{TQ}}^{0.5}$ of mitigated disruptions (Fig. 6), where W is the energy deposited to a wall surface of area F within the thermal quench time $t_{\mathrm{TQ}}^{0.5}$. In the calculation we have assumed that half of the kinetic energy content of the plasma is deposited to the wall within a time $t_{\mathrm{TQ}}^{0.5}$~0.5 ms × a[m], with a peaking factor a 3 accounting for local inhomogeneity. The resulting heat impact factor significantly exceeds the crack limit for tungsten (~5 MJ/m^2/s$^{0.5}$), which means that, within the parameter range investigated here, large area wall damage by mitigated disruptions cannot be prevented via the choice of design parameters.

It should be noted that in all cases investigated in this scan the normalized thermal plasma beta β_N assumes values between 1.8 and 2.5 (the higher values at low aspect ratio), such that the ideal beta limit is not violated.

In a second scan, the aspect ratio was varied along with the H factor in order to see which benefits could arise if a better plasma confinement could be achieved (Figs. 7 and 8). In this scan, the maximum field at the TF coil was held constant at $B_{\mathrm{max,TF}}$ = 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,

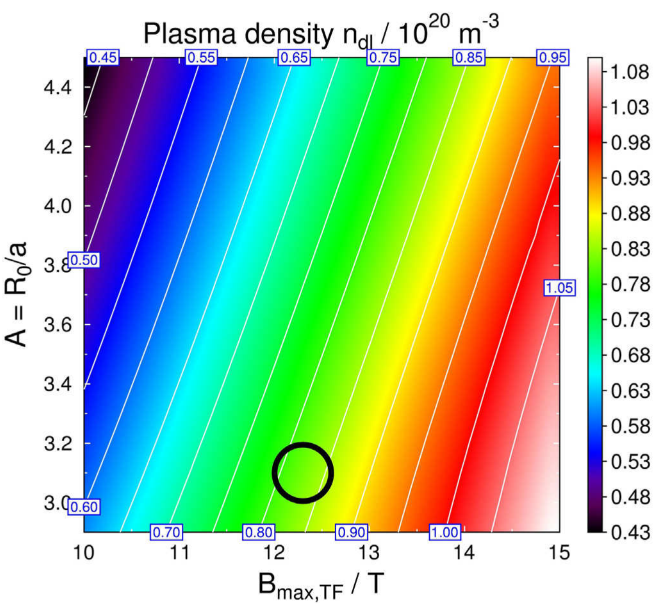

Fig. 5. Scan #1: Plasma density.

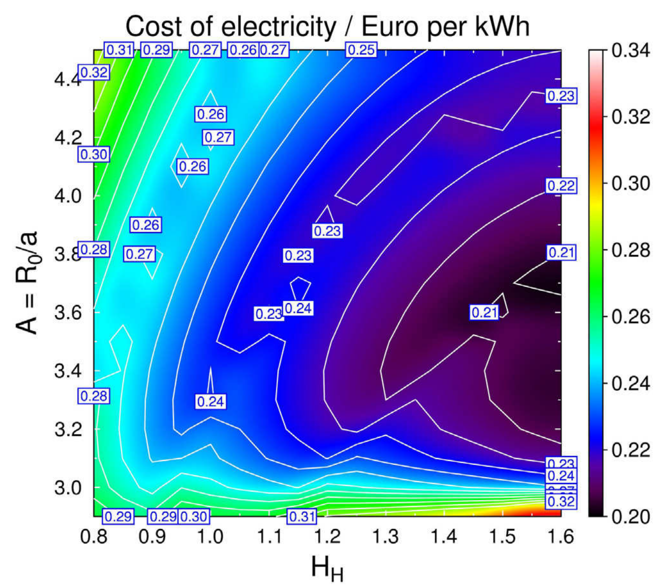

Fig. 7. Scan #2: Cost of electricity.

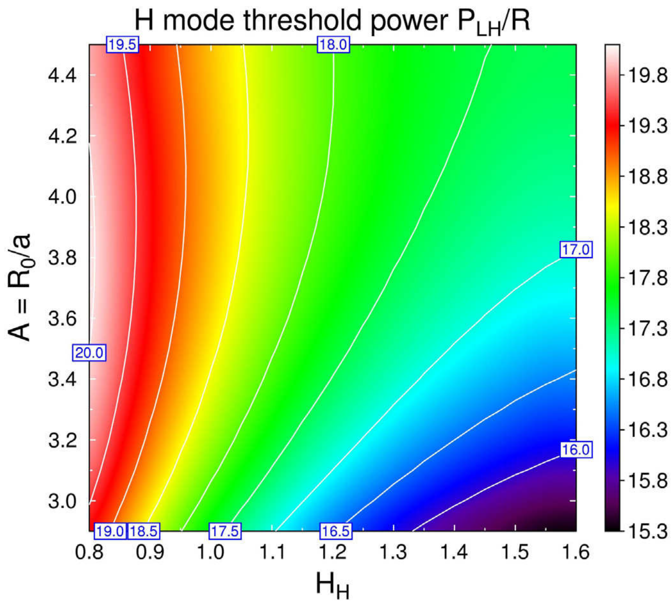

Fig. 8. Scan #2: H mode threshold power.

