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Cost Assessment of a Tokamak Fusion Reactor with an Inventive Method for Optimum Build Determination

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Abstract: An inventive method was applied to determine the minimum major radius, R_0, and the optimum build of a tokamak fusion reactor that simultaneously meets all physics, engineering, and neutronics constraints. With a simple cost model, tokamak systems analyses were carried out over ranges of system parameters to find an optimum build of a tokamak fusion reactor at minimum cost. The impact of a wide range of physics parameters and advanced engineering elements on costs were also addressed. When a central solenoid was used to ramp up a plasma current, design solutions with a cost of electricity (COE) between 109 and 140 mills/kWh, direct capital cost between 5000 and 6000 M/USD, and net electric power, P_e between 1000 and 1600 MW could be found with a minimum R_0 between 6.0 and 7.0 m, and fusion power, P_{fusion} between 2000 and 2800 MW. When the plasma current was driven by a non-inductive external system, the system size and costs could be reduced further; a COE between 98 and 130 mills/kWh, direct capital cost between 4000 and 5000 M$, and P_e between 1000 and 1420 MW could be found with a minimum R_0 between 5.1 and 6.7 m, and P_{fusion} between 2000 and 2650 MW.

Keywords: coupled systems analysis; cost assessment; tokamak fusion reactor

Citation: Jo, G.; Kwon, J.-M.; Cho, A.; Chung, H.-K.; Hong, B.-G. Cost Assessment of a Tokamak Fusion Reactor with an Inventive Method for Optimum Build Determination. *Energies* **2021**, *14*, 6817. https://doi.org/10.3390/en14206817

Academic Editor: Vincenzo Paolo Loschiavo

Received: 27 August 2021
Accepted: 14 October 2021
Published: 18 October 2021

1. Introduction

The size of a tokamak fusion reactor has a large impact on cost, and reducing the size can make fusion an economically viable energy source. An optimum reactor build is composed of multiple components, each with their own function, and each of which can play a role in size reduction. The optimum build of the reactor and its performance are determined by various limits imposed by plasma physics and engineering technology. Physics and technology applied to the ITER designs should be improved to design a tokamak fusion reactor that can produce electricity at a competitive cost [1–6], which includes better plasma pressure, density, confinement, and divertor heat management; advancements in engineering implementations such as the maximum allowable magnetic field at the toroidal field (TF) coil; advanced materials for blankets and shields, and innovations in tritium breeding blanket concepts, power handling, and high-temperature Brayton thermal cycle.

From a cost perspective, a larger toroidal magnetic field at the plasma center, B_T, is preferable for a smaller, high-performance reactor and a small major radius with a corresponding large B_T can be a design goal. The achievable B_T is limited by the allowable magnetic field, B_{max} at the TF coil, as determined by Ampere's law. To increase the B_T, the space between the plasma and the TF coil must be minimized while considering the space for the blanket to breed tritium and for the shield to protect the TF coil from neutron damage. Since the tritium breeding ratio (TBR) is found to reach to a saturated value as the outboard blanket thickness increases [7], the thickness of the inboard blanket can be determined using a fixed outboard blanket thickness. The need for a central solenoid (CS) also has a large impact on the inboard radial build, the thickness of which is determined by the plasma current ramp-up scenario.

A tokamak systems analysis coupled with neutron transport calculations [6,7] can address not only plasma physics and engineering constraints, but also neutronic constraints. Such an analysis can play a key role in the determination of self-consistent system parameters and the optimum build of a small tokamak fusion reactor at a minimum cost. The minimum major radius, R_0, and the corresponding B_T can be determined once the plasma and engineering parameters are specified.

We carried out a parametric study of a tokamak fusion reactor utilizing a coupled systems analysis that included a simple cost model. Though a systems analysis for a single of set of parameters can be performed with a modest computing capability, systematic systems analyses with exhaustive scanning of all parameters require substantially increased computing power. In this paper, we also developed a parallel computing technique to efficiently carry out the tokamak systems analyses involving multi-dimensional parametric scanning.

This paper is organized as follows. We describe the physics and engineering constraints, the inventive method for determining R_0 and the optimum build, and the cost model in Section 2. In Section 3, we explore the impact of physics and engineering parameters on the design and cost of the reactor. Our main findings are summarized in Section 4.

2. Model and Methods

2.1. Physics and Engineering Constraints

The physics and engineering constraints were similar to those used in previous studies [6,7]. Key elements of the model are summarized here for consistency of nomenclature.

A Nb_3Sn superconducting material was used for the magnetic field coil. The TF coil winding-pack was composed of SUS316 stainless steel (30%), Nb_3Sn (25%), copper (25%), an organic insulator (10%), and liquid helium (10%), and its current density was assumed to be 25 MA/m^2. The B_{max} at the inner leg of the TF coil was assumed to be 16 T [8].

A shield composed of ferritic martensitic steel (FMS) and filled with tungsten carbide was cooled with water. Shielding requirements for the superconducting TF coil included a fast-neutron fluence for the superconductor of less than 10^{19} n/cm^2, a nuclear heating rate for the winding pack of less than 1 mW/cm^3, radiation damage for the copper stabilizer of less than 5.0×10^{-4} dpa, and a maximum radiation dose absorbed by an organic insulator of 1.0×10^9 rad, with an assumed reactor lifetime of 40 years.

A lithium lead blanket [2,9,10] was considered for the blanket. A blanket of 90% enriched ^{6}Li was used as breeding material, neutron multiplier, and coolant. For the structural material, oxide dispersion strengthened FMS with a flow channel insert made of SiC was assumed. The volume fractions of the breeding material, the coolant, and the structure were set to 80, 10, and 10%, respectively, and they were assumed to be homogenously mixed. With a fixed outboard blanket thickness, the inboard blanket thickness was determined according to the tritium self-sufficiency requirement, which was assumed to be a TBR > 1.35. This value produces a TBR > 1.08 in a three-dimensional geometry on the assumption that the blanket covers 80% of the plasma surface. Tungsten was used as armor material for the first wall.

