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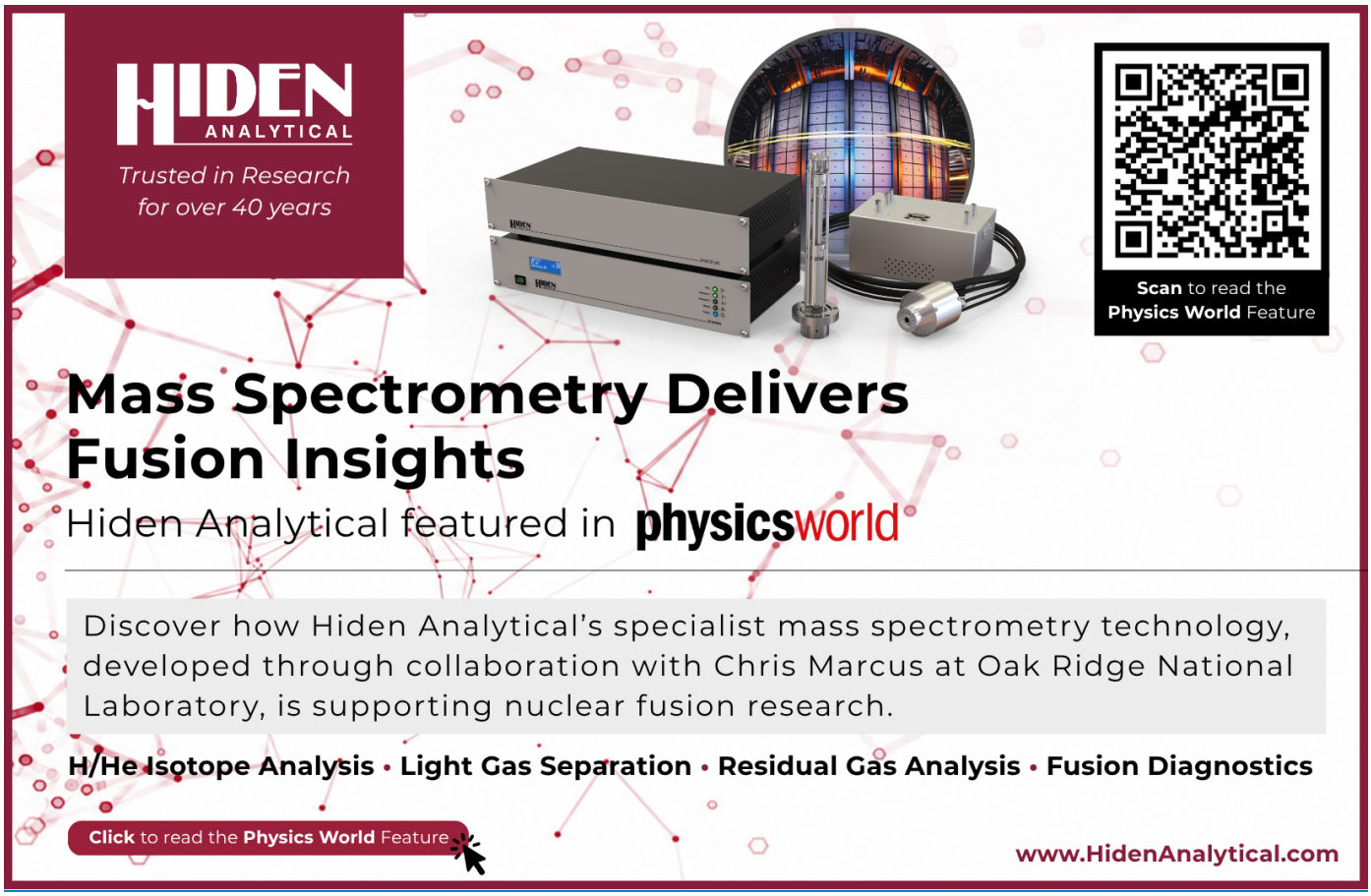

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Tokamak Energy's pre-concept design for a fusion power plant: an overview of ST-E1

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Abstract

Climate change and rapidly rising energy demand, driven in part by artificial intelligence and data-centre growth, create an urgent need for stable, low-carbon, and abundant power. Fusion is a promising long-term solution, yet its commercialisation faces a fundamental paradox in today's investment environment: pilot plants are essential to de-risk physics, engineering, and operations, but their limited lifetime energy output and high upfront costs make them difficult to finance. This paper presents Tokamak Energy's response: ST-E1, a pre-concept design for a low-aspect-ratio tokamak power plant engineered specifically to overcome this challenge.

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ST-E1 is designed from the outset for phased operation—pilot and commercial phases, with an upgrade phase in between—with emphasis on commercial viability, maintainability, nuclear engineering, modularity, and upgradability. A key design principle is the deliberate separation of long-lived assets, such as the magnet cage and vacuum vessel, from replaceable in-vessel systems. This provides an attractive and credible investment approach to generate operational data and de-risk key technologies while preserving most capital-intensive assets for later commercial phases. The architecture supports continuous optimisation toward high net electric power (targeting 800–1000 MW net electric), a normalised capital expenditure of $ 12–14k/kW of net electric power, and high availability (targeting $> 80\%$). A tokamak core with a 5 m major radius, aspect ratio of 2.3, and on-plasma axis toroidal field of 5.25 T was selected to meet these objectives. This paper summarises the ST-E1 design philosophy, principal features, and development methodology. It introduces a Focus Collection of 11 papers detailing the pre-concept design of the entire tokamak and corresponding plant.

Keywords: ST-E1, Tokamak Energy, fusion power plant

(Some figures may appear in colour only in the online journal)

1. Introduction

Tokamak Energy (TE) has completed the pre-concept design of a low-aspect ratio fusion power plant, ST-E1. This paper introduces the 11-paper Focus Collection entitled *Pre-concept design of Tokamak Energy's fusion power plant, ST-E1* [1–10]. The collection presents a complete, self-consistent, and integrated overview of all major plant systems.

Human-driven climate change and global warming have intensified the search for alternative low-carbon energy sources [11–13]. At the same time, global electricity demand is increasing rapidly due to the accelerated adoption of AI technologies and the electrification of transport [14, 15]. These trends are also influenced by the growing emphasis on energy security and national energy independence [16]. Although renewable technologies such as wind and solar continue to expand rapidly, their inherent variability creates a critical need for substantial amounts of firm, stable low-carbon baseload generation to ensure system stability and reliability [17–21]. Controlled thermonuclear fusion offers a promising pathway toward such a solution. Among fusion approaches, magnetic confinement—particularly the tokamak configuration—is widely regarded as the most commercially viable option [22]. Decades of research on tokamak devices have culminated in the ongoing development towards first plasma in several advanced superconducting devices across the world: ITER in France [23], DTT in Italy [24], SPARC in the US [25], and BEST in China [26]. Beyond such flagship projects under construction, the global fusion community has built and operated almost 200 tokamak devices, each of which plays a role in developing the scientific knowledge and technological capabilities required for practical fusion energy [27]. In preparation for commercial fusion, several conceptual reactor designs have been produced by governments and private entities [28–39].