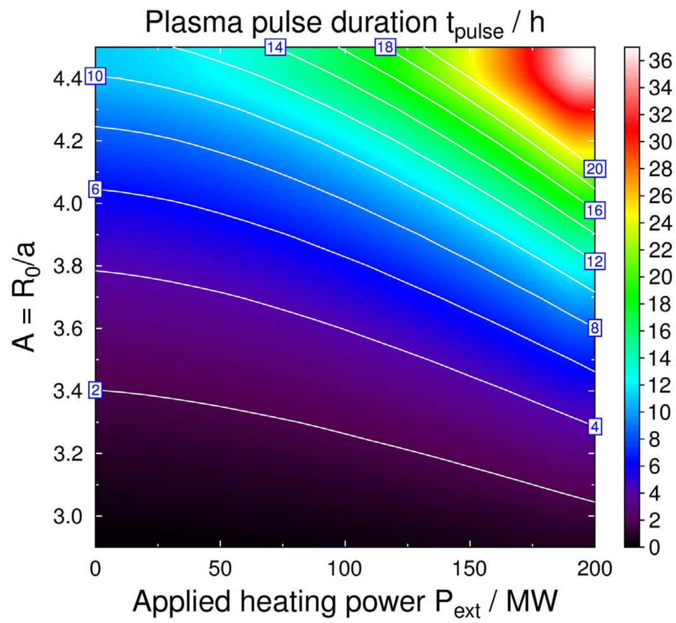

Fig. 10. Scan #3: Plasma pulse duration.

higher confinement at constant $B_{\mathrm{max,TF}}$ leads to some reduction of the H mode threshold power, so that the power exhaust problem is slightly alleviated, see Fig. 8.

For the high confinement cases at low aspect ratio shown here, the normalized plasma beta β_N approaches values up to 3.8, such that the ideal MHD limit might be challenged.

In the third scan, the aspect ratio was varied together with the applied heating power, in order to see the benefits of low heating power (low recirculating power) but also to study the increase of pulse duration 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 $B_{\mathrm{max,TF}} = 13\,\mathrm{T}$, 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 $A \sim 3.5$ 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 $Q = P_{\mathrm{fus}}/P_{\mathrm{ext}}$ clearly provides a cost 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, but obtain a significant increase of CoE, see Figs. 9 and 10.

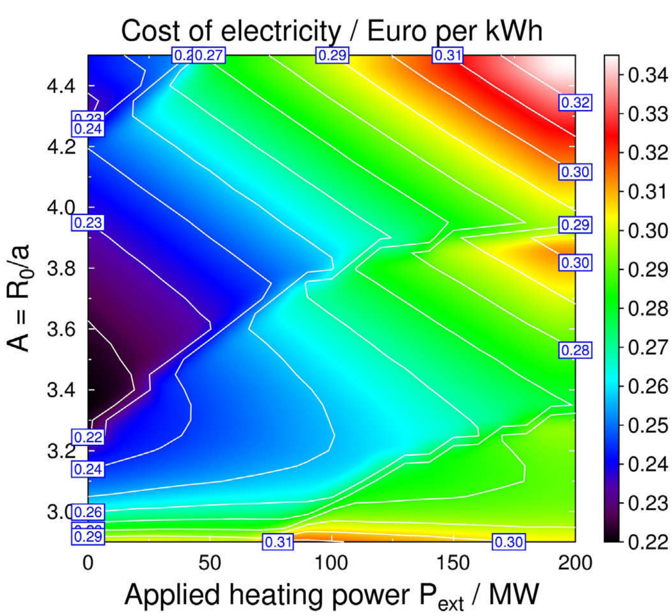

Fig. 9. Scan #3: Cost of electricity.

4. Conclusions

A wider parameter space around the parameters of the EU "DEMO 1" baseline for a pulsed tokamak reactor 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 $B_{\mathrm{max,TF}}$ would result in an increase of the tokamak major radius, but would allow to reduce the divertor load and obtain longer plasma pulses, at essentially the same cost of electricity. Contrary, for the pulsed tokamak case a too high magnetic field at low aspect ratio leads to a low duty cycle and hence to unfavourable cost of electricity. Better plasma confinement than the standard H mode ($H \gg 1$) at constant $B_{\mathrm{max,TF}}$, 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 size and hence cost reduction. On the other hand, with the moderate values for confinement quality, current drive and wall plug efficiencies assumed here, steady state operation can still not be achieved when using up to 200 MW heating power, while the cost of electricity would significantly increase as compared to operation with low external heating power.

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.

In summary, assuming that the exchange of in-vessel components represents a significant cost figure over the plant lifetime and takes significant time, choosing somewhat larger tokamak dimensions (low power density version) provides an interesting route for optimization of a pulsed tokamak reactor with respect to divertor loads and overall availability. In any case, the discrete number of exchanges of in vessel components over the assumed plant lifetime should be carefully considered when choosing the final reactor design parameters.

Acknowledgements

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