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Conceptual design workflow for the STEP Prototype Powerplant

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ABSTRACT

The STEP (Spherical Tokamak for Energy Production) Programme aims to deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion. To deliver on this aim, we have performed initial scoping to identify the design point for a spherical tokamak prototype powerplant producing at least $100\,\mathrm{MW_e}$ of net electric output and that is self-sufficient in tritium production. We have developed a rapid workflow that starts with the systems code PROCESS to develop an initial design point that can then be iterated with the plasma codes JETTO and FIESTA to add fidelity. Promising solutions can then be taken forward for further analysis of the tritium breeding ratio, inboard build, exhaust and power balance. Solutions that are best matched to the requirements are then taken on by a wider team to assess. We have applied this technique using a number of different input technology options and assumptions and arrived at the most promising design point for the STEP Prototype Powerplant (SPP). The unique features of this device are no inboard breeding blanket, the use of remountable magnet joints, high β and bootstrap plasma operation, a Super-X divertor, the use of just microwave heating, and enclosing poloidal field coils inside the toroidal field cage.

1. Introduction

Fusion has the potential to be a source of safe, sustainable, low-carbon energy. In recognition of the major role fusion could play in energy generation internationally, in 2021 the UK Government released a UK Fusion Strategy [1]. The two overarching goals of the strategy were:

1. For the UK to demonstrate the commercial viability of fusion by building a prototype fusion power plant in the UK that puts energy on the grid.
2. For the UK to build a world-leading fusion industry which can export fusion technology around the world in subsequent decades.

Aligned with this strategy, the STEP (Spherical Tokamak for Energy Production) programme began in 2019 aiming to deliver a UK prototype fusion energy plant, targeting 2040, and a path to commercial viability of fusion.[1] In 2022, West Burton in Nottinghamshire, UK was selected to be the site of the STEP Prototype Powerplant (SPP) [2]. STEP aims to build on the experience of START, MAST and MAST-U to produce a spherical tokamak fusion powerplant.

Spherical tokamaks have a number of features that could yield a more cost effective fusion energy powerplant (see review article by Ono & Kaita [3]). High β (ratio of the plasma pressure to the magnetic pressure) and high bootstrap current fraction gives a more efficient plasma; that allows for the elimination of the need for inductive current drive giving steady-state operation. With a smaller major radius and higher natural elongation, a smaller overall footprint and diameter of the outer poloidal field coils should bring down the capital cost relative to more conventional aspect ratio tokamaks. Additionally, the simpler geometry of a spherical tokamak should enable different maintenance approaches. Based on these features, a number of spherical tokamak powerplants have been previously proposed in the literature [4–8].

An important starting point in designing the SPP was to identify the Measures of Effectiveness (MoEs). These defined the core needs that should be met, which allowed more detailed requirements to be formulated around. The MoEs identified for STEP were:

- Safety and Environment
- Confidence in delivering $100\,\mathrm{MW_e}$ net electric output
- Confidence in delivering tritium self-sufficiency
- Maintainability
- Development Flexibility
- Schedule
- Cost