Fusion power plants must simultaneously satisfy a broad set of physics, engineering, and commercial constraints to be reliable and economically competitive. The design space is strongly coupled and inherently multidisciplinary, with progress in one area often creating new requirements or limitations in another. The principal constraints that must be reconciled in an integrated plant design include:

- **High-performance plasma scenarios**: leveraging favourable plasma physics regimes while remaining compatible with divertor heat loads, radiation balance, and plasma control requirements.
- **Maintainability**: establishing streamlined maintenance schemes adds engineering and physics design constraints but is paramount to ensure machine operability and high availability.
- **Commercial relevance**: achieving commercial relevance through long component lifetimes, low normalised capital expenditure (CapEx), and high availability factor. A higher plant availability inherently enables higher eventual capacity factor if market conditions allow.
- **Net electrical output**: delivering steady and meaningful electrical power to the grid.
- **Fuel self-sufficiency**: ensuring that the breeding solution produces tritium at a rate that is greater than that at which tritium is burned, decayed, and retained in components.
- **Safety by design**: designing the tokamak and plant for technical safety and compliance with regulations.
- **Deliverable**: mandating a timeline for delivery to force design decisions to reflect realistic technologies and plans to grow supply chains.

A plant design that fails on any of these criteria will not produce a viable concept.

TE has designed, constructed, and operated a high field Spherical Tokamak (ST), ST40, which has demonstrated the highest plasma triple product achieved to date by any private fusion company [40, 41]. In parallel, the company has advanced high-temperature superconducting (HTS) magnet technology through projects such as Demo4, which delivered the first high-field HTS magnet system to be built, commissioned, and tested in a tokamak configuration [42]. These developments have culminated in the pre-conceptual design of a low-aspect-ratio fusion power plant, ST-E1, which draws

extensively from the experience obtained from the development and operation of the preceding devices.

Through an initial series of design and trade studies, a reference design point (RDP) has been established for the pre-conceptual design phase. The ST-E1 power plant concept is characterised by a steady state plasma tokamak with a major radius of 5 m, an aspect ratio of 2.3, and an on-plasma axis toroidal field (TF) of 5.25 T. ST-E1 is conceived as a device capable of delivering commercially relevant net electrical output, underpinned by a design philosophy that prioritises maintainability, robust nuclear engineering, modularity, and upgradability. A key enabler of this is the radial maintenance scheme, which enables the removal of entire toroidal segments along a single axis.

The pre-concept design of ST-E1 forms part of TE's deliverable of the US Department of Energy's Milestone-Based Fusion Development Program [43]. This paper provides an overview of ST-E1 where: section 2 describes the mission and operating targets of ST-E1, section 3 presents the tokamak overview and key design features, including the reference design point and the distinction between permanent and upgradable systems, and section 4 describes the design methodology and the approach to systems engineering (SE).

2. Mission

The mission of TE's ST-E1 programme has been defined in response to evolving public and private funding environments, informed by extensive engagement with stakeholders across government, industry, and the fusion supply chain. Recent trends in fusion funding increasingly emphasise demonstrable pathways to commercial viability, motivating a shift from experimental objectives toward performance metrics relevant to power-plant deployment. Consequently, the ST-E1 programme is structured to demonstrate operation consistent with commercially viable fusion power plants, with target performance metrics including:

- Normalised CapEx[8] of $ 12–14k/kW of net electric power
- Plant availability of >80%

To achieve these objectives, ST-E1 adopts a phased operational strategy designed to progressively de-risk technology, physics, and plant operation while converging toward commercial performance targets.

In the pilot phase, ST-E1 will operate for a period equivalent to one full-power year (FPY), with the primary objective of de-risking critical technologies, validating plasma and systems physics, and establishing operational experience. Although this phase is not expected to achieve commercially competitive normalised CapEx or availability, it will still provide significant power to the grid (300–500 MW), as well as essential data to inform component design, maintenance strategies and demonstrations of increasing plant availability, regulatory engagement, and supply-chain development. Hence, it establishes the technical and institutional foundations for subsequent commercial phases while still producing significant net electricity. The design scope of the pilot phase is defined by time limitations associated with assumptions of technology readiness, constrained by a target of 10 years from project inception to first plasma operations. This constraint is a key design driver that ensures that system choices are guided by pragmatic assessments of technology maturity and deployability rather than aspirational performance alone.

The upgrade phase incorporates operational insights and data obtained during the pilot phase to improve component reliability, reduce internal recirculating power, and increase the fusion power output. This phase focuses on integrating design refinements and system-level improvements required to transition from pilot-scale operation to sustained high-performance operation.

The commercial phase represents full-performance operation of the ST-E1 plant, with a full-power operational lifetime of approximately 30 years. In this phase, the programme aims to achieve a normalised CapEx of $ 12–14k/kW of net electric power and an availability exceeding 80%, consistent with benchmarks required for competitive electricity generation. Net electric power will be on the order of 800–1000 MW.

This mission retains key aspects of the 2021 report of the *National Academies of Sciences, Engineering, and Medicine, Bringing Fusion to the U. S. Grid* [44] which emphasises the necessity of pilot-scale facilities to de-risk fusion technologies. However, the ST-E1 programme explicitly integrates pilot operations with a defined path to commercial performance, recognising that sustained investment in fusion energy requires credible demonstration of long-term operational and economic viability.

3. Design overview

This section summarises the major features of plant design, providing the context in which the accompanying papers in this focused collection develop the physics, engineering, and technology basis in more detail. The term *pre-concept design* describes a design phase in which the problem space, mission needs, and high-level technical requirements are established and a consistent RDP is identified. Scoping studies performed with the TE whole plant systems modelling code, PyTok, can be found in [1] by Wilson *et al*. These underpin the choice of the main device parameters and operating regime and establishes the attenuation of neutron flux at the centre column, and corresponding shielding thickness, as the key driver of machine major radius, particularly in low aspect ratio tokamaks. The main design features of the device, figure 1, are dictated by the requirement that the second phase of the machine be commercially competitive, and this paper shows that the design point achieves the desired normalised CapEx. A PyTok-generated Sankey diagram showing all major power flows of the commercial phase of the plant can be seen in figure 2. An initial cost analysis was conducted for the ST-E1 concept; however, detailed results fall outside the scope of this

[8] Normalised CapEx is defined as the Overnight Capital Cost (OCC) to realise construction of the plant up to start of commercial operations, normalised per unit power produced at full nameplate capacity.

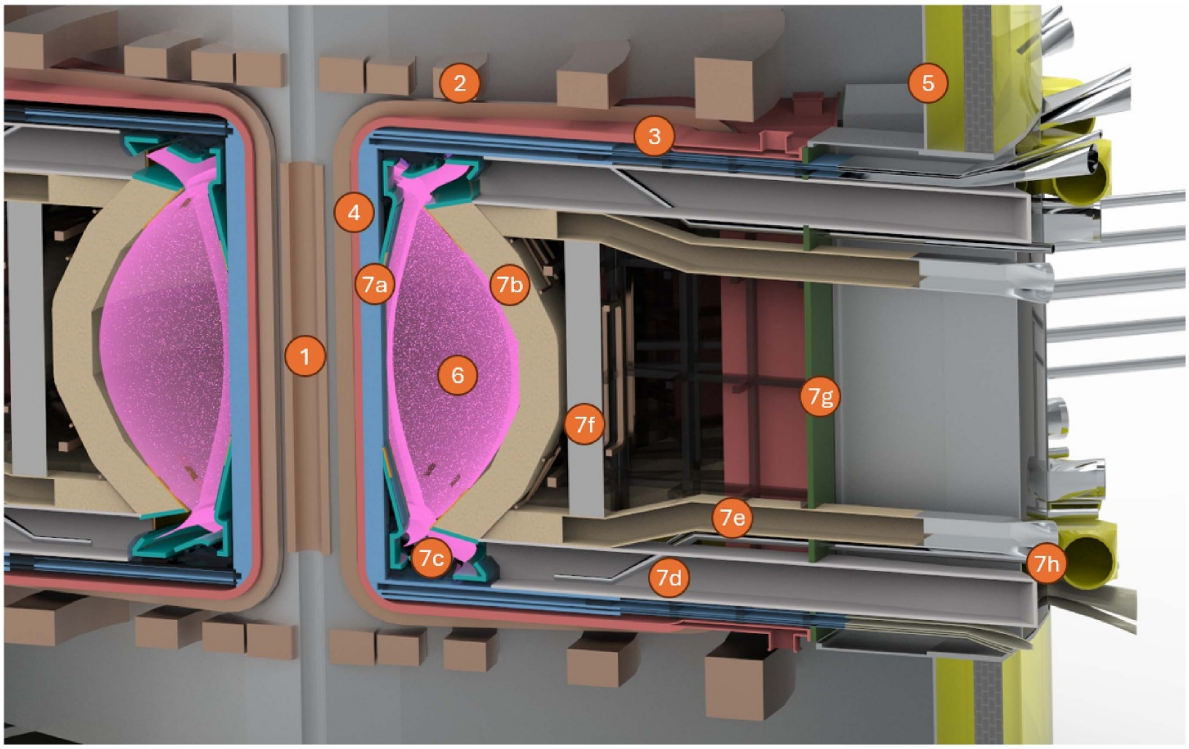

Figure 1. The major features of the Reference Design Point; 1. 30Vs HTS central solenoid(brown), 2. Lifetime HTS PF coils (brown) outside TF coils (brown), 3. Permanent vacuum vessel (red), 4. Permanent core shielding (integrated within vacuum vessel, red), 5. Cryostat (yellow) 6. Plasma 7. ***Removable segment:*** 7(*a*). Replaceable core shielding (blue) 7(*b*). Breeder blanket (beige), 7(*c*). Solid tungsten PFCs (teal), 7(*d*). Vacuum duct (grey), 7(*e*). Lithium duct (beige), 7(*f*). Outboard vertical shielding (grey), 7(*g*). Cryostat vacuum flange (green), 7(*h*). External service connections.

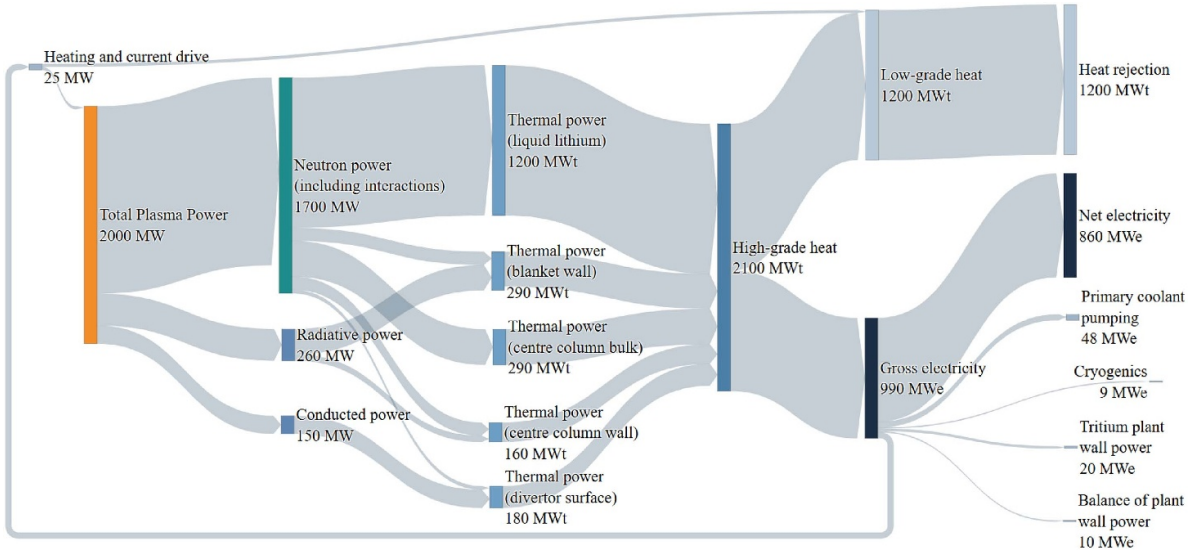

Figure 2. A Sankey diagram outlining the main power flows of a representative case of ST-E1's commercial phase. The power numbers have been rounded to two significant figures.

paper. The analysis considered overnight capital cost, operational expenditure, and the levelised cost of electricity (LCOE) for the proposed plant configuration. The design exploration was driven by assessment of the normalised CapEx as part of PyTok, ensuring that the RDP was aligned with the mission statement.

3.1. Reference design point

At the pre-concept stage, the ST-E1 design is represented by an RDP that is used consistently across the associated paper cohort. Specifically, it distinguishes between full-life components and replaceable components. Table 1 summarises

Table 1. Pre-conceptual reference design point (RDP) for the pilot and commercial configurations. In cases where the parameters are identical between phases, the commercial column is left empty.

Parameter	Pilot Phase	Commercial
R_0 (m)	5.0	—
A (−)	2.3	—
B_0 (T)	5.25	—
I_p (MA)	∼15	—
Solenoid flux-swing (Vs)	30	—
δ / κ	0.5 / 2.4	—
$P_{\text{fus,nom}}$ (MW)	∼1500	∼2000–2500
P_{sep}/R (MW m^{-1})	∼25	∼30–40
$P_{\text{aux,nom}}$ (MW)	∼75	∼25–50
Q (−)	∼20	∼50–70
$P_{\text{elec,net}}$ (MWe)	300–500	∼800–1000
TF lifetime (FPY)	> 30	—
Neutron wall loading (MW m^{-2})	< 1.6	∼2.1–2.6
TBR (−)	> 1.14	—
Maintenance scheme	Radial	—

the pre-conceptual RDP for the pilot phase and the corresponding commercial configuration. The two key differences between the Pilot and Commercial phases are; the Commercial phase has a reduced recirculating power, as enabled by technological advancements in key areas; and a requirement for higher fusion power and higher fusion gain, as enabled through learnings from the Pilot phase. Along with the RDP, subsequent studies identified a range of devices with equivalent plant wide performance with aspect ratios in the range $A = 1.9 – 2.3$ and corresponding toroidal magnetic field strengths of $B_0 = 3.25 – 5.25$ T.

3.1.1. Permanent architecture. The permanent features of ST-E1, figure 3, are specified and designed for the entire life, including the commercial phase, and include:

- A low aspect ratio magnet cage, including lifetime TF, poloidal field (PF), and centre solenoid (CS) coils.
- A permanent vacuum vessel, neutron shielding, and cryostat.
- Civil structures and plant, acknowledging that some plant systems may have to be up-rated to handle a higher power throughput.

This ensures that the major capital-intensive systems can be retained through the planned upgrade, while more uncertain or rapidly evolving technologies are placed in upgradable subsystems. The design of the maintenance scheme to enable this upgrade was a key driver behind the choice of the radial maintenance concept [5].

3.1.2. Upgradable systems. Components directly coupled to the plasma environment or to high-temperature, high-neutron flux regions of the power core are treated as replaceable and upgradable. This is enabled by the replaceable toroidal segment concept and includes:

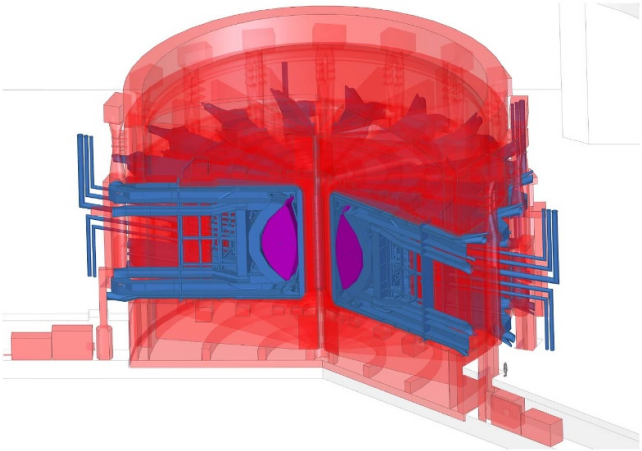

Figure 3. Permanent, lifetime components of the tokamak (highlighted in red), and replaceable components (in blue).

- The breeding blanket system
- In-vessel structural items
- The divertor and other plasma-facing components (PFCs)
- High performance neutron shielding
- Elements of the heating and current-drive (H&CD) systems and diagnostic suite
- In-vessel auxiliary control and field correction coils

In the pre-concept design, these systems are specified primarily to meet the pilot phase requirements, with commercial performance assumed to be achieved through replacement or re-optimisation of the replaceable components within the fixed permanent envelope. This separation allows the pilot configuration to rely on established or intermediate technologies while reserving an upgrade path to higher power density, lower recirculation power, better thermal efficiency, and more advanced plasma scenarios in the commercial phase.

3.2. Tritium breeding, nuclear engineering, and power extraction

The breeder blanket concept is based on a pure lithium breeder, combined with vanadium-alloy structural materials and helium cooling in the initial configuration. Pure, unenriched lithium is selected to achieve a high volumetric tritium breeding ratio (TBR) without reliance on Li-6 enrichment or extensive use of toxic neutron multipliers such as beryllium, and maintains acceptable radioactive waste. The tritium breeding system is further explored by Yildirim *et al* in [3]. The target of the TBR is $\gtrsim 1.1$, leaving a sufficient margin to cover decay and processing losses [3]. Vanadium alloys provide a wide allowable temperature window and favourable neutron-absorption properties compared to reduced-activation steels, enabling higher temperatures and improved thermodynamic efficiency.

To shield the magnet structures and provide a safe environment for personnel, high performance tungsten carbide shielding is used in modular units in addition to boron carbide, reduced activation ferritic/martensitic (RAFM) and stainless steels. Chandrasekhar *et al* [2] further explore the damage and activation of these materials, where replaceable segments are engineered for 5 FPYs, with tungsten plasma facing armour tiles potentially requiring a higher replacement frequency in the high damage mid-plane region, and permanent structures are engineered for 30 FPYs.

Throughout the plant, helium is used as the primary coolant. For the blanket, a conservative assumption is made that helium removes the majority of the blanket's heat. This choice reduces the near-term technology complexities of dual- and self-cooled pure lithium designs but imposes significant compression power penalties on the helium system. The extracted heat is converted to electricity by an advanced supercritical Rankine cycle. The cycle incorporates high-pressure, intermediate-pressure and low-pressure turbines, co-ordinated with steam superheaters and reheaters, and multiple regenerative feedwater heaters optimised for maximum heat recovery. At design conditions, the system achieves a thermal-to-electric conversion efficiency in excess of 47%. This is further discussed by Mohamed *et al* in [4]. The remaining major heat sources are the centre-column shielding and the PFCs, with multiple PFC options explored for their effect on helium pressure drop and overall integration into the thermal balance of plant. The architecture deliberately preserves an upgrade path, particularly for the blanket, where increased utilisation of liquid lithium as a self- or dual-cooled option in commercial configuration could reduce recirculating power by up to 200 MW, and increase the thermal cycle efficiency. A $\leqslant$5 min plasma current ramp-up to full fusion power, while limiting peak grid power draw to 500 MW, requires only modest on-site energy storage of 1–2 MWh, leaving plenty of room for further reductions in grid power demand.