The plasma performance of a tokamak fusion reactor is limited by the plasma current, plasma beta, and density. The plasma current, I_p was calculated using the following expression [11]:

$$I_p = \frac{5a^2 B_T}{R_0 q_e} \frac{\left[1 + \kappa^2 + \left(1 + 2\delta^2 - 1.2\delta^3\right)\right]}{2} \frac{1.17 - 0.65/A}{\left(1 - 1/A^2\right)^2}, \tag{1}$$

where a is the minor radius (m), A is the aspect ratio, q_e is the safety factor at the edge, κ is the plasma elongation, and δ is the plasma triangularity. The Troyon beta limit, β_N, is given by:

$$\beta_N = \frac{aB_T}{I_p}\beta_T \tag{2}$$

and the Greenwald density limit, n_G, is given by:

$$n_G = \frac{I_p}{\pi a^2} \tag{3}$$

The plasma power balance equation can be expressed as:

$$P_\alpha + P_{aux} = \frac{W_{th}}{\tau_E} + P_{Brem} + P_{Sync} + P_{rad}, \tag{4}$$

where P_α is the α-particle heating, P_{aux} is the auxiliary heating, W_{th} is the thermal energy of the plasma, P_{Brem} is the power loss from bremsstrahlung radiation, P_{Sync} is the power loss from synchrotron radiation, and P_{rad} is the radiated power in the plasma from impurities, which were assumed to be 30% of P_α + P_{aux} [12,13]. For the energy confinement time, τ_E, we used the H-mode IPB98y2 scaling law [14] with a confinement enhancement factor, H, of

$$\tau_E = H\tau_E^{IPB98(y,2)}. \tag{5}$$

We assumed that the plasma current was non-inductively driven and composed of an externally driven current and a bootstrap current. A formula derived by Pomphrey [15] was used to estimate the bootstrap current fraction, f_{BS}. A neutral beam current drive was used as the external current drive, and the beam current drive efficiency, γ_{cd}, was calculated by utilizing a formula derived by Hirshman [16] and Lin-Liu [17]:

$$P_{cd} = (1 - f_{BS})I_p/\gamma_{cd} \tag{6}$$

The power transferred to the scrape-off layer was calculated as follows:

$$P_{SOL} = P_\alpha + P_{aux} - P_{Brem} - P_{rad}. \tag{7}$$

P_{SOL}/R_0 was used as a measure of divertor power handling and the divertor heat load, q_{div} was estimated with the Harrison–Kukushkin divertor model [11].

Models of other plasma performance can be found in previous studies [18–20].

2.2. Determination of the Minimum Major Radius and the Optimum Build

An inventive method was developed to determine the minimum major radius and the minimum reactor size of a tokamak fusion reactor that simultaneously meets all the physics and engineering constraints. A tokamak systems analysis coupled with neutron transport calculations [6,7] can incorporate not only plasma physics and engineering constraints, but also neutronic constraints, and it plays a key role in the determination of self-consistent system parameters and the optimum build for a minimum-cost tokamak fusion reactor.

Given the fusion power and plasma performance, the dependence of R_0 on B_{T} can be determined. As B_{T} increases, R_0 decreases and neutron wall loading increases. For a small reactor, an inboard blanket thickness with a large outboard blanket thickness is preferable. Outboard blanket thickness can be fixed because the TBR saturates as the blanket thickness increases. For a given B_{T}, the inboard blanket and the shield thicknesses are determined to satisfy the requirements for tritium self-sufficiency and shielding of the superconducting coils. The effect of reflected neutrons from the shield on the TBR and the shielding contribution from the blanket are considered. The TF coil thickness is determined by the winding-pack current density, B_{max}, and stress requirements. The ripple requirement determines the location of the outer TF coil. The CS thickness is determined by the winding-pack current density and its role in plasma current ramp-up scenarios. The CS bore radius, R_{bore} is determined by Ampere's law. As B_{T} increases, the inboard blanket thickness decreases slightly, shield thickness increases due to increased neutron wall loading, the TF coil thickness increases due to the decreased winding-pack current density, and the R_{bore} decreases. The minimum R_0 with the corresponding maximum B_T, and the optimum build are determined when the magnetic field at the TF coil or the R_{bore} reaches its limit. The minimum R_0 and optimum build are therefore determined by the plasma performance, role of the CS, B_{max}, and ripple requirement, while the neutronic requirements for tritium self-sufficiency and shielding determine the thicknesses of internal components such as the blanket and the shield.

The plasma was assumed to have a double-null configuration. The TF coil, vacuum vessel, shield, blanket, and the first wall were D-shaped in the cross-section as shown in Figure 1. The distance from top x-point to the divertor, *vgap1*, the thickness of divertor structure, *divst* and the gap distance to TF coil, *vgap2* were input parameters.

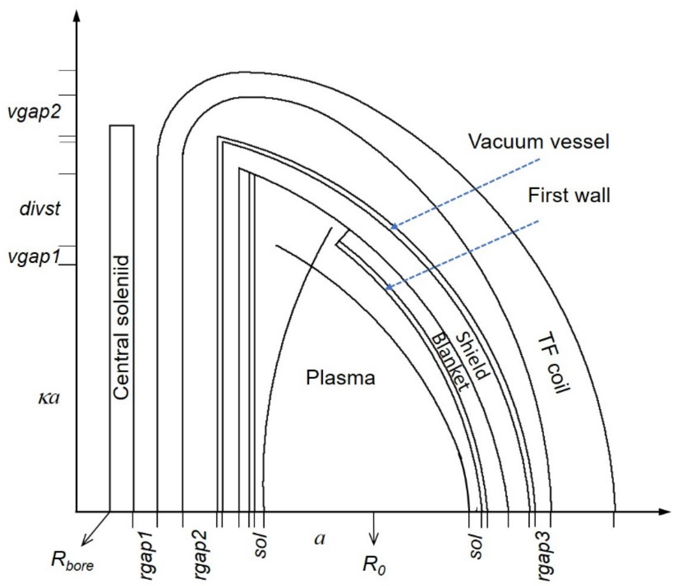

Figure 1. Cross section of the tokamak fusion reactor.

2.3. Cost Model

The cost analysis in this study used a methodology developed in [21,22] due to its simplicity and conformity with more sophisticated cost models used in the study of specific fusion reactors [1,3,4]. All capital cost (hereafter in this paper, capital cost denotes direct capital cost) and cost of electricity (COE) calculations were in 2010 U.S. dollars consistent with the cost model in [21].