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[1] https://step.ukaea.uk/

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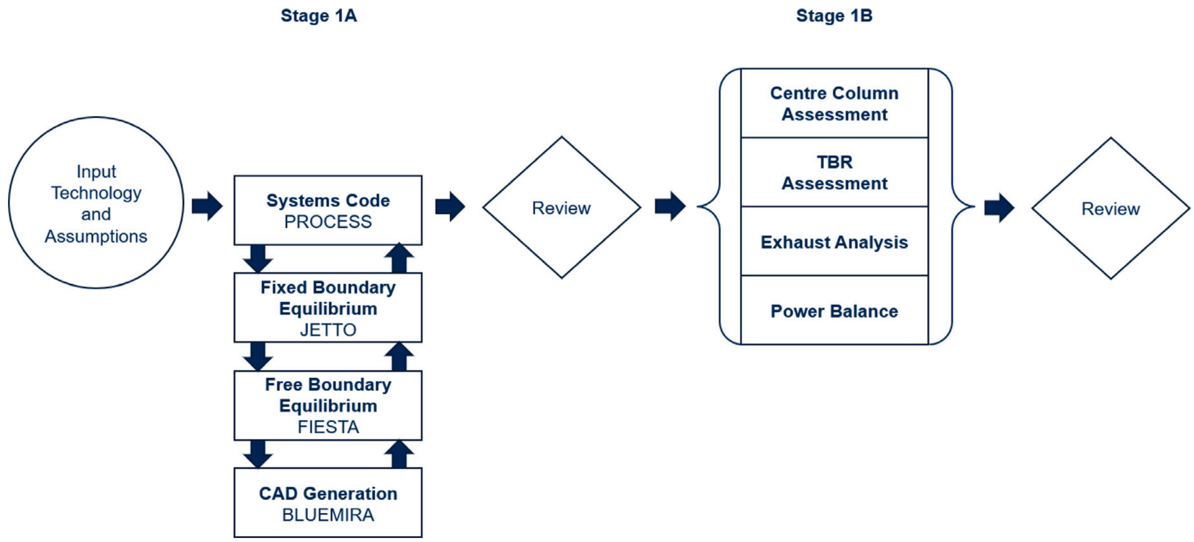

Fig. 1. An overview of the workflow developed for STEP to rapidly and iteratively produce integrated concepts. The workflow is split into two stages. Stage 1A uses a systems code and fixed and free boundary equilibrium codes to create the initial point, while Stage 1B is around further engineering assessment.

Firstly, all fusion projects must ensure safety. $100\,\mathrm{MW_e}$ net electric output was chosen to show meaningful electricity production, while also recognising that the SPP is a prototype pilot plant. Tritium self-sufficiency is a requirement given the limited availability of tritium [9]. Given the first-of-a-kind nature of the project and the uncertainty that needs to be managed, maintainability and flexibility are required. Finally, schedule and cost must be considered from the start as a measure of deliverability.

To guide the design process, STEP has adopted the Concept Maturity Level (CML) structure developed by NASA [10]. Starting with a 'Cocktail Napkin' (CML1) level sketch, 'Initial Feasibility' is then explored (CML2). The 'Trade Space' (CML3) is then explored to identify different options before a 'Point Design' (CML4) is settled on and then developed into a 'Baseline Concept' (CML5).

This paper will explore the trade space exploration towards a point design (CML 2-4). We will start by describing in Section 2 the workflow we developed to quickly scope initial concepts. In Section 3 we will summarise the design space we explored, before describing the point design in Section 4.

2. Design workflow

Within fusion powerplant design, there is a need for a rapid, integrated and iterative design process (see e.g. [11]). Integration is required from the start as systems cannot be designed in isolation, otherwise their competing requirements will lead to infeasible designs. Additionally, some technologies may not be compatible and these need to be identified in a rapid manner, so time is not wasted developing concepts that will not work.

The workflow developed for STEP is shown in Fig. 1. It was devised by identifying what are the quickest assessments that can be done to show there is a feasible concept, while also noting the dependencies different analyses have. This separated the workflow into two stages which will be discussed further in following subsections.

Prior to starting the workflow, a common set of decisions and assumptions must be set. These include discrete technology options that we want to explore and the design limits that will be imposed. Once these are set and agreed, an initial design point can be developed.

2.1. Stage 1A

The goal of Stage 1A is to produce an initial concept based on the input technology options and assumptions, and assess whether the concept is viable to take forward. This is achieved by focusing on the link between the plasma and engineering.

The initial step is to use the systems code PROCESS [12–14] to develop an integrated design point. PROCESS uses a set of simplified relations to model the interaction of the entire fusion powerplant; from plasma to electrical output, including a cost model. The code is linked to an optimiser allowing a figure-of-merit to be set, such as minimise the major radius, which PROCESS will solve for obeying the input constraints. PROCESS has recently been open-sourced and is available on GitHub [15]. Once a design point has been output from PROCESS, it is assessed to evaluate whether it should be developed further.