3.3. Plasma scenarios, magnets, heating and current drive, and power exhaust

In order to maximise availability and reduce cyclic fatigue, the plasma scenario in the commercial phase of ST-E1 will be steady-state. This requires fully non-inductive sustainment of the plasma current. Steady-state operating points require high self-driven bootstrap current fractions, with performance strongly dependent on plasma confinement and stability. The most promising concepts are the ST [40, 45–47] and Advanced Tokamak (AT) [37, 48–50], which share many favourable features, such as enhanced confinement and improved stability. The impact of key parameters on the flat-top operating space is discussed in the focus collection paper by McNamara *et al* [7]. This includes the assumed pedestal characteristics, level of external heating power, assumed density limit, core radiation fraction and divertor power loading. These investigations informed the definition of a series of candidate operating points with different characteristics that have been used to further assess the magnetohydrodynamic stability, controllability and micro-stability properties of these plasmas. The radiation corrected energy confinement enhancement factor, relative to the ITER98(y,2) scaling [51] , are in the range $H^*_{98} = 1.3 - 1.6$ and the Greenwald density fraction, defined on the top of the pedestal, is $f^{\mathrm{ped}}_{\mathrm{GW}} = 0.8 - 1.0$. The safety factor is kept above $q_{\mathrm{min}} \geqslant 2.2$ to avoid sawteeth, low order neo-classical tearing modes, and coupling to infernal modes. The core radiated power fractions in the range $f_{\mathrm{rad,core}} = 45\% - 69\%$, and all scenarios typically operate marginally above the no-wall normalised beta limit. However, operation below the no-wall normalised beta limit is possible, removing the need for resistive wall mode control, with increased external heating & current drive power. The current RDP has two up/down symmetric divertors. Operating in a well-balanced double null plasma requires the vertical position and x-points of the $\sim$10 m tall plasma to be controlled with an accuracy on the order of one scrape-off layer width, or 1–10 mm ($\sim$0.01%–0.1%). Acknowledging the plasma diagnostics and control challenges associated with this, the ST-E1 is being designed to initially operate with a disconnected double null plasma scenario.

Being able to ramp to full current and access fusion burn early helps to aid the overall energy balance of the plant and reduces the instantaneous grid power demand. The complete ramp-up of plasma current and fusion power can be achieved in less than five minutes with the assistance of a substantial, 30 Vs HTS solenoid and external vertical-field coils. Strategically using external field coils in combination with a solenoid has been a general philosophy throughout the ST-E1 design evolution [52]. Time-dependent physics scenario development is discussed by Zhang *et al* in [8]. Using the solenoid to ramp-up a large fraction of the plasma current avoids the dependence on the slow relaxation of electron-cyclotron (EC)-driven current during ramp-up, which substantially increases

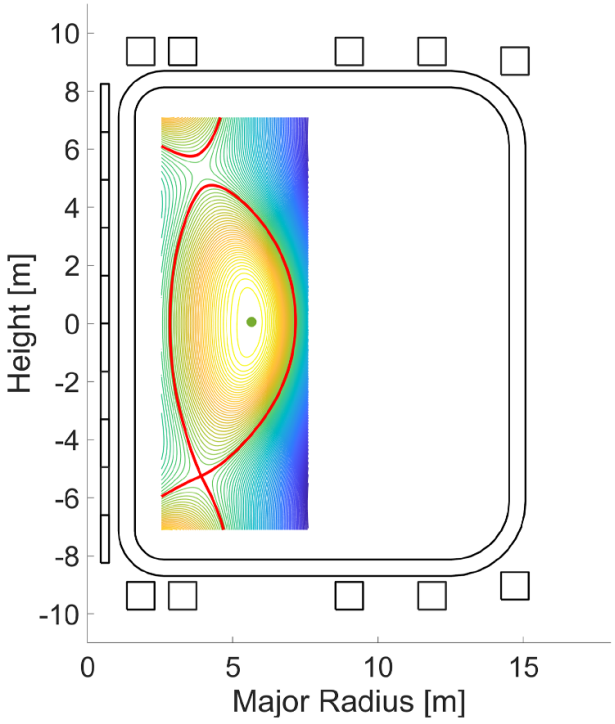

Figure 4. Representative disconnected double-null magnetic equilibrium of ST-E1.

the ramp-up time. The ST-E1 ramp-up scenario is characterised by a relatively high plasma density ($\gtrsim 80\%$ Greenwald fraction) throughout to facilitate early transition to H-mode confinement and early fusion burn. A stable, disruption-free current ramp-down is also critical to the operation of the plant. The time-scale for the ramp-down is set by the current diffusion time at the end of the flat-top phase and a representative ramp-down lasting 1000 s has been developed. The main driver for the ramp-down scenario is maintaining a favourable safety factor profile and moderate internal inductance for MHD stability and controllability, while remaining in H-mode far into the current ramp-down to avoid issues associated with undergoing an HL back transition at high plasma current.

The auxiliary H&CD system is mainly based on electron-cyclotron resonance heating and current drive (ECRH/ECCD) during the ramp-up, flat-top and ramp-down phases. The option of augmenting this with ion-cyclotron resonance heating (ICRH) during ramp-up to accelerate access to fusion-burn conditions remains. A novel method has been developed to rapidly scope out the EC system [53], providing rapid and computationally efficient estimates for key design parameters. Using this approach, it has been determined that an EC system with commercially available frequencies ($\sim$170 GHz) can provide the required current drive during all phases of the plasma duration. This is further discussed by Lopez *et al* in [9]. The steady-state flattop ECCD efficiency is found to be around 50 kA MW^{-1}.

All HTS magnets in ST-E1 will be monolithic, meaning they will not be re- or de-mountable. The inherent resistive losses in de-mountable HTS joints, the challenges of balancing structural mass warm-up/cool-down time, cryogenic power draw and maintenance interval duration, and the level of required technological development meant the design moved away from reliance on jointed coils. TF, PF, and CS magnets must be designed with respect to the maintenance scheme (see more about the maintenance scheme in section 3.4). This places them away from the midplane; see figure 4. A range of in-vessel auxiliary coils are expected to be necessary for ST-E1: fast-acting, copper vertical and radial control coils (VCCs, RCCs), HTS error-field correction coils, and a set of 3D copper coils outside the breeding blanket will allow for disruption and edge-localised mode control. These auxiliary coils have been put in as placeholders in the design but not further specified. An integrated workflow effort was undertaken to ensure that the coils can ramp and hold a range of ramp-up and flat-top scenarios. By designing a single set of PF and CS coils for a large range of ramp-up, flat-top, and ramp-down scenarios, it is possible to avoid over-optimisation of coils for a given scenario early on in the design process. A peak field on coil of 20 T is used as a guideline during the pre-concept design of the HTS coils. The current density is not typically a limiting factor. The coils are more often limited by the fraction of current density to critical current (typically referred to as *gamma*), in order to consider radiation degradation effects. The resulting reference magnet cage was then further designed and analysed in a more detailed coil engineering workflow, as discussed by Maartensson *et al* in [6]. Plasma stability and control studies of the highly elongated plasma have also been performed, with preliminary results indicating acceptable controllability [7].