The COE (mills/kWh, where 1 mill = 0.001$) was calculated utilizing the constant-dollar-levelized COE formulation in [22]:

$$\mathrm{COE} = 10^6 \left(C_{C0} F_{CR0} + C_f + C_{om} \right) / (8760 \cdot f_{av} \cdot P_e) + 1.0 + 0.5. \quad (8)$$

The total capital cost, C_{C0} (M$) was

$$C_{C0} = C_D f_{CAP0} f_{IND}, \quad (9)$$

where C_D is the direct capital cost, f_{CAP0} is the constant-dollar capitalization factor, and f_{IND} is the indirect charge for a 6-year construction time. F_{CR0} is the constant-dollar fixed-charge rate. The total direct capital cost, C_D (M$) was given by

$$C_D = 1.15 (C_{bop} + C_{bldg} + C_{FI}) \quad (10)$$

The multiplier of 1.15 is a contingency factor. The first term, C_{bop}, includes the cost of the balance of plant (BOP), and is calculated by $C_{bop} = (900 + 900\frac{P_e}{1200})(\frac{P_{th}}{4150})^{0.6}$. P_e is the net electricity power (MW) and P_{th} is the recovered thermal power (MW). The second term, C_{bldg} includes the cost of the reactor building, hot cells, vacuum systems, power supplies and peripherals, and cryogenic systems, and is calculated by $C_{bldg} = 839(\frac{V_{FI}}{5100})^{0.67}$. V_{FI} is the volume of the fusion island and can be calculated by a coupled systems analysis. The cost of the fusion island, C_{FI} (M$) was given by

$$C_{FI} = 221(P_{th}/4150)^{0.6} + 1.1C_{reactor} + 1.5C_{coil} + 1.1C_{aux} \tag{11}$$

where the first term is the cost of the main heat-transfer steam system, the second term is the cost of the shielding and structure multiplied by a factor of 1.1 to account for extra shielding, the third term is the cost of the coils multiplied by 1.5 to account for extra coils, and the fourth term is the cost of the auxiliary heating system multiplied by a 1.1 to account for spares. The blanket and diverter costs are included in the annual fuel cycle cost. Costs of the second to the fourth terms are calculated based on the coupled systems analysis.

C_f (M/year) was annual fuel cycle cost and had several components:

$$C_f = C_{blka} + C_{diva} + 0.1C_{aux} + 7.5 \tag{12}$$

The annual blanket and divertor costs ($M/year) were given by:

$$C_{blka} = 1.1[1.1C_{blk}F_{CR0} + \left(\frac{f_{av}N\Gamma_n}{F_{blk}} - 1\right)C_{blk}/N] \tag{13}$$

$$C_{diva} = 1.2[1.1C_{div}F_{CR0} + \left(\frac{f_{av}Nq_{div}}{F_{div}} - 1\right)C_{div}/N], \tag{14}$$

where f_{av} is the plant availability, N is the assumed lifetime of the reactor, Γ_n is the neutron wall load, F_{blk} is the lifetime limit of the neutron fluence on the blanket, q_{div} is the divertor heat load, and F_{div} is the lifetime limit of the heat fluence on the divertor. The initial blanket cost, C_{blk} and the initial divertor cost, C_{div}, are calculated from the coupled systems analysis.

Annual auxiliary heating costs were assumed to be 10% of the initial capital cost, C_{aux} of the auxiliary heating system. Miscellaneous replacements and fuel costs were 7.5 M$/year.

C_{om} represents the annual operation and maintenance cost scaled as $108{\cdot}(P_e/1200)^{0.5}$ [M$/year]. A decommissioning charge of 1.0 mill/kWh (the second term of Equation (8)) and a waste charge of 0.5 mill/kWh (the third term of Equation (8)) were assumed.

3. Results

To establish viable design points, physics and system parameters were scanned over a set of ranges. Once the plasma and system parameters were given, R_0 and B_T were determined by the method explained in Section 2. A list of scanned parameters for the coupled systems analysis is shown in Table 1 along with other fixed parameters.

For a single systems analysis, only a few minutes are required on a nominal personal computer. However, for the million cases of scanning, a substantially larger computing time is required. We extended the systems code to perform scanning of a large number of cases efficiently on a parallel computer. For the parameters and their selected ranges in Table 1, the input files for the whole cases were generated: a total of 2,700,000 cases (300,000 random calls with nine cases for q_e). Then, the cases were equally distributed over a given number of CPU cores for parallel executions. Using 960 CPU cores of CRAY XC50 (Intel Xeon Cascade Lake 2.4 GHz CPU), the scanning took about 19 h.

During the scanning, there appeared cases either with a solution outside of our interests or without an optimal solution. To avoid the former cases, we applied a series of filters to eliminate unphysical or undesirable solutions: $R_0 < 10.0$ m, auxiliary heating power, $P_{aux} < 150$ MW, $P_{aux} > P_{cd}$, and divertor heat load < 20 MW/m^2. To avoid the latter ones, we set a maximum computing time for a single case. For the total 2,700,000 cases, we obtained more than 300,000 design points.

Table 1. Range of system parameters.

Parameter	Value	Increment
P_{fusion} [MW]	2000–4000	Random
Aspect ratio, A	3.0–3.5	Random
Elongation, κ	2.0	
Triangularity, δ	0.6	
α_N	0.5	
α_T	1.0	
β_N	2.0–5.0	Random
n_e/n_G	0.4–1.5	Random
q_e	3.0–5.0	0.25
H	1.0–1.5	Random
η_{th}	0.3–0.6	Random

The coupled systems analysis produced values for the length, area, volume, and mass of the reactor components. Unit costs used to calculate the cost of the reactor, the coils, and the auxiliary heating system are listed in Table 2. The outboard blanket thickness was fixed at 70 cm. We assumed that all charged particles power was recovered and the wall-plug power to neutral-beam conversion efficiency, η_{wp2NB}, was 0.6. Parameters for the cost calculation are summarized in Table 3.

Table 2. Unit costs used for the cost calculation.

Component	Unit Cost
$C_{reactor}$	
First wall	$6{\cdot}10^4$ USD/m^2
Divertor	$6{\cdot}10^5$ USD/m^2
Blanket	20 USD/kg for PbLi
	90 USD/kg for FMS structure
	100 USD/kg for SCI insert
Shield	40 USD/kg
C_{coil}	
TF coil	$2{\cdot}10^3$ \$/m
PF coil	0.02 USD/(A-m-T)
CS	$1{\cdot}10^3$ USD/m
Case	50 USD/kg
C_{aux}	
NBI	5.3 USD/W

When the CS was used to ramp up the plasma current, Figures 2 and 3 show the variations of the COE (Figure 2) and the capital cost (Figure 3) on R_0, B_T, P_e, η_{th}, n_e/n_G, β_N, A, H, and P_{fusion}, with the design solutions under the conditions: $P_{aux} < 100$ MW, a neutron wall loading < 3.0 MW/m^2, a divertor heat load < 10 MW/m^2, $P_e > 1000$ MW, capital cost < 6000 M\$, and a COE < 140 mills/kWh.