If a design is taken forward from PROCESS, the next step is to explore the plasma in more detail through fixed-boundary equilibrium using the code JETTO [16]. JETTO is a 1.5 D transport code which is capable of solving the plasma diffusion equations averaged over magnetic surfaces in a time dependent axisymmetric configuration.

Prior to the application of the design workflow, PROCESS and JETTO were bench-marked to make sure that the simplified output from PROCESS made for a suitable input point to JETTO. Both codes gave a consistent solution, however the real strength of using JETTO is in the physics not captured by PROCESS: such as having an accurate q-profile (safety factor) compared to a parameterised version through two points, and heating and current drive deposition locations, hence giving more accurate current drive efficiency.

Based on the PROCESS and JETTO output, the next step of Stage 1A is to perform a free-boundary equilibrium calculation using FIESTA [17]. This allows for the plasma boundary to be more accurately determined and for the Poloidal Field (PF) coil currents to be calculated. To aid with the placing of PF coils, keep-in zones are defined in which the coils can be placed without clashing with other structures.

Once completed, the findings from the equilibrium calculations are fed back into PROCESS for a new iteration, aligning the output of all the codes. BLUEMIRA [11,18] is also used to generate a simplified CAD geometry. The design point is then evaluated against the MoEs listed in Section 1. Those that show the most promise of meeting the MoEs are progressed to Stage 1B, while those that do not meet the MoEs are parked.

2.2. Stage 1B

Stage 1B differs from Stage 1A as activities happen in parallel and are only dependent on the input from Stage 1A. Four activities were identified as being required to give confidence in a concept.

The centre column is one of the most congested and challenging areas in a spherical tokamak, due to the drive to minimise major radius while the low aspect ratio naturally constrains the inboard size. An assessment of the Toroidal Field (TF) coils ability to handle the current and forces needs to initially be checked. Additionally the level of shielding needs to be assessed to determine the lifetime of the centre column and heat handing capability.

In the drive to minimise size, in the SPP we have dispensed with inboard tritium breeding and rely solely the outboard blanket for breeding. This makes fuel self-sufficiency extremely challenging and therefore the Tritium Breeding Ratio (TBR) needs to assessed early in the design. At this stage, it is done in a parametric way to consider different breeding materials and with assumptions about the likely loss of coverage due to ports and penetrations.

The divertor can be a major size driver in fusion powerplant concepts such as the EUROfusion-DEMO design [19]. Therefore an analysis of the exhaust needs to be made beyond the basic metrics imposed in PROCESS such as P_{sep}/R or $P_{sep}B/q_{95}AR$, where P_{sep} is the power crossing the separatrix, R is the major radius, B is the toroidal field, q_{95} is the safety factor on the magnetic surface which encloses 95% of the magnetic flux of the confined plasma, and A is the aspect ratio. For the SPP alternative divertor designs, such as the Super-X, are being considered, therefore this also makes a simple metric more complex to use.

The final parallel study shown in Fig. 1 is a further analysis of the power balance. PROCESS uses a set of high level conversion and wall-plug efficiencies to compute the first pass at the power balance. Using an open-source power balance model developed for STEP [20], more detailed analysis can be performed such as linking efficiencies to the primary coolant outlet temperature. A full description of the model is given in Petrov et al. [21].

2.3. Review

Designs are reviewed against the MoEs described in Section 1. For designs that have made it through the analysis and continue to perform well in the reviews, these can be taken on for further study. The Stage 2 design procedure takes the form of sprints around each design looking at unique challenges. It is therefore not a singular design approach.

The application of this workflow is not only iterative within the cycle of analysing a design point, but also from concept to concept. The learning of what technology options work together to achieve the MoEs, and which do not, can be combined into the inputs of the next concept. This allows for better initial seeds and helps to develop towards the point design that will be taken forward.