Finding viable power exhaust solutions remains a challenge for all tokamak designs. Robinson *et al* [10] have carried out an investigation on the viability of ST-E1 power exhaust. Detaching the inner and outer divertor legs by radiating all incoming power is a requirement for PFCs to survive the heat and particle flux. This has been achieved in a long-leg configuration using SOLPS-ITER modelling with argon as the impurity gas, without the use of advanced divertor-shaping, likely requiring PF coils within TFs. For plasma facing armour applications, potassium doped tungsten is the selected material.The erosion of first wall PFCs and the transport of eroded material inside the device remain a concern, in addition to finding a solution for high-heat flux cooling compatible with net power production requirements. This leads to active pursuit of the development of liquid lithium PFCs alongside solid PFCs solutions. Liquid lithium PFCs, if solid PFC solutions are found to be inadequate, are envisaged to culminate in the commercial phase as a sufficiently de-risked, fully-toroidal liquid lithium divertor to aid in improving divertor lifetime, heat flux handling, impurity removal, and erosion effects.

3.4. Maintenance strategy and plant architecture

Achieving a commercially relevant availability factor relies heavily on identifying the cost and schedule drivers of maintenance operations. Given that plant availability is one of the dominant determinants of fusion power plant economics, the early specification of the tokamak configuration and overall plant architecture that enables highly efficient and periodic scheduled maintenance is of critical importance. The

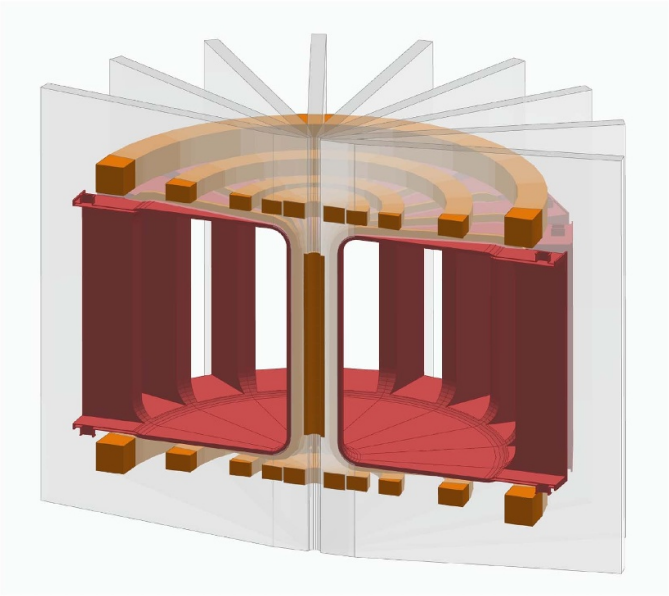

Figure 5. Tokamak magnet cage (orange) and vacuum vessel (red). Magnet cage support structure (grey) shown for context.

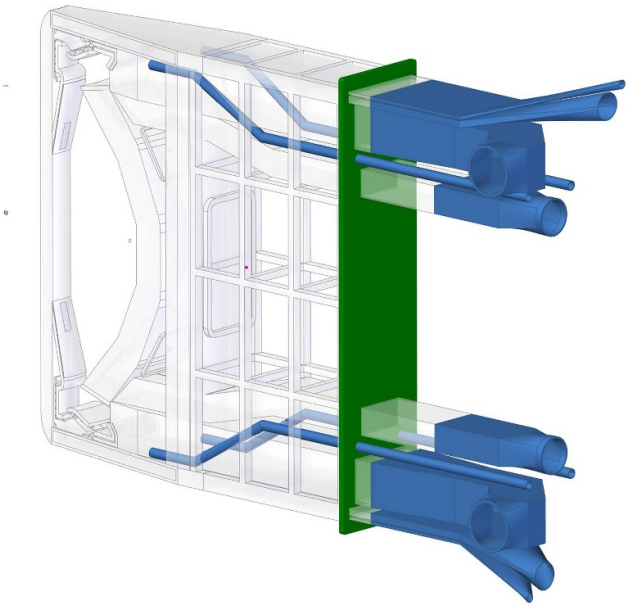

Figure 6. Tokamak core segment, showing vacuum sealing flange (green) and external service routing (blue).

chosen maintenance architecture uses a replaceable, toroidally segmented core structure combined with a radial extraction method. This reduces operational risk by minimising complexity and avoiding operations in high-radiation and space constrained areas. The scheme is also a project risk mitigation strategy by enabling ease of access to the entire inside of the machine without complex in-vessel operations. Furthermore, it allows for a complete upgrade of all in-vessel components and geometries to implement learnings from the pilot phase to achieve commercial relevance. A further discussion of these points can be found in Willis *et al* [5]. The core of the tokamak, thus, consists of the permanent vacuum vessel and magnet cage, with the former incorporating large, full-height, vacuum ports that extend radially between TF coil support limbs, seen in figure 5.

The removable core segments align with the vacuum ports and are fully self contained, with all services routed through the rear of each segment and exiting via the vacuum sealing flange, per figure 6.

By keeping the movement of a replaceable segment to a single axis, the kinematic complexity of removing a segment is greatly reduced. It also minimises the need for in-vessel activities and allows all service joining operations to be conducted in the relatively benign environment of the hot cell area. Little to no scheduled welding activities will be conducted in-vessel, saving time and reducing the inspection burden and risk of weld failures.

A split-shield concept separates a replaceable high-performance inner shield, as part of the core segment, which protects the vacuum vessel, from a permanent outer shield that provides the remaining neutron attenuation for the TF magnets. Early, iterative neutronics simulations were conducted in order to validate the approach and are discussed further in [2]. This reduces the volume of highly activated material that must be disposed of.

4. Design methodology

4.1. Workflow and design evolution

The pre-concept design of ST-E1 evolved from a mission statement to the current RDP through two consecutive workflows. Early two-week iterations rapidly explored the solution space using PyTok [54]. Each iteration focused on a specific design point to identify potential deal-breakers and inform the next cycle [1]. After convergence at this lower fidelity, the process transitioned to two-month iterations that examined the design

in greater depth. Fixed time frames constrained scope creep and schedule slippage. Each iteration concluded with a full design review, with major changes implemented between iterations. A final six-month iteration matured the design to pre-concept level.