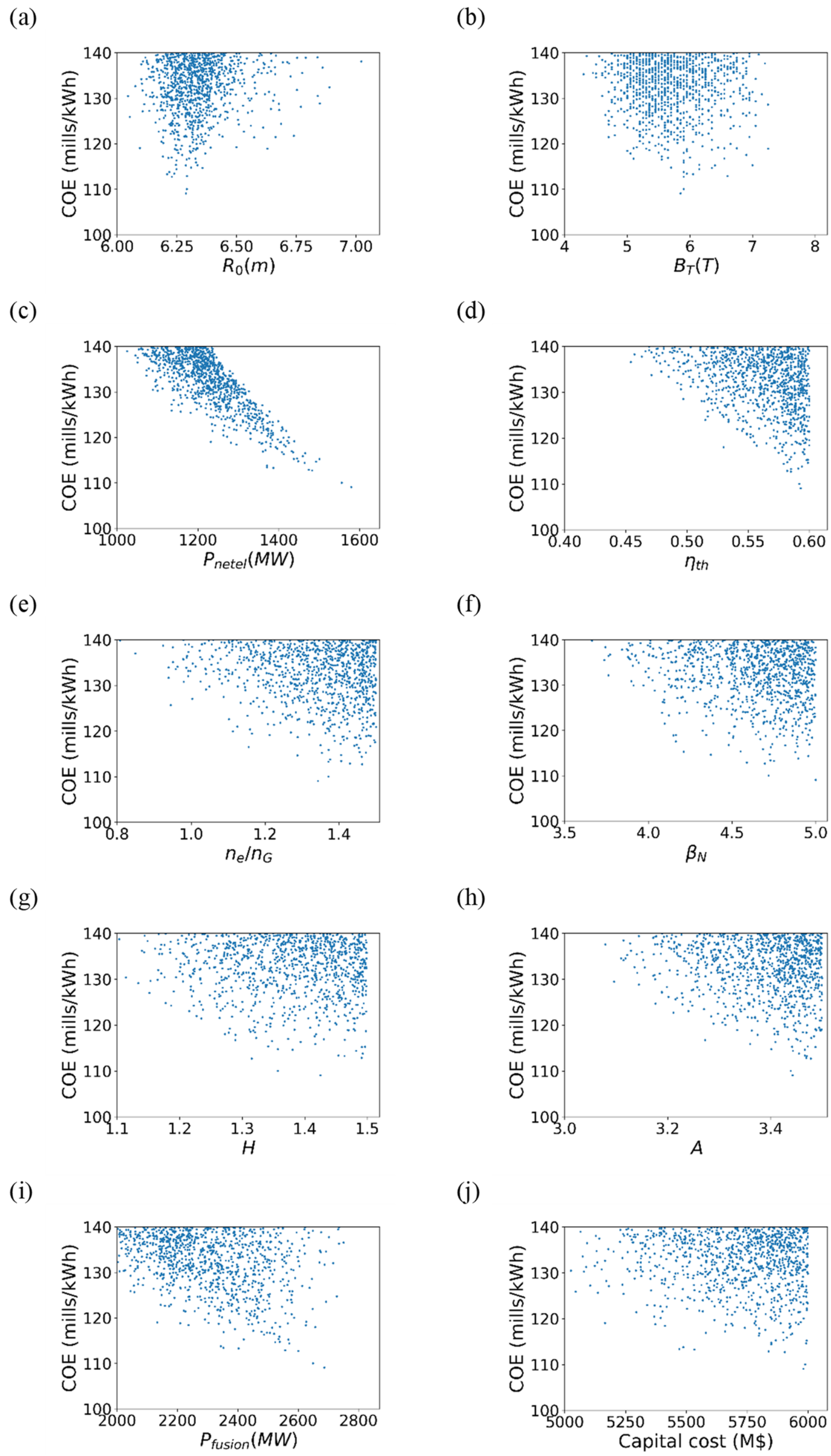

Figure 2. Variation of COE on (**a**) R_0; (**b**) B_T; (**c**) P_e; (**d**) η_{th}; (**e**) n_e/n_G; (**f**) β_N; (**g**) H; (**h**) A; (**i**) P_{fusion}, and (**j**) capital cost with the CS case.