An example of this learning is with cryo-resistive TF coils. For low TF field strength designs ($\sim$ 2 T), taking advantage of the spherical tokamak's high β, compact designs achieving the MoEs can be produced (c.f. [4,5]). If the TF strength is required to increase, however, then the cryogenic power requirements increases rapidly to offset the higher TF coil current. This leads to very large cryoplants being required and significant increases in fusion power to offset the cryoplant power demand.

3. Trade space exploration

To explore the trade space, first the input technology options need to be decided. A set of major architecture defining decisions were agreed, corresponding to the items in the top half of Table 1.

For the plasma, key decisions surrounded the triangularity, plasma confinement, heating and current drive, and divertor configuration. Negative triangularity has been shown to give higher levels of confinement for L-mode plasmas and therefore is an attractive option for fusion powerplants (see e.g. [22]). Other modes of confinement were also explored such as I and H. Heating and current drive options such as Neutral Beam Injection (NBI), Electron Cyclotron (EC) and Electron Bernstein Waves (EBW) were considered, both in terms of current drive efficiency and wall-plug efficiency. Alternative divertor configurations were explored within the space of machine, in addition to whether to use a single or double-null.

Table 1
The point design parameters for the 2023 CML4 STEP Prototype Powerplant.

Triangularity (Sign)	Positive
Plasma edge	Edge Pedestal
Heating & current drive mix	Electron Cyclotron & Electron Bernstein Waves
Primary divertor configuration	Dynamic Double-null
Secondary divertor configuration (Inboard)	Flat Top: X Type Ramp Up: Perpendicular
Secondary divertor configuration (Outboard)	Extended Leg
Toroidal field conductor type	REBCO
Tokamak morphology (Radial Build)	Plasma/Wall/Blanket/Vessel
Primary maintenance access route	Vertical
Remountable toroidal field coils	16 TF coils 3 Remountable Joints per TF
Peak steady state divertor heat flux	<20 MW/m^2
Tritium breeder material	Liquid Lithium
Centre column coolant	Water
Divertor coolant	Water
Outboard first wall, blanket, outboard limiter coolant	Helium
Blanket coolant outlet temperature	600 °C
Direct or indirect cycle	Indirect
Fusion power	1.6–1.8 GW
Net electric output	100–200 MW$_e$
Inboard build	1.6 m
Major radius	3.6 m
Toroidal field	3.2 T
Elongation	2.93
Triangularity	0.5

Magnet technology is a key driver for the device. Spherical tokamaks can operate at high β, therefore there is potential for a lower field on axis compared with conventional aspect ratio devices. Another key consideration is the shielding requirement for the centre column magnet, as this additionally contributes to the device size. For scoping, we considered cryogenic aluminium as well as low and high temperature superconductors.

The other major architectural decisions surround the breeding blanket, coolant and outlet temperature. This system is intrinsically linked to the two main aims of tritium self-sufficiency and net electric output. The impact of different outlet temperatures was considered through its affect on thermal-to-electric conversion efficiency.

Fig. 2 gives a flavour of the different outputs generated for exploration. The wide spread of the data is caused by the large range of parameters and technology options being considered. Despite the spread in values, many of these design points are aimed at producing 100 MW$_e$ net electric output as shown in Fig. 3. The purpose of generating these concepts is to understand the key drivers in order to aid downselection to a point design.

A variety of sized devices were created in the trade space exploration. The JETTO simulations used predictive temperature and density. Heating power densities were prescribed and the current drive was calculated empirically using the local temperature, density and Z_{eff}. A Bohm/gyro-Bohm transport model was used with a global scale factor in feedback control in order to fix β_N to a target value. The Greenwald fraction was kept fixed through feedback on the pellet source rate. Primarily a simplified impurity model was used with a constant Z_{eff} prescribed throughout the plasma. Where more sophisticated impurity modelling was performed a combination of Argon and Xenon was used.