New modelling tools and capabilities were continuously developed as part of the workflows. New understanding, models, heuristics, and limits could be applied to each iteration, since the design point was fixed but also fed back to PyTok for a more accurate system code representation. The flexibility of a fixed design point per iteration allowed the continuous improvement of design processes, tools, and the resulting machine. Examples of outputs from this include: SCOPE, a PF and CS coil optimisation tool [6, 55]; novel understanding and optimisation tools for EC H&CD [52, 53, 56]; further development of low-aspect ratio physics such as magneto-hydrodynamic stability, divertor optimisation, and gyrokinetics transport; concept designs of novel and high-performance centre-column shielding and liquid lithium blankets.

A crucial challenge when doing iterative design work is to address the inherent uncertainty of the physics scenario, whilst acknowledging the simultaneous engineering need of having a single design point to work on. This was naturally solved through design iterations and freezing of changes until between iterations. Physics investigations could proceed in parallel to design work, and any resulting issues or differences could be consolidated at each design update.

4.2. Systems engineering framework

A highly integrated power plant design requires robust systems engineering (SE) approaches from an early stage. This helps align stakeholder expectations, provides key functionality, and enables appropriate trade-off between system requirements. During the pre-concept design of ST-E1, an SE approach has been applied, bringing together technical management processes into a Model Based Systems Engineering (MBSE) framework.

A Concept of Operations has been established to define how the plant will meet the mission statement and key programme drivers, constraints, and stakeholder expectations. This forms the basis for deriving high-level technical requirements.

The MBSE approach consisted of the development of linked functional, logical, and physical architectures. This provides clear allocation of functional requirements to logical systems, while keeping the design space open and avoiding over-constraining the physical solution. It is particularly valuable in the pre-concept phase, where developing a stable definition of the problem space and technical requirements is key. A subsequent physical architecture can then be defined to describe the components that realise the logical elements, to which there may be many possible solutions. The MBSE framework directly supports design iteration and decision-making, making it a valuable platform from which to progress into a conceptual design phase and beyond. This is especially important as the number of potential engineering, construction, and science partners increases, each with the need to interface consistently with the project.

5. Conclusions

The commercialisation of fusion is constrained by a fundamental paradox: pilot plants are required to de-risk plasma physics, engineering integration, and operational models, yet these same facilities are difficult to finance due to their limited lifetime energy output and high upfront capital costs. Resolving this tension is essential if fusion is to transition from scientific promise to commercial reality.

The ST-E1 pre-conceptual design is explicitly structured to address these technological and economic constraints. It adopts time-bounded technology choices, directly incorporates key commercial and market drivers, and advances all major plant systems in parallel rather than sequentially. This approach yields a plant and machine architecture that is tolerant to change with a staged deployment model that preserves the majority of long-lived capital assets—including the magnet cage, vacuum vessel, and balance of plant—while allowing replaceable in-vessel systems to evolve. As a result, the design supports substantial performance upgrades and commercially viable operation over its 30 year full-power lifetime.

The proposed configuration is anchored in commercial performance targets intended to enable parity with established generation technologies such as nuclear fission and natural gas. Specifically, the design aims to achieve a normalised capital cost of $ 12–14k/kW of net electric output. A radial maintenance strategy is employed to support an availability exceeding 80%, while the low-aspect-ratio configuration enables steady-state plasma operation, improving overall energy balance and reducing cyclic fatigue. Finally, significant effort has been invested in the development of integrated workflows and design tools to support iterative development and continuous improvement throughout the design process.

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References

[1] Wilson C.L. *et al* 2026 Design scoping and systems modelling of ST-E1 using the PyTok whole-plant code *Nucl. Fusion* **66** 086002
[2] Chandrasekhar K. *et al* 2026 Nuclear technology considerations and neutronics for the ST-E1 fusion power plant *Nucl. Fusion* **66** 086003
[3] Yildirim E. *et al* 2026 Tritium production and processing systems for ST-E1 *Nucl. Fusion* **66** 086004
[4] Mohamed M.K.E. *et al* 2026 Thermal management and net-power evaluation of the ST-E1 fusion power plant *Nucl. Fusion* **66** 086005
[5] Willis J. *et al* 2026 Maintenance strategy, structural design and site layout of the ST-E1 fusion power plant *Nucl. Fusion* **66** 086006
[6] Maartensson E.N.J. *et al* 2026 Integrated physics and magnet design for the ST-E1 fusion power plant *Nucl. Fusion* **66** 086007
[7] McNamara S.A.M. *et al* 2026 Physics basis for the reference flat-top plasma scenario in the ST-E1 fusion power plant *Nucl. Fusion* **66** 086008
[8] Zhang X. *et al* 2026 Time-dependent scenario modelling for the ST-E1 fusion power plant *Nucl. Fusion* **66** 086009
[9] Lopez N.A. *et al* 2026 Auxiliary heating and current-drive physics for the ST-E1 fusion power plant *Nucl. Fusion* **66** 086010
[10] Robinson M. *et al* 2026 Power and particle exhaust in the ST-E1 fusion power plant *Nucl. Fusion* **66** 086011
[11] Intergovernmental Panel on Climate Change 2022 *IPCC 6th Assessment Report (AR6), Working Group III: Mitigation of Climate Change—Chapter 6: Energy Systems* (available at: www.ipcc.ch/report/ar6/wg3/chapter/chapter-6/) (Accessed 28 November 2025)
[12] IPCC 2021 Climate change 2021: the physical science basis *Contribution of Working Group I to the 6th Assessment Report of the IPCC* (Cambridge University Press)
[13] IPCC 2023 Climate change 2023: synthesis report *6th Assessment Report of the IPCC*
[14] International Energy Agency 2024 Energy demand from AI *Energy and AI* (available at: www.iea.org/reports/energy-and-ai/energy-demand-from-ai)
[15] International Energy Agency 2024 Global EV outlook 2024 (available at: www.iea.org/reports/global-ev-outlook-2024)
[16] International Energy Agency 2021 Net zero by 2050: a roadmap for the global energy sector (available at: www.iea.org/reports/net-zero-by-2050)
[17] Climate Change Committee 2023 A reliable, secure and decarbonised power system by 2035 is possible—but not at this pace of delivery (available at: www.theccc.org.uk/2023/03/09/a-reliable-secure-and-decarbonised-power-system-by-2035-is-possible-but-not-at-this-pace-of-delivery/) (Accessed 28 November 2025)
[18] UK Climate Change Committee 2023 Delivering a reliable decarbonised power system (available at: www.theccc.org.uk/publication/delivering-a-reliable-decarbonised-power-system)
[19] Sepulveda N.A., Jenkins J.D., de Sisternes F.J. and Lester R.K. 2018 The role of firm low-carbon electricity resources in deep decarbonization of power generation *Joule* **2** 2403–20
[20] Williams J.H. *et al* 2021 What is different about different net-zero carbon electricity systems? *iScience* **24** 102211
[21] International Energy Agency 2021 The role of low-carbon fuels in the clean energy transitions of the power sector (available at: www.iea.org/reports/the-role-of-low-carbon-fuels-in-the-clean-energy-transitions-of-the-power-sector)
[22] Chang H.J. and Wang S.W. 2025 Advancements in tokamak technology for fusion energy: a bibliometric and patent trend analysis (2014-2024) *Energies* **18** 4450
[23] Loarte A. *et al* 2025 The new ITER baseline, research plan and open R&D issues *Plasma Phys. Control. Fusion* **67** 065023
[24] Romanelli F. *et al* 2024 Divertor Tokamak Test facility project: status of design and implementation *Nucl. Fusion* **64** 112015
[25] Creely A.J. *et al* 2020 Overview of the SPARC tokamak *J. Plasma Phys.* **86** 865860502
[26] Song Y. *et al* 2022 Engineering design of the CFETR machine *Fusion Eng. Des.* **183** 113247
[27] International Atomic Energy Authority (IAEA) 2025 Fusion device information system (FusDIS) (available at: https://nucleus.iaea.org/sites/fusionportal/Pages/FusDIS.aspx) (Accessed 28 November 2025)
[28] Tillack M.S. *et al* 2006 The ARIES-AT advanced tokamak, advanced technology fusion power plant *Fusion Eng. Des.* **80** 19–40
[29] Emmert G.A., Kulcinski G.L., Santarius J.F. 1990 Fusion reactor design studies: ARIES tokamak U.S. Department of Energy. (available at: www.osti.gov/biblio/6477567)
[30] The MANTA Collaboration et al 2024 MANTA: a negative-triangularity NASEM-compliant fusion pilot plant *Plasma Phys. Control. Fusion* **66** 105006