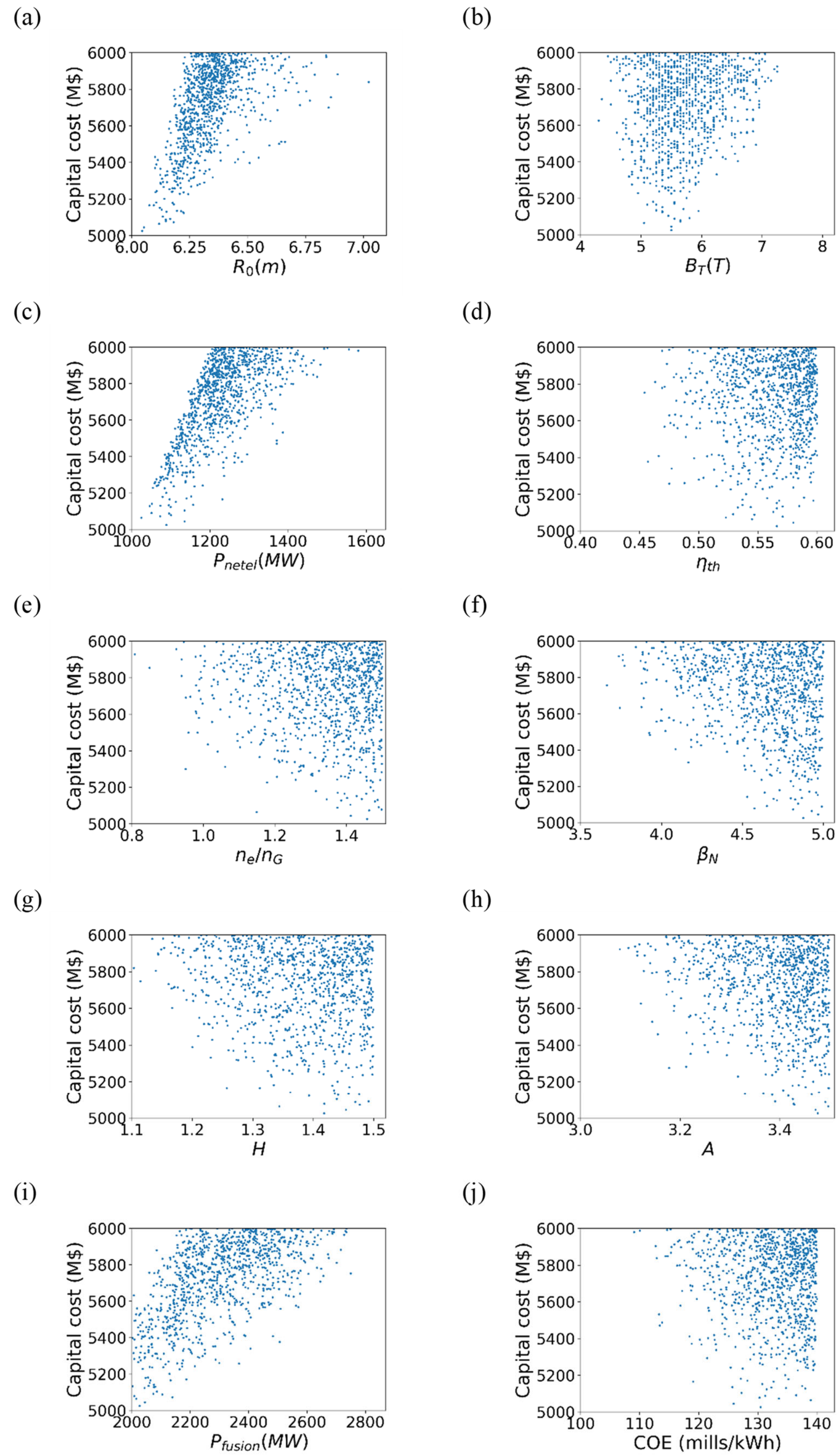

Figure 3. Variation of capital cost on (**a**) R_0; (**b**) B_T; (**c**) P_e; (**d**) η_{th}; (**e**) n_e/n_G; (**f**) β_N; (**g**) H; (**h**) A; (**i**) P_{fusion}, and (**j**) COE for the case with the CS.

Table 3. Costing parameters.

Costing Parameter	Value
Neutron energy multiplication	1.17
Process contingency	0.15
Indirect charge, f_{IND}	0.375
Construction time	6 years
Constant-dollar capitalization factor, f_{CAP0}	1.075
Constant-dollar fixed-charge rate, F_{CR0}	0.1
Lifetime of reactor, N	40 years
Reactor availability, f_{av}	0.75
Blanket fluence limit, F_{blk}	20.0 MW·year/m^2
Divertor heat fluence limit, F_{div}	15.0 MW·year/m^2

In Figures 2 and 3, interesting trends are observed between the costs and the various physics and engineering parameters. Generally, as the COE and capital cost are reduced, the allowed windows of the parameters are narrowed in varying degrees. R_0 was between 6.0 and 7.0 m; for a smaller COE, the range narrowed, with the mean value fixed; for smaller capital cost, the range narrowed, with the mean value shifted to a smaller R_0. The B_T value was between 4.3 and 7.3 T, with B_{max} < 13 T, and the range narrowed for a lower COE and capital cost. The capability of B_{max} = 16 T was not fully implemented. P_e was in a range of 1000 to 1600 MW; for a lower COE, the range narrowed, with the mean value shifted to a larger P_e; for a lower capital cost, the range narrowed, with the mean value shifted to a smaller P_e. The COE was between 109 and 140 mills/kWh and the capital cost was between 5000 and 6000 M$; a larger capital cost was required for a lower COE.

The ranges of the required physics and system parameters were; n_e/n_G > 0.8, β_N > 3.7, and H > 1.1, and the minimum values increased for a lower COE and capital cost; A was in a range of 3.0 to 3.5, and the minimum value increased for a lower COE and capital cost; P_{fusion} was in a range of 2000 to 2800 MW; for a lower COE, the range narrowed, with the mean value shifted to larger P_{fusion}, and for a lower capital cost, the range narrowed, with the mean value shifted to a smaller P_{fusion}; η_{th} > 0.46 was required and the minimum value increased for a lower COE, but the capital cost was not strongly dependent on the η_{th}.

We can narrow down selected parameters further to look at the dependences more closely. Figure 4 shows the variation of the COE on the required physics and the system parameters when we restrict the major radius as $R_0 \leq 6.2$ m for the cases with the CS. The red color represents the COE less than 120 mills/kWh, the blue color represents the COE between 120 and 130 mills/kWh, and the purple color represents the COE between 130 and 140 mills/kWh. More samplings in the scanning parameter ranges were taken. The limiting values of n_e/n_G, β_N, A, and η_{th} increased as the COE decreased, and the gradient became steeper than the case in Figure 2. The COE was marginally dependent on the H, and smaller P_{fusion} than the case in Figure 2 was required. The tokamak fusion reactor with a power level above that of the ITER but with a comparable or less reactor size, and with the COE less than 120 mills/kWh and the capital cost < 6000 M$ seems viable if plasma physics and engineering technology compared to those adapted in the design of the ITER are improved.

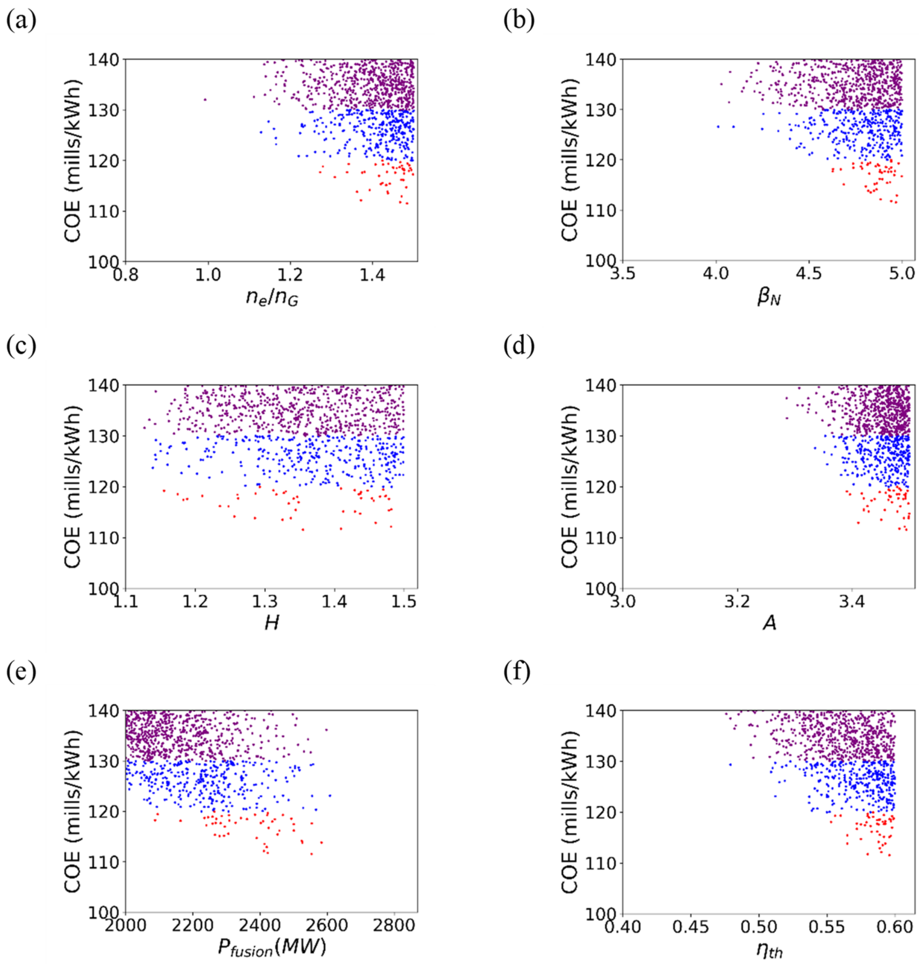

Figure 4. Variation of COE (red for COE $\leq$ 120 mills/kWh, blue for 120 mills/kWh < COE $\leq$ 130 mills/kWh, and purple for 130 mills/kWh < COE $\leq$ 140 mills/kWh) on the required physics and system parameters when $R_0 \leq 6.2$ m for the case with the CS: (**a**) n_e/n_G; (**b**) β_N; (**c**) H; (**d**) A; (**e**) P_{fusion}, and (**f**) η_{th}.

It is generally expected that the costs can be further reduced by removing the CS and introducing an external current drive system. Figures 5 and 6 show the dependences of the COE (Figure 5) and capital cost (Figure 6) on R_0, B_T, P_e, η_{th}, n_e/n_G, β_N, A, H, and P_{fusion} without the CS. Other conditions for the design solutions are set as the same for the cases with the CS: P_{aux} < 100 MW, neutron wall loading < 3.0 MW/m^2, and divertor heat load < 10 MW/m^2, P_e > 1000 MW, capital cost < 5000 M$, and COE < 130 mills/kWh.

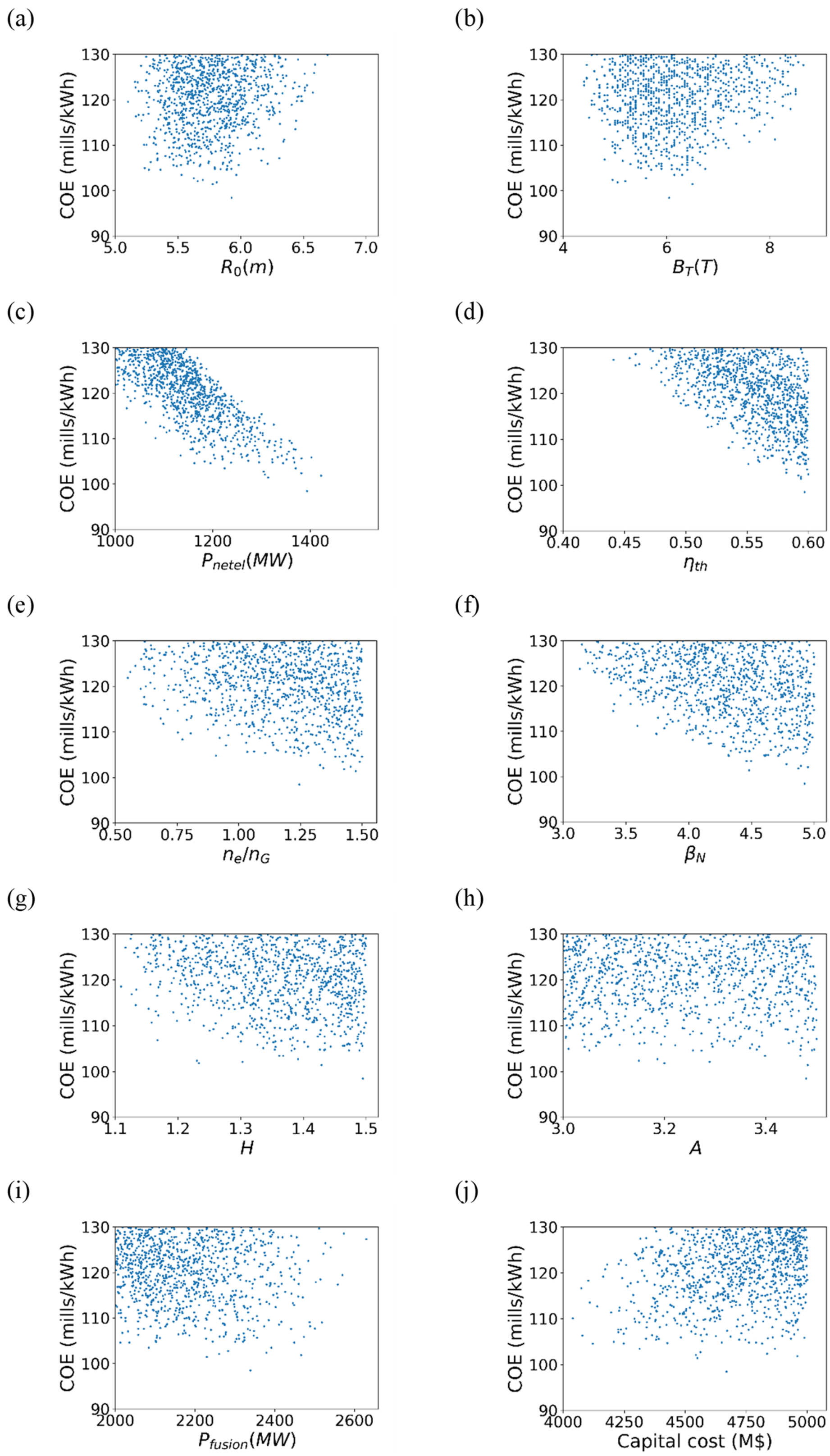

Figure 5. Variation of COE on (**a**) R_0; (**b**) B_T; (**c**) P_e; (**d**) η_{th}; (**e**) n_e/n_G; (**f**) β_N; (**g**) H; (**h**) A; (**i**) P_{fusion}, and (**j**) capital cost without the CS case.

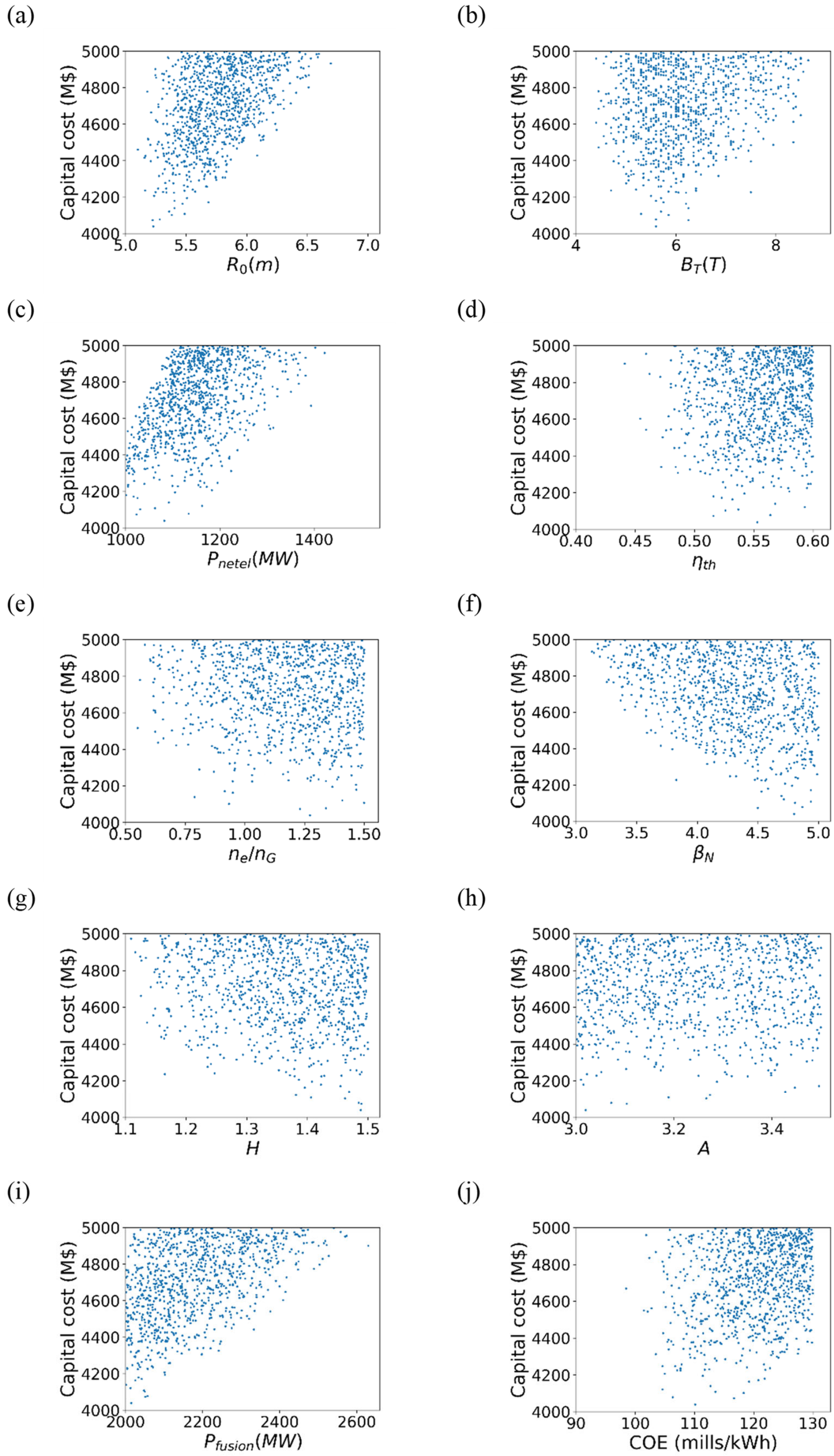

Figure 6. Variation of capital cost on (**a**) R_0; (**b**) B_T; (**c**) P_e; (**d**) η_{th}; (**e**) n_e/n_G; (**f**) β_N; (**g**) H; (**h**) A; (**i**) P_{fusion}, and (**j**) COE for the case without the CS.

Similar trends appear with wider windows for the cost reductions. R_0 was between 5.1 and 6.7 m; for a lower COE, the range narrowed, with the mean value fixed; for a lower capital cost, the range narrowed, with the mean value shifted to a smaller R_0. The B_T value was between 4.4 and 8.7 T, with $B_{max} < 16$ T, and the range narrowed with a smaller B_{max} for a lower COE and capital cost. P_e was between 1000 and 1420 MW; for a lower COE, the range narrowed, with the mean value shifted to a larger P_e; for a lower capital cost, the range narrowed, with the mean value shifted to smaller P_e. The COE was in a range of 98 to 130 mills/kWh and the capital cost was between 4000 and 5000 M$.

The required physics and system parameters ranges were n_e/n_G > 0.6, β_N > 3.1, and H > 1.1, and the minimum values increased for a lower COE and capital cost; A was between 3.0 and 3.5, P_{fusion} was between 2000 and 2650 MW, irrespective of the COE. However, for a lower capital cost, the range narrowed, with the mean value shifted to smaller P_{fusion}; η_{th} > 0.44 is required and the minimum value increased for a lower COE, but the capital cost was only marginally dependent on the η_{th}.

Figure 7 shows the variation of the COE on the required physics and the system parameters when $R_0 \leq 5.5$ m for the case without the CS. The red color represents the COE less than 110 mills/kWh, the blue color represents the COE between 110 and 120 mills/kWh, and the purple color represents the COE between 120 and 130 mills/kWh. More samplings in the scanning parameter ranges were taken. The limiting values of n_e/n_G, β_N, H, and η_{th} increased as the COE decreased, and they were similar in magnitude to the case in Figure 5. The COE had little dependence of A, and smaller P_{fusion} than the case in Figure 5 was required. Thus, when the CS is not used, the design solution exists with smaller R_0, smaller COE, and smaller capital cost, and the mitigated requirements on the physics and the system parameters compared to the case with the CS.