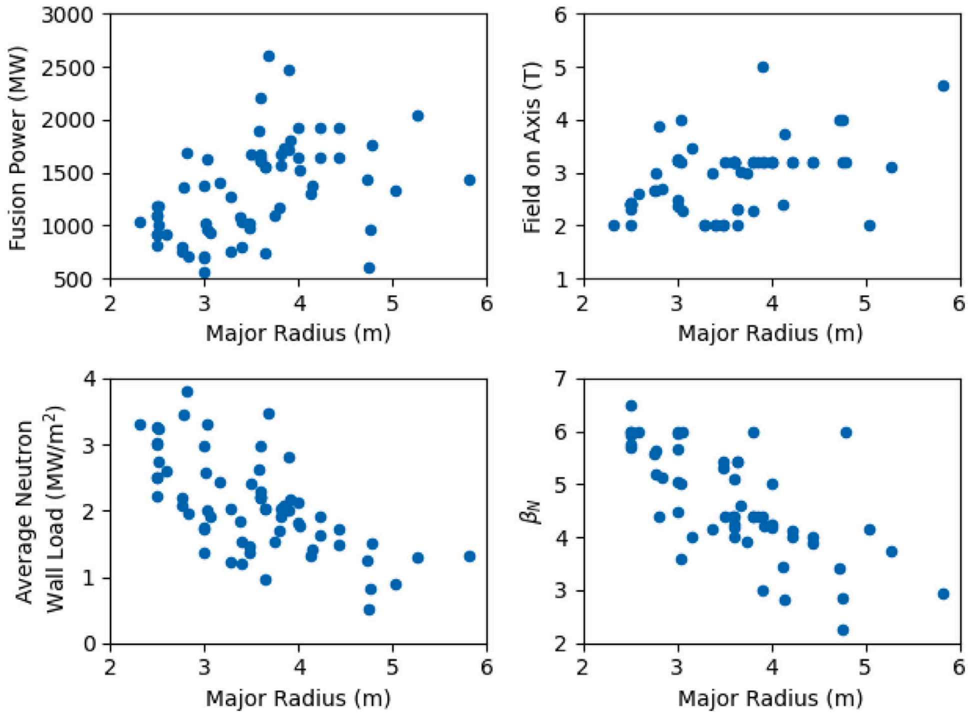

Fig. 2. Plots giving an illustration of the parameter space covered in relation to geometric major radius: Top-left fusion power, top-right toroidal field on axis, bottom-left average neutron wall load and bottom-right nomalised β. The large scatter present in this figure is due to the range of parameter constrains and input technologies being considered. The goal of which is to identify a starting point for further study.

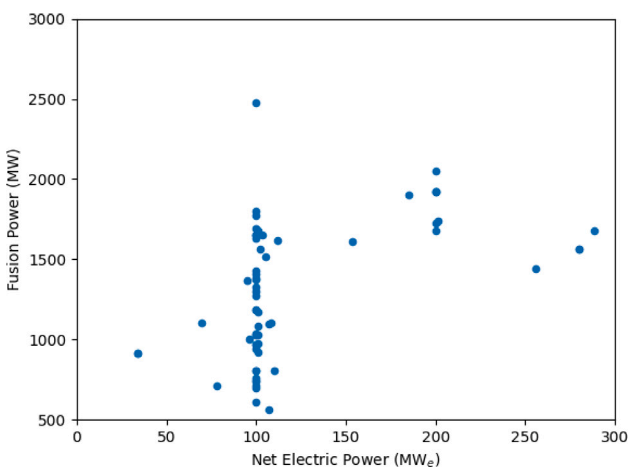

Fig. 3. Fusion power in relation to the net electric output of the concept. Most points are clustered around 100 MW_e as this was the chosen design point. It should be noted that a large range of fusion powers can achieve this depending on the input assumptions used.

4. Point design

Based on the trade space exploration performed in Section 3, Table 1 gives the point design parameters adopted for the STEP Prototype Powerplant.

The main size driver of the device is the inboard build, with the minimum size setting the minimum major radius. A variety of sized devices were created in the trade space exploration, as shown in Fig. 2, however after investigation 1.6 m was the minimum viable inboard build to include all components after detailed sizing. This set the major radius of the device at 3.6 m for an aspect ratio of 1.8, chosen to maximise the available inboard build space while keeping the major radius to a minimum and not requiring inboard breeding.

EC and EBW were chosen for heating and current drive due to the large nature of NBIs being unsuitable. EBW has higher current drive efficiency than EC, so gives the opportunity to reduce the auxiliary power and increase the net electric output, as shown in Fig. 4. EBW will not penetrate the centre of the plasma and therefore cannot be used alone, hence both EC and EBW are needed for this scenario. EC and EBW have competing field requirements with EC needing a higher toroidal field and EBW a lower field to couple to the plasma. To allow both options, a field of 3.2 T on the geometric major radius was selected which is within the overlapping window for the plasma density. EBW will be explored on MAST-U, however to mitigate the risk, scenarios with EC only and EC and EBW have been developed. Both of these scenarios deliver on the 100 MW_e net electric output requirement, as shown in Fig. 4.

After setting the field requirement on the plasma, High Temperature Superconducting REBCO was chosen as the magnet technology to deliver this. Double-null with a Super-X outer leg divertor was chosen to manage the heatload, with double-null pushing more power to the outboard side. An edge pedestal was chosen for the plasma, as negative triangularity was found to be not vertically stable at high elongation. To maximise tritium breeding and net electric output, a helium cooled liquid lithium breeder blanket was chosen, with outlet temperature of 600 °C.

5. Conclusions

In this paper we have outlined the design workflow used to develop the STEP Prototype Powerplant. This consisted of two stages, initially Stage 1A focused on the plasma-machine integration through systems and equilibrium codes. Stage 1B is a parallel workflow that focused on the wider engineering. This workflow has been implemented by running the codes discretely, however long term this workflow could be automated into a framework, such as BLUEMIRA. BLUEMIRA can already run PROCESS and generate CAD, while the free-boundary equilibrium is being improved currently. The next phases would be to include fixed-boundary equilibrium and neutronics.

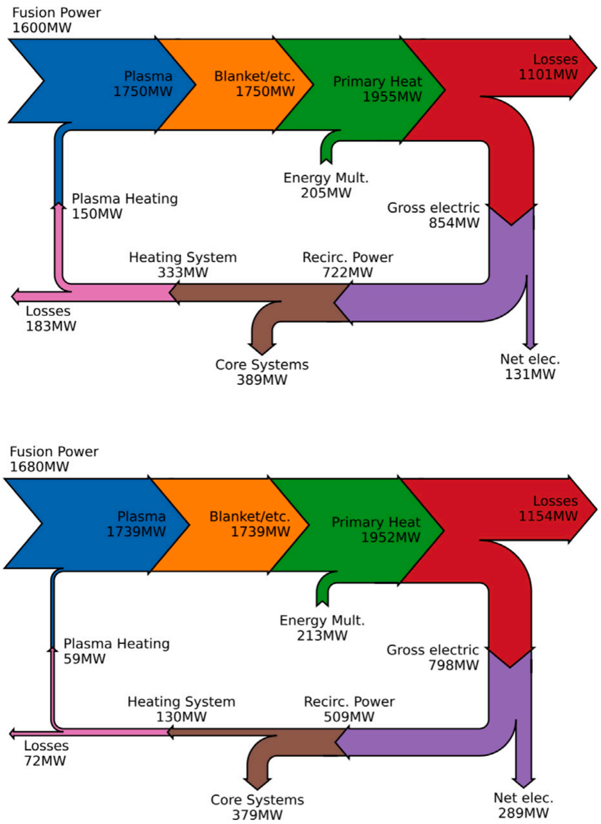

Fig. 4. Sankey diagrams illustrating the power flow for two different plasma solutions. The top case corresponds to Electron Cyclotron heating and current drive only. The bottom case used Electron Cyclotron and Electron Bernstein Wave heating, taking advantage of the latter's increased current drive efficiency.

The outcome of applying these tools is the point design described in Table 1. The unique features of this device are no inboard breeding blanket, the use of remountable magnet joints, high β and bootstrap plasma operation, a Super-X divertor, the use of just microwave heating and enclosing poloidal field coils inside the toroidal field cage. This is just the starting point of a more detailed design procedure and the design will continue to evolve over the coming years.

CRediT authorship contribution statement

Stuart I. Muldrew: Conceptualization, Formal analysis, Methodology, Software, Supervision, Visualization, Writing – original draft, Writing – review & editing. **Chris Harrington:** Conceptualization, Methodology. **Jonathan Keep:** Conceptualization, Methodology. **Chris Waldon:** Conceptualization, Methodology. **Christopher Ashe:** Formal analysis, Software, Visualization. **Rhian Chapman:** Formal analysis, Software. **Charles Griesel:** Formal analysis, Software. **Alexander J. Pearce:** Formal analysis, Software. **Francis Casson:** Formal analysis, Methodology, Software, Supervision. **Stephen P. Marsden:** Formal analysis, Software, Writing – review & editing. **Emmi Tholerus:** Formal analysis, Software.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

PROCESS and BLUEMIRA codes used in this paper are available on GitHub by following the links in the paper.

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References

[1] UK Government, Towards fusion energy: the UK fusion strategy, 2021, https://www.gov.uk/government/publications/towards-fusion-energy-the-uk-fusion-strategy.
[2] UK Atomic Energy Authority, West Burton selected as home of STEP fusion plant, 2022, https://www.gov.uk/government/news/west-burton-selected-as-home-of-step-fusion-plant.
[3] M. Ono, R. Kaita, Recent progress on spherical torus research, Phys. Plasmas 22 (4) (2015) 040501.
[4] F. Najmabadi, The ARIES Team, Spherical torus concept as power plants—the ARIES-ST study, Fusion Eng. Des. 65 (2) (2003) 143–164.
[5] H. Wilson, J.-W. Ahn, R. Akers, D. Applegate, R. Cairns, J. Christiansen, J. Connor, G. Counsell, A. Dnestrovskij, W. Dorland, M. Hole, N. Joiner, A. Kirk, P. Knight, C. Lashmore-Davies, K. McClements, D. McGregor, M. O'Brien, C. Roach, S. Tsaun, G. Voss, Integrated plasma physics modelling for the Culham steady state spherical tokamak fusion power plant, Nucl. Fusion 44 (8) (2004) 917.
[6] J. Menard, T. Brown, L. El-Guebaly, M. Boyer, J. Canik, B. Colling, R. Raman, Z. Wang, Y. Zhai, P. Buxton, B. Covele, C. D'Angelo, A. Davis, S. Gerhardt, M. Gryaznevich, M. Harb, T. Hender, S. Kaye, D. Kingham, M. Kotschenreuther, S. Mahajan, R. Maingi, E. Marriott, E. Meier, L. Mynsberge, C. Neumeyer, M. Ono, J.-K. Park, S. Sabbagh, V. Soukhanovskii, P. Valanju, R. Woolley, Fusion nuclear science facilities and pilot plants based on the spherical tokamak, Nucl. Fusion 56 (10) (2016) 106023.
[7] V. Chuyanov, M. Gryaznevich, Modular fusion power plant, Fusion Eng. Des. 122 (2017) 238–252.
[8] T. Brown, J. Menard, Architectural development of an ST fusion device, Fusion Eng. Des. 192 (2023) 113583.