[31] Merola M. *et al* 2024 Re-design of EU DEMO with a low aspect ratio *Fusion Eng. Des.* **204** 114518
[32] Federici G., Kembleton R. and Siccinio M. 2021 Overview of the DEMO design-staged approach in Europe *Proc. 27th IAEA Fusion Energy Conf.* (*Gandhinagar, India*, *22–27 October 2018*) pp 1–32
[33] Kessel C.E. *et al* 2017 Overview of the fusion nuclear science facility, a credible break-in step on the path to fusion energy *Fusion Eng. Des.* **5** 135
[34] Brown T.G. *et al* 2026 The spherical tokamak advanced reactor (STAR) fusion power plant design *Fusion Eng. Des.* **222** 115461
[35] Sorbom B.N. *et al* 2015 ARC: a compact, high-field, fusion nuclear science facility and demonstration power plant with demountable magnets *Fusion Eng. Des.* **100** 378–405
[36] Tobita K. *et al* 2019 Japan's efforts to develop the concept of JA DEMO during the past decade *Fusion Sci. Technol.* **75** 372–83
[37] Buttery R.J. *et al* 2021 The advanced tokamak path to a compact net electric fusion pilot plant *Nucl. Fusion* **61** 046028
[38] Menard J. *et al* 2022 Fusion pilot plant performance and the role of a sustained high power density tokamak *Nucl. Fusion* **62** 036026
[39] Chapman I.T., Cowley S.C. and Wilson H.R. 2024 The Spherical Tokamak for Energy Production: theme issue introduction *Phil. Trans. R. Soc. A* **382** 20230416
[40] McNamara S.A.M. *et al* 2024 Overview of recent results from the ST40 compact high-field spherical tokamak *Nucl. Fusion* **64** 112020
[41] Kingham D. and Gryaznevich M. 2024 The spherical tokamak path to fusion power: opportunities and challenges for development via public-private partnerships *Phys. Plasmas* **31** 042507
[42] Dunbar G. 2025 Demo4—presentation of the assembly, commissioning and testing of a representative set of high field HTS magnets in a reactor relevant configuration *Talk Presented at the 17th European Conf. on Applied Superconductivity (EUCAS 2025)* (*Porto, Portugal*, *21–25 September 2025*) Session 1-LO-FM.4, 17:30–17:45
[43] US Department of Energy 2024 DOE announces new decadal fusion energy strategy (available at: www.energy.gov/articles/doe-announces-new-decadal-fusion-energy-strategy) (Accessed 28 November 2025)
[44] National Academies of Sciences E, Medicine 2021 Bringing fusion to the U.S. Grid The National Academies Press (available at: https://nap.nationalacademies.org/catalog/25991/bringing-fusion-to-the-us-grid)
[45] Wilson H. *et al* 2020 STEP-on the pathway to fusion commercialization *Commercialising Fusion Energy: How Small Businesses Are Transforming Big Science* (IOP Publishing) pp 8-1–8-18
[46] Najmabadi F. *et al* 2003 Spherical torus concept as power plants-the ARIES-ST study *Fusion Eng. Des.* **65** 143–64
[47] Berkery J.W. *et al* 2024 NSTX-U research advancing the physics of spherical tokamaks *Nucl. Fusion* **64** 112004
[48] Najmabadi F. *et al* 2006 The ARIES-AT advanced tokamak, advanced technology fusion power plant *Fusion Eng. Des.* **80** 3–23
[49] Oyama N. *et al* 2009 Overview of JT-60U results towards the establishment of advanced tokamak operation *Nucl. Fusion* **49** 104007
[50] Luce T. 2005 Development of steady-state advanced tokamak research in the DIII-D tokamak *Fusion Sci. Technol.* **48** 1212–25
[51] ITER Physics Expert Group on Confinement and Transport, ITER Physics Expert Group on Confinement Modelling and Database and ITER Physics Basis Editors 1999 Chapter 2: plasma confinement and transport *Nucl. Fusion* **39** 2175–249
[52] Takase Y. *et al* 2026 Plasma ramp-up and sustainment scenarios for Tokamak Energy's Fusion Pilot Plant. under review *EPJ Web Conf.* **346** 02002
[53] Lopez N.A., Alieva A., McNamara S.A.M. and Zhang X. 2025 Fast physics-based optimization for electron cyclotron current drive *Plasma Phys. Control. Fusion* **67** 055012
[54] Maartensson E.N.J. and McNamara S.A.M. 2024 Early design workflow and progress of Tokamak Energy's high-field spherical tokamak fusion pilot plant for the US DOE Milestone-Based Fusion Development Program. Abstract BO05.004 *Presentation at the APS Division of Plasma Physics Meeting*
[55] Welch N. and Marsden C. 2025 SCOPE: Simple Coil Optimization for Plasma and Engineering (arXiv:2512.16546)
[56] Alieva A., Lopez N., McNamara S.A.M., Takase Y. and Zhang X. (TE FPP Team) Progress in the pre-conceptual design of the auxiliary heating and current drive system for the Tokamak Energy Fusion Pilot Plant *EPJ Web Conf.* **346** 02014