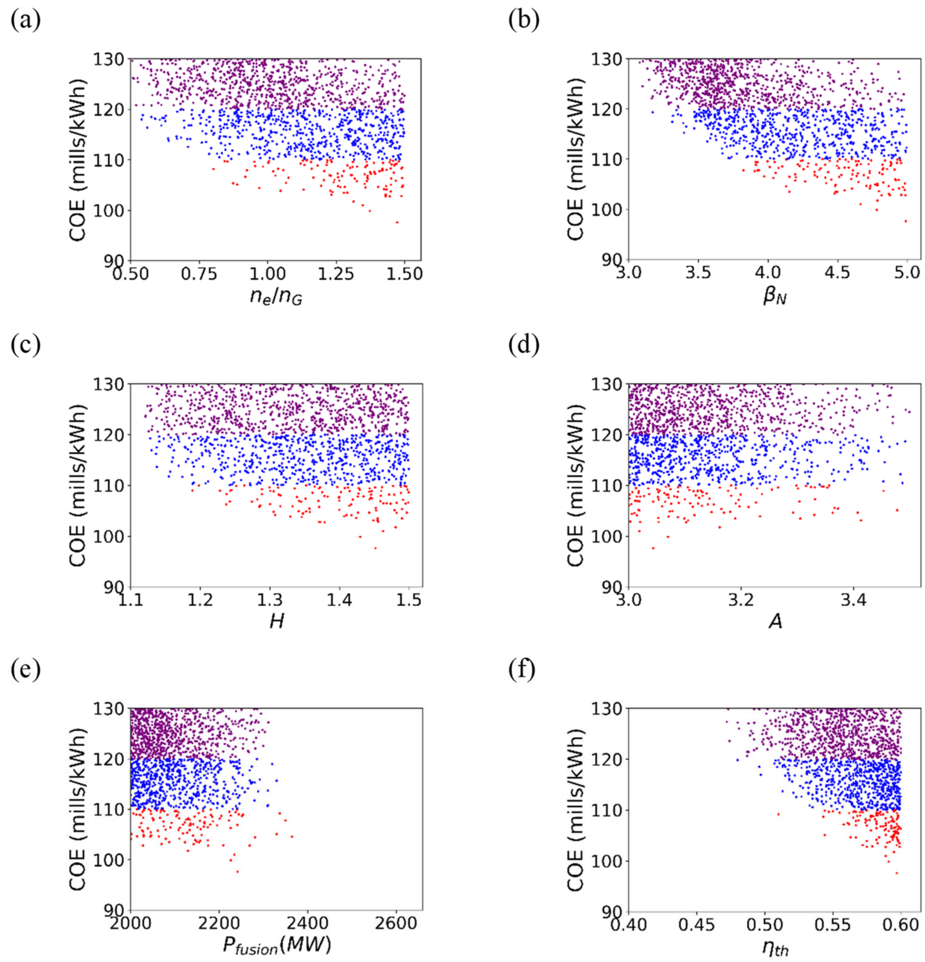

Figure 7. Variation of COE (red for COE $\leq$ 110 mills/kWh, blue for 110 mills/kWh < COE $\leq$ 120 mills/kWh, and purple for 120 mills/kWh < COE $\leq$ 130 mills/kWh) on the required physics and system parameters when $R_0 \leq 5.5$ m for the case without the CS: (**a**) n_e/n_G; (**b**) β_N; (**c**) H; (**d**) A; (**e**) P_{fusion}, and (**f**) η_{th}.

4. Conclusions

We developed and applied an inventive method to find the minimum major radius, R_0, and the optimum build of a tokamak fusion reactor that simultaneously meets all physics, engineering, and neutronics constraints. Parallelly scanning the physics and engineering parameters, we could determine the optimum builds of multiple components for the tokamak fusion reactor, which led to reductions in both the size and cost of the reactor. Improvements in plasma pressure, density, confinement, and divertor heat load, combined with implementation of advanced engineering approaches, such as a maximum allowable magnetic field at the TF coil, advanced materials for blankets and shields, an innovative tritium breeding blanket concept, power handling, and a high-temperature Brayton thermal cycle were required. A tokamak systems analysis coupled with neutron transport calculations and a simple cost model enabled determination of self-consistent system parameters and an optimum build at a minimum cost. The minimum R_0 and the optimum build were determined by the plasma performance, the role of the CS, the maximum allowable magnetic field at the TF coil, $B_{\max}$, and the ripple requirement, while the neutronic requirements for tritium self-sufficiency and shielding determined the thicknesses of the internal components, such as the blanket and the shield.

The impact of the physics parameters as well as the advanced engineering elements on reactor costs were addressed. We looked for a design solution with an auxiliary heating power of < 100 MW, a neutron wall loading < 3.0 MW/m^2, and a divertor heat load < 10 MW/m^2. A total of 2,700,000 cases were performed using the KAIROS supercomputer at the Korea Institute of Fusion Energy.

With the CS, design solutions with a COE between 109 and 140 mills/kWh, direct capital cost between 5000 and 6000 M$, and P_e between 1000 and 1600 MW were possible with a minimum R_0 between 6.0 and 7.0 m, and a P_{fusion} between 2000 and 2800 MW. The required physics and system parameters ranges were $n_e/n_G > 0.8$, $\beta_N > 3.7$, $H > 1.1$, and $\eta_{th} > 0.46$.

With no CS, design solutions with a COE between 98 and 130 mill/kWh, direct capital cost between 4000 and 5000 M$, and a P_e between 1000 and 1420 MW could be found with a minimum R_0 between 5.1 and 6.7 m, and P_{fusion} between 2000 and 2650 MW. The required physics and system parameters ranges were $n_e/n_G > 0.6$, $\beta_N > 3.1$, $H > 1.1$, and $\eta_{th} > 0.44$.

Author Contributions: Conceptualization, B.-G.H., J.-M.K. and H.-K.C.; methodology, G.J. and A.C.; software, G.J. and B.-G.H.; investigation, A.C.; data curation, G.J.; writing—original draft preparation, G.J. and B.-G.H.; writing—review and editing, B.-G.H., J.-M.K. and H.-K.C.; visualization, G.J. All authors have read and agreed to the published version of the manuscript.

Funding: This work was supported by the R&D Program of the Korea Institute of Fusion Energy (KFE), which is funded by the Ministry of Science and ICT of the Republic of Korea (KFE-EN2141-7). This work was also supported by research funds from Jeonbuk National University in 2021 and a National Research Foundation of Korea (NRF) grant under contract No. 2021R1F1A1045388. Computing resources were provided on the KFE computer, KAIROS, funded by the Ministry of Science and ICT of the Republic of Korea (KFE-EN2141-7).

Acknowledgments: This work was also supported by research facilities at the Plasma Application Institute of Jeonbuk National University.

Conflicts of Interest: The authors declare no conflict of interest.

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