[9] M. Kovari, M. Coleman, I. Cristescu, R. Smith, Tritium resources available for fusion reactors, Nucl. Fusion 58 (2) (2017) 026010.
[10] R.R. Wessen, C. Borden, J. Ziemer, J. Kwok, Space mission concept development using concept maturity levels, 2013, URL https://hdl.handle.net/2014/44299.
[11] M. Coleman, S. McIntosh, BLUEPRINT: A novel approach to fusion reactor design, Fusion Eng. Des. 139 (2019) 26–38.
[12] M. Kovari, R. Kemp, H. Lux, P. Knight, J. Morris, D. Ward, "PROCESS": A systems code for fusion power plants – Part 1: Physics, Fusion Eng. Des. 89 (12) (2014) 3054–3069.
[13] M. Kovari, F. Fox, C. Harrington, R. Kembleton, P. Knight, H. Lux, J. Morris, "PROCESS": A systems code for fusion power plants – Part 2: Engineering, Fusion Eng. Des. 104 (2016) 9–20.
[14] S.I. Muldrew, H. Lux, G. Cunningham, T.C. Hender, S. Kahn, P.J. Knight, B. Patel, G.M. Voss, H.R. Wilson, "Process": Systems studies of spherical tokamaks, Fusion Eng. Des. 154 (2020) 111530.
[15] J. Morris, S. Muldrew, P. Knight, M. Kovari, A. Pearce, S. Kahn, J. Maddock, T. Nunn, R. Chapman, C. Swanson, H. Lux, J. Foster, K. Ellis, M. Kumar, G. Turkington, J. Lion, S. Chislett-McDonald, A. Brown, J. Edwards, C. Ashe, E. Fable, K. Zarebski, C. Griesel, R. Kemp, S. Gubbins, S. Pickering, J. Matthews, G. Graham, C. Mould, T. Miller, S. Gadgil, D. Short, J. Cook, P. Lloyd, PROCESS, 2023, URL https://github.com/ukaea/PROCESS.
[16] G. Cenacchi, A. Taroni, JETTO: A free boundary plasma transport code, Tech. Rep ENEA-RT-TIB–88-5, ENEA, 1988, URL https://inis.iaea.org/search/search.aspx?orig_q=RN:19097143.
[17] A. Hudoba, G. Cunningham, S. Bakes, Magnetic equilibrium optimisation and divertor integration in spherical tokamak reactors, Fusion Eng. Des. 191 (2023) 113704.
[18] M. Coleman, J.E. Cook, F. Franza, I.A. Maione, S. McIntosh, J. Morris, D. Short, A.I. Blair, M. Bluteau, H. Brooks, I. Chiang, S. Desai, M. Foord, O. Funk, G.A. Graham, L. Humphrey, M. Johnson, S. Kahn, C. MacMackin, S. Mason, H. Saunders, D. Vaccaro, BLUEMIRA, 2023, URL https://github.com/Fusion-Power-Plant-Framework/bluemira.
[19] M. Siccinio, G. Federici, R. Kembleton, H. Lux, F. Maviglia, J. Morris, Figure of merit for divertor protection in the preliminary design of the EU-DEMO reactor, Nucl. Fusion 59 (10) (2019) 106026.
[20] A. Petrov, K. Zarebski, R. Griffiths, S. Wray, T. Stroud, M. Abdallah, C. Clements, T. Owoeye, M. Cannon, N. Brewer, J. Acres, V. Laksharam, S. Stewart, K. Rochford, S. Gribben, C. Finlay, T. Jackson, Power balance models, 2022, URL https://github.com/ukaea/powerbalance.
[21] A. Petrov, T. Stroud, D. Blackburn, T. Owoeye, S. Wray, K. Zarebski, J. Acres, M. Abdallah, C. Clements, M. Cannon, F. Christie, T. Jackson, V. Laksharam, S. Stewart, R. Griffiths, N. Brewer, K. Rochford, S. Gribben, An open-source power balance model for the estimation of tokamak net electrical power output, Fusion Eng. Des. 191 (2023) 113563.
[22] W. Han, N. Offeddu, T. Golfinopoulos, C. Theiler, C. Tsui, J. Boedo, E. Marmar, the TCV Team, Suppression of first-wall interaction in negative triangularity plasmas on TCV, Nucl. Fusion 61 (3) (2021) 034003.