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STEP EXHAUST SYSTEM – ARCHITECTURE AND TECHNOLOGY DEVELOPMENT OVERVIEW

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Abstract

Heat and particle exhaust is a fundamental challenge for the STEP Prototype Plant (SPP). The STEP programme is pursuing a spherical tokamak design, in order to take advantage of reduced capital costs compared to conventional tokamak designs. The smaller radius of a spherical tokamak increases the exhaust challenge due to the reduced area available to dissipate heat and manage particle exhaust. The paper presents an overview of the current STEP Exhaust System architecture and technology development activities. The STEP design features an up-down symmetric double null configuration to reduce heat flux seen by plasma facing components (PFCs). Each divertor comprises of a tightly baffled super-X divertor on the outboard leg and shorter inboard legs approaching an X-divertor. Heat and particle loads vary spatially throughout the divertor. All PFCs are water-cooled with tungsten armour as the plasma facing material, but with different cooling configurations providing a balance of performance and complexity. A gas cooled Divertor cassette design with Eurofer97 as structural material aims to reduce activated waste inventories and provide high grade of heat. Tungsten (W) alloys will be embedded to aid shielding performance. Successful realisation of the STEP Exhaust System requires addressing many technical challenges. Key challenges include characterisation of relevant materials, development of reliable manufacturing and joining methods and high heat flux (HHF) testing of PFCs. Recent technology developments have addressed some of these challenges and increased confidence in the engineering design. Results of these manufacturing trials will be presented regarding tungsten microbrush manufacture, tungsten tile to Cu alloy joining and manufacturing trial for the divertor cassette section. An overview of planned development activities over the coming years will also be presented.

1. INTRODUCTION

Spherical Tokamak for Energy Production (STEP) is a UK-led programme aiming to deliver a first-of-a-kind fusion prototype powerplant [1]. The section view of STEP tokamak machine is shown in Fig. 1. The STEP key technical parameters have been summarised in Table 1. The STEP plasma places extