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# **System Studies of Spherical Tokamaks**

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# “PROCESS”: Systems Studies of Spherical Tokamaks

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## Abstract

With a reduced aspect ratio, spherical tokamaks have a number of attractive features for a fusion power plant. This can be studied using systems codes which allow for the rapid conceptual study of power plants covering everything from the plasma through to electricity generation. In this paper we describe models in the systems code PROCESS that have been added specifically for spherical tokamaks. Within PROCESS an alternative relation for the plasma current is included which accounts for the increased ratio of  $I_p/aB$ . We have tested this against a series of equilibria created with the free boundary equilibrium code FIESTA and additionally performed our own fit. We also outline the engineering changes that can be made to the device and describe a water-cooled copper centrepost model. To test our models we recreate the published designs for the Fusion Nuclear Science Facility (FNSF) and a High Temperature Superconducting Pilot Plant (HTS-PP) and find good agreement. We conclude by highlighting the efficiencies needed to produce net electricity from small fusion devices.

*Keywords:* Spherical Tokamak, Fusion Reactor, Power Plant, System Studies, PROCESS

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## 1. Introduction

Spherical tokamaks offer a number of potential advantages for a future fusion power plant. They have a high ratio of thermal to magnetic field pressure ( $\beta$ ) and strong flows, either of which could result in reduced turbulence. Fewer toroidal field coils and a different geometry offers the potential for new methods of remote maintenance and lower magnet costs. (For more information see review articles e.g. [1, 2]).

Systems codes can be employed to scope out parameter space quickly by using a set of simplified, yet comprehensive, models to rapidly determine feasible tokamak designs. Spherical tokamaks have a number of differences compared with their conventional aspect ratio counterparts, and in this paper we present the spherical tokamak specific models implemented in the systems code PROCESS. PROCESS has previously been used extensively to study conventional aspect ratio devices such as the European-DEMO [3], CFETR [4] and SST-2 [5].

To model spherical tokamaks an alternative relation between the plasma current and the ratio of the toroidal magnetic field to the safety factor is implemented, to account for an increased ratio of  $I_p/aB$  that can be accommodated at low aspect ratio. We also include the contribution of the diamagnetic current to the overall plasma current, which is higher than in a conventional aspect ratio

device due to the higher  $\beta$ . Various options are available to alter the build of the device; these include the ability to remove the central solenoid and avoid inboard breeding blankets, to join the TF coils to a single centerpost, to reposition the shaping poloidal field coils within the TF coil, and to increase the divertor space. These design modifications are aimed at overcoming the challenges presented, such as the limited inboard space and the increased divertor heat loads.

The rest of this paper is structured as follows. In Section 2 we describe the spherical tokamak specific models within PROCESS in detail and then in Section 3 we apply them to two examples from the literature. We conclude in Section 4. Throughout this work we are using PROCESS version tag: 1.0.16-130-g0d018fd5.

## 2. Spherical Tokamak Models in PROCESS

The physics and engineering models in PROCESS have previously been described in [6, 7]. These papers cover the models that are used for conventional aspect ratio tokamaks. A number of spherical tokamak specific models have also been developed, principally based on [8, 9, 10, 11]. Here we give an overview of these models with further details available in the references.

### 2.1. Inboard Radial Build

One of the challenges of spherical tokamaks is the limited space on the inboard side to fit all the components into. To optimise this, an alternative inboard build can be

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adopted. The primary difference is that the toroidal field coils all join onto a single centrepost that runs through the centre of the device. This carries all of the current and takes less space than individual limbs. Additionally space can be saved by dispensing with the central solenoid and operating in steady state, as well as not having an inboard breeding blanket. Both of these are design choices and PROCESS retains the capability to have them if required.

## 2.2. Centrepost

The default spherical tokamak toroidal field coil magnet model in PROCESS is a water-cooled copper centrepost linked to copper return limbs. PROCESS retains the capability to use low and high temperature superconductors, however these follow the conventional aspect ratio model described in [7] and are individual coils instead joining onto a single centrepost. No model for a superconducting centrepost, such as that proposed by [12], is currently implemented.

The water-cooled copper centrepost is tapered in shape. It is straight from the ends to the height of the plasma, before reducing in thickness to its thinnest point at the midplane following an arc. The maximum radius is given by:

$$r_{\text{top}} = R - \delta a - 3\Delta_{\text{SoL}} - \Delta_{\text{FW}} \quad (1)$$

where  $R$  and  $a$  are the major and minor radius,  $\delta$  is the plasma elongation and  $\Delta_{\text{SoL}}$  and  $\Delta_{\text{FW}}$  are the thicknesses of the scrape-off layer and first wall.

The resistivity of the centrepost is temperature dependent and is parameterised in the range of interest by:

$$\rho = 10^{-8}(1.72 + 0.0039T_{\text{av}})/0.92 \quad (2)$$

where  $T_{\text{av}}$  is the average temperature of the centrepost and the factor 0.92 corresponds to Glidcop [9]. The average temperature of the centrepost is given by:

$$T_{\text{av}} = T_{\text{in}} + \Delta T_{\text{io}}/2 + \Delta T_{\text{film}} + \Delta T_{\text{con}} \quad (3)$$

where  $T_{\text{in}}$  is the inlet coolant temperature (typically 40°C),  $\Delta T_{\text{io}}$  is the temperature rise in the coolant,  $\Delta T_{\text{film}}$  is the temperature rise across the coolant/tube film boundary and  $\Delta T_{\text{con}}$  is the temperature difference in the conductor. These are given by:

$$\Delta T_{\text{io}} = \frac{P_{\text{tot}}}{\rho_{\text{H}_2\text{O}} v_{\text{av}} A_{\text{cool}} C_{p_{\text{H}_2\text{O}}}} \quad (4)$$

where  $\rho_{\text{H}_2\text{O}}$  is the density of water,  $v_{\text{av}}$  the average coolant flow speed,  $A_{\text{cool}}$  the coolant cross-sectional area and  $C_{p_{\text{H}_2\text{O}}}$  the specific heat capacity of water.

The total power ( $P_{\text{tot}}$ ) is given by the sum of the resistive and nuclear heating. The nuclear heating is approximated by assuming a point source at the centre of the plasma. The fraction of neutrons hitting the centrepost is determined from the solid angle and an average path

length of the centrepost diameter at the midplane, with an e-folding length of 0.08m, is assumed for absorption.

The temperature rise across the coolant/tube film boundary is given by:

$$\Delta T_{\text{film}} = \frac{P_{\text{tot}}}{h 2\pi r_{\text{cool}} n_{\text{cool}} l_{\text{cool}}} \quad (5)$$

where  $r_{\text{cool}}$  is the radius of a coolant tube,  $n_{\text{cool}}$  is the number of coolant tubes and  $l_{\text{cool}}$  is the length of the coolant channels. These are being used to determine the surface area of all the tubes.  $h$  is given by:

$$h = \frac{N_{\text{u}} k_{\text{H}_2\text{O}}}{d_{\text{cool}}} \quad (6)$$

where  $d_{\text{cool}}$  is the coolant channel diameter,  $k_{\text{H}_2\text{O}}$  is the thermal conductivity of water and  $N_{\text{u}}$  is the Nusselt number (given by  $0.023 R_{\text{n}}^{0.8} P_{\text{rnd}}^{0.3}$  where  $R_{\text{n}}$  is the Reynolds number and  $P_{\text{rnd}}$  the Prandtl number).

Finally, the temperature gradient in the conductor is estimated using an average distance between the coolant tubes [9]. The average temperature difference in the copper is:

$$\Delta T_{\text{con}} = \left( \frac{P_{\text{tot}}}{2k_{\text{cp}} V_{\text{cp}} (r_0^2 - r_{\text{cool}}^2)} \right) f_r \quad (7)$$

where  $k_{\text{cp}}$  is the centrepost thermal conductivity,  $V_{\text{cp}}$  is the centrepost volume,  $r_0$  is the average distance between coolant tubes and:

$$f_r = r_{\text{cool}}^2 r_0^2 - 0.25 r_{\text{cool}}^4 - 0.75 r_0^4 + r_0^4 \ln(r_0/r_{\text{cool}}) \quad (8)$$

The peak temperature in the centrepost is also computed in order that it is constrained to prevent weakening of the structure. This is given by:

$$T_{\text{max}} = T_{\text{in}} + \Delta T_{\text{io}} + \Delta T_{\text{film}} + \Delta T_{\text{con-max}} \quad (9)$$

where:

$$\Delta T_{\text{con-max}} = \frac{P_{\text{tot}}}{2k_{\text{cp}} V_{\text{cp}}} \left[ \frac{r_{\text{cool}}^2 - r_0^2}{2} + r_0^2 \ln(r_0/r_{\text{cool}}) \right] \quad (10)$$

The pump power is calculated, to be added to the recirculating power, and is given by:

$$P_{\text{pump}} = \frac{\Delta P A_{\text{cool}} v_{\text{max}}}{\eta_{\text{pump}}} \quad (11)$$

where  $v_{\text{max}}$  is the peak flow speed,  $\eta_{\text{pump}}$  is the pump efficiency and  $\Delta P$  is the pressure drop through the pipe given by:

$$\Delta P = f_{\text{fric}} \frac{l_{\text{cool}}}{d_{\text{cool}}} \rho_{\text{H}_2\text{O}} \frac{v_{\text{cool}}^2}{2} \quad (12)$$

$f_{\text{fric}}$  is the friction factor and is taken from [13]:

$$f_{\text{fric}} = \left( -2 \ln \left\{ \frac{r_{\text{gh}}}{3.7} - \frac{5.02}{R_{\text{n}}} \ln \left[ \frac{r_{\text{gh}}}{3.7} + \frac{14.5}{R_{\text{n}}} \right] \right\} \right)^{-2} \quad (13)$$

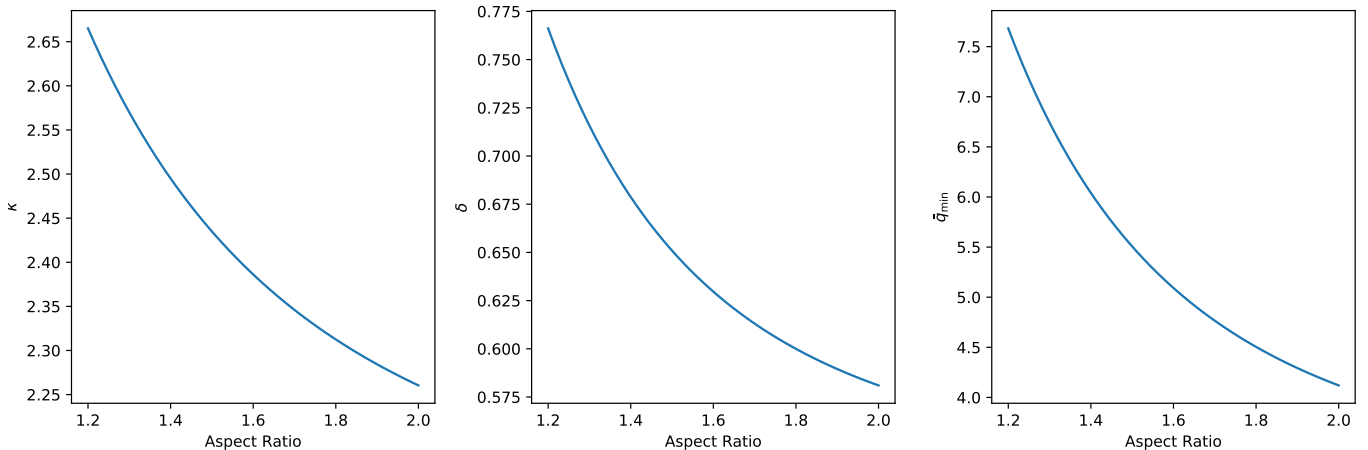


Figure 1: The elongation (left), triangularity (middle) and safety factor (right) with aspect ratio as recommended by [11] for stability.

where the roughness factor is estimated as  $r_{\text{gh}} = 4.6 \times 10^{-5}/d_{\text{cool}}$ .

The required inlet pressure is also calculated to guarantee that the pressure in the tubes remains below the saturation pressure. The saturation pressure is calculated from a fit to [14] using the peak coolant temperature:

$$T_{\text{cool-max}} = T_{\text{in}} + \Delta T_{\text{io}} + \Delta T_{\text{film}} + T_{\text{marg}} \quad (14)$$

where  $T_{\text{marg}}$  is the temperature margin taken as  $10^\circ\text{C}$ .

### 2.3. Divertor

One of the largest challenges for spherical tokamaks is handling the high exhaust heats generated in the divertor. Detailed modelling of divertor configurations remains highly uncertain and experimental campaigns with the new MAST-U Super-X divertor [15] have yet to be conducted. Therefore very simplified models are adopted here and their results are treated with caution.

For spherical tokamaks we use a double-null configuration to spread the heatload to the top and bottom and to minimise the power on the inboard side. PROCESS has previously considered double-null configurations geometrically and recently a simple power sharing model has also been added [16]. The default spherical tokamak divertor model comes from [10] and is a closed divertor with a gaseous target that uniformly radiates within the divertor volume. The heat load on the wall is calculated, however based on the idealised assumption of uniform radiation, these values are low. Realistically this model is used for space allocation in the build and a limit to  $P_{\text{sep}}/R$  is applied to define the allowable heat.

### 2.4. Plasma Shaping

For stability the following options are available for setting the elongation ( $\kappa$ ), triangularity ( $\delta$ ) and minimum edge safety factor ( $\bar{q}$ ) based on [11]:

$$\kappa_x = 2.05(1.0 + 0.44\epsilon^{2.1}) \quad (15)$$

$$\delta_x = 0.53(1.0 + 0.77\epsilon^3) \quad (16)$$

$$\bar{q}_{\text{min}} = 3.0(1.0 + 2.6\epsilon^{2.8}) \quad (17)$$

where  $\epsilon = a/R = 1/A$  is the inverse aspect ratio. These equations are illustrated in Figure 1, however their use does not have to be enforced.

### 2.5. Plasma Current

PROCESS calculates the plasma current based on a relation with the plasma shape and edge safety factor. For conventional aspect ratio tokamaks the ITER Physics Design Guidelines: 1989 [17] are used, however at tight aspect ratios this relation no longer holds. As described in [8], for spherical tokamaks the poloidal field becomes comparable with, or larger than, the toroidal field in the outboard region and is comparable in the inboard region. Meanwhile, the toroidal circumference is comparable to the poloidal circumference in the outboard, but is shorter in the inboard. This leads to highly pitched field lines in the outboard region resulting in only a small amount of toroidal rotation, but moderately pitched field lines in the inboard region resulting in a large amount of toroidal rotation. The overall result in a strongly enhanced total toroidal rotation for a given plasma current, or stated another way, a higher plasma current for a given safety factor. Hence the need for alternative relation that captures the enhancement of  $I_p/aB_t$ .

Peng, Galambos and Shipe [11] proposed such a relation for double-null D-shaped plasmas with  $R/a < 3$ :

$$I_p = \frac{5\kappa a B_t}{2\pi^2 \bar{q}} (F_1 + F_2) \left( \frac{\arcsin(E_1)}{E_1} + \frac{\arcsin(E_2)}{E_2} \right) \quad (18)$$

where  $F_1$  and  $F_2$  are functions of  $\kappa$ ,  $\delta$  and  $\epsilon$  and  $E_1$  and  $E_2$  are functions of  $\kappa$  and  $\delta$  that have not been reproduced here for brevity, but can be found in [11].  $\bar{q}$  is the ‘‘edge’’ safety factor and is related to  $q_{95}$  through:

$$q_{95} = 1.3\bar{q}(1.0 - \epsilon)^{0.6} \quad (19)$$

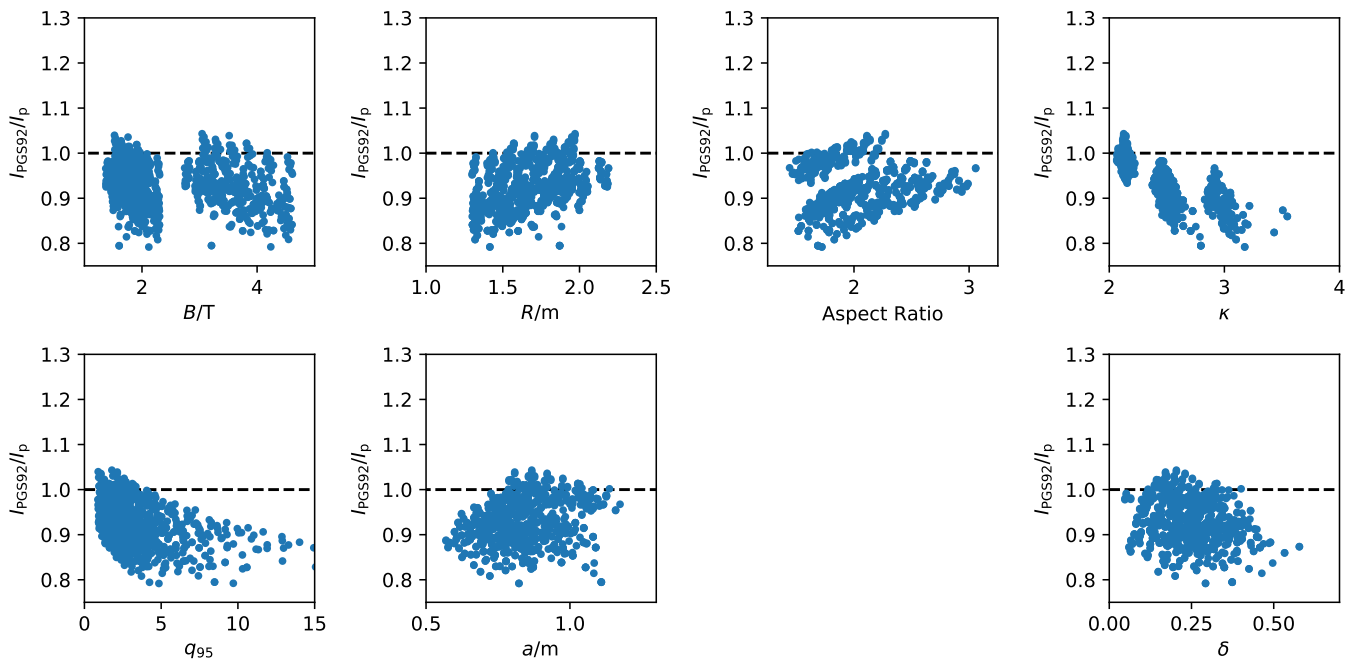


Figure 2: The ratio of the predicted plasma current from Equation 18 [11] to the value from the free boundary equilibria code FIESTA. Each panel plots the equilibria against a different dependency.

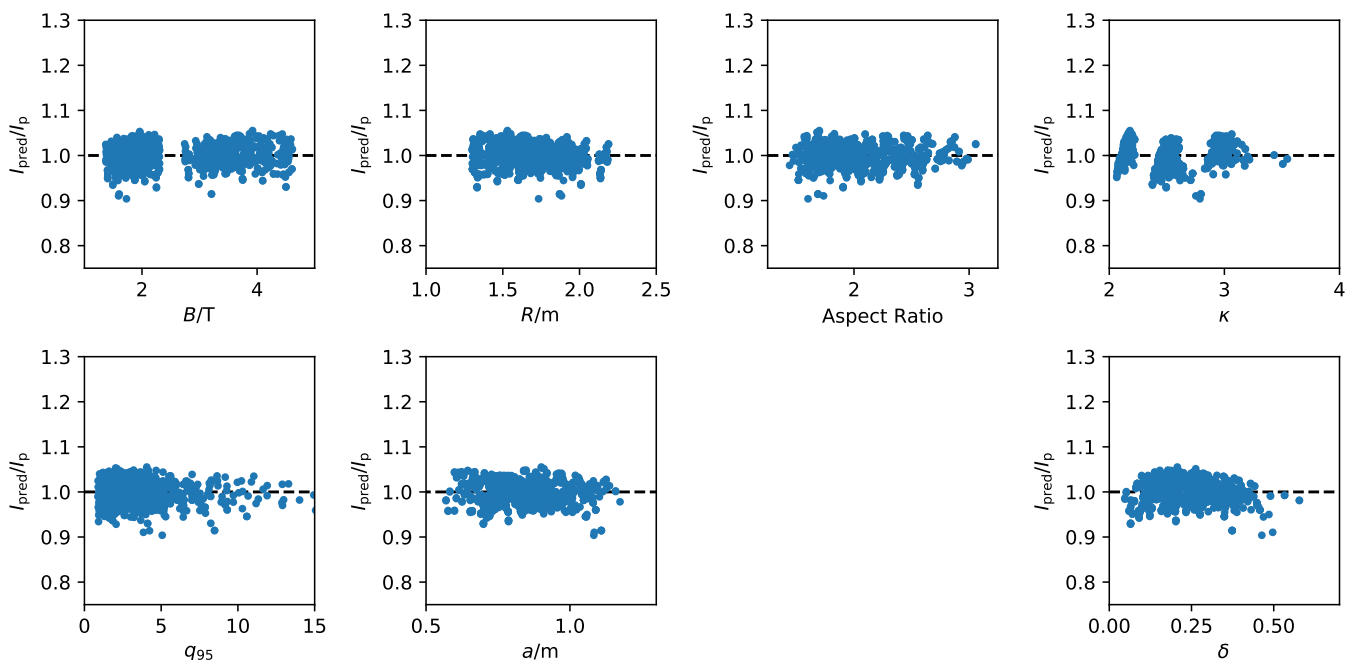


Figure 3: The ratio of the predicted plasma current from Equation 21 to the value from the free boundary equilibria code FIESTA. Each panel plots the equilibria against a different dependency. The four panels on the left represent variables where the fit has been taken from [18] while the three variables on the right have been fitted using the FIESTA equilibria.



To investigate the applicability of Equation 18 we created a series of free boundary equilibria using the code FIESTA. In Figure 2 we calculate the plasma current using Equation 18 and compare it to the value obtained from FIESTA. For the low elongation equilibria, the calculated values for the plasma current are close to those from FIESTA, however moving to higher elongations causes an underestimate by up to 20 per cent.

Given the parameter dependencies illustrated in Figure 2 we chose to generate a new plasma current relation based on fits to our FIESTA equilibria. From [18], the plasma current for a large aspect ratio circular cross-section is given by:

$$I_p = \frac{2\pi a^2 B_t}{\mu_0 R q} \quad (20)$$

The assumptions of large aspect ratio and circular cross-section can be broken by adding functions of  $\epsilon$ ,  $\kappa$  and  $\delta$ . We apply simple power laws giving the following equation to fit to the equilibria:

$$I_p = \left( \frac{a^2 B_t}{R q_{95}} \right) c_1 (1.0 + c_2 \epsilon^{c_3}) \kappa_{95}^{c_4} \delta_{95}^{c_5} \quad (21)$$

Fitting Equation 21 to the equilibria using a non-linear least squares method we find  $c_1 = 3.521$ ,  $c_2 = 2.440$ ,  $c_3 = 2.736$ ,  $c_4 = 2.154$  and  $c_5 = 0.060$ . This fit is illustrated in Figure 3 which shows that there is no bias with any parameter fitted and that the fit is accurate to 10 per cent.

By default spherical tokamaks modelled using PROCESS are designed to operate in steady state. The plasma current is composed of the driven component, the bootstrap fraction and the diamagnetic fraction. For spherical tokamaks we use the Wilson model [19] to calculate the bootstrap fraction and take the diamagnetic fraction as:

$$f_{\text{dia}} = \frac{\beta}{2.8} \quad (22)$$

based on fitting. Given the higher values of  $\beta$  achieved in a spherical tokamak, the diamagnetic fraction is more significant than in a conventional aspect ratio where we use the Sauter bootstrap model [20] by default and neglect the diamagnetic current.

The ratio of the plasma current to the current in the centrepost can be restricted using the constraint:

$$\frac{I_p}{I_{\text{cp}}} < 1.0 + 4.91(\epsilon - 0.62) \quad (23)$$

however this leads ratios above one at aspect ratio below  $\sim 1.6$ , and therefore we usually restrict this ratio further.

## 2.6. Poloidal Field

The poloidal field is given by [11] as:

$$B_p = B_t \frac{F_1 + F_2}{2\pi \bar{q}} \quad (24)$$

where  $F_1$  and  $F_2$  are the same functions of  $\kappa$ ,  $\delta$  and  $\epsilon$  from Section 2.5. This replaces using Ampère's law with the perimeter of the plasma.

Parameter	Ref	PROCESS
Major Radius, $R$ (m)	1.70	1.70
Aspect Ratio, $A = R/a$	1.75	1.75
Elongation, $\kappa$	2.75	2.75
Triangularity, $\delta$	0.5	0.5
Fusion Power, $P_{\text{fus}}$ (MW)	162	162
Auxiliary Power, $P_{\text{aux}}$ (MW)	80	80
Toroidal Field, $B_T$ (T)	3.0	3.0
Plasma Current, $I_p$ (MA)	8.9	9.5
Safety Factor, $q_{95}$	8.9	6.1
Normalised Beta, $\beta_N$	5.5	4.9
H-factor, $H_{\text{IPB98}(y,2)}$	1.25	1.36
Greenwald Fraction, $n/n_{\text{GW}}$	0.75	0.77

Table 1: Selected parameters for FNSF from PROCESS compared to those given in reference [21].

## 2.7. Poloidal Field Coils

A simple PF coil model was proposed in [8] that takes advantage of the natural elongation of spherical plasmas and the typical shapes associated with the relations described in Section 2.4 and Figure 1. A pair of coils are positioned top and bottom for shaping (SF), and in the vertical build these are located inboard of the TF coil. A second pair of coils are located radially outside the TF coil and are used to generate the vertical field (VF). Their currents are set by:

$$I_{\text{SF}} = 0.3A^{1.6}I_p \quad (25)$$

$$I_{\text{VF}} = -0.4I_p \quad (26)$$

The conventional model can also be used that positions all the coils outside of the TF coil.

## 3. Benchmarking

In order to benchmark our models to test their applicability, we have chosen to compare with the proposed Fusion Nuclear Science Facility (FNSF) and a High Temperature Superconducting Pilot Plant (HTS-PP) described in [21, 22, 23].

### 3.1. Fusion Nuclear Science Facility (FNSF)

FNSF is a proposed  $R = 1.7$  m,  $A = 1.75$  and  $P_{\text{fus}} = 162$  MW device with resistive TF coils that will provide a nuclear environment to develop fusion materials and components [21]. In Table 1 we list some of the key parameters and compare them to our PROCESS run.

Overall PROCESS reproduces FNSF well, however the most noticeable difference is the plasma current and safety factor. PROCESS finds a higher plasma current and lower safety factor and this remains regardless of whether Equation 18 or 21 is used to calculate it. Running PROCESS

Parameter	Ref	PROCESS
Major Radius, $R$ (m)	3.0	3.0
Aspect Ratio, $A = R/a$	2.0	2.0
Elongation, $\kappa$	2.5	2.5
Triangularity, $\delta$	0.6	0.6
Fusion Power, $P_{\text{fus}}$ (MW)	500	500
Auxiliary Power, $P_{\text{aux}}$ (MW)	50	52
Toroidal Field, $B_{\text{T}}$ (T)	4.00	4.08
Plasma Current, $I_{\text{p}}$ (MA)	12	11.3
Normalised Beta, $\beta_{\text{N}}$	4.00	3.80
H-factor, $H_{\text{IPB98(y,2)}}$	1.8	1.50
Greenwald Fraction, $n/n_{\text{GW}}$	0.8	0.81

Table 2: Selected parameters for HTS-PP from PROCESS compared to those given in reference [23].

with Equation 18 instead of 21 yields the same plasma current, but a lower safety factor to achieve it.

We have performed this run using the water-cooled centrepost model described in Section 2.2. The model finds an average temperature in the centrepost of  $73^{\circ}\text{C}$  with a resistive loss of 83 MW and a required pumping power of 0.1 MW.

### 3.2. High Temperature Superconducting Pilot Plant (HTS-PP)

HTS-PP is a proposed  $R = 3.0\text{ m}$ ,  $A = 2$  and  $P_{\text{fus}} = 500\text{ MW}$  device, with HTS TF coils, that has a high neutron fluence and will be tritium and electrically self-sufficient [21]. In Table 2 we list some of the key parameters and compare them to our PROCESS run. The main disagreement is again the plasma current, however this time PROCESS finds a lower value. The H-factor is also lower for the PROCESS run. It should be noted that although the Greenwald fractions are the same, the densities are not because of the differing plasma current affecting the normalisation.

HTS-PP highlights the need for efficiency gains for low fusion power plants to be viable. Taking the energy multiplication in the blanket as 1.269 and a thermal to electric conversion efficiency of 0.375, which are typical for the European-DEMO, then the gross electrical power is only 238 MW. With a neutral beam wall plug efficiency of 0.3 the auxiliary power system alone is using 172 MW leaving very little to power the rest of the device, let alone produce the target 100 MW net electric output.

## 4. Conclusions

In this paper we have presented the spherical tokamak specific models in PROCESS. These include alternative relations for the plasma current, one of which is presented here for the first time. Additionally we have detailed the many engineering options including a water-cooled centrepost for the TF coil system. We have applied our models

to two reference cases to demonstrate their impact, illustrating the need to accurately predict the plasma current required. This is important, especially for small machines, as it will impact the amount of auxiliary current drive required and in turn the recirculating power, and hence the net electrical output of any pilot power plant.

Efficiencies play a large role in dictating the required fusion power of a power plant. The net electric output can be defined based on a target market, whether that is a pilot plant, something comparable to a small modular reactor or a full scale power plant. However recirculating power does not scale linearly with fusion power and hence for smaller devices it is essential this is minimised. Identifying methods of operating at high confinement is an option, however if this is not achieved then minimising losses is the only solution. Once the required fusion power for a given output is defined, only then can the material and exhaust challenges be identified that will set the size of the device.

## 5. Acknowledgements

To obtain further information on the data and models underlying this paper please contact PublicationsManager@ukaea.uk.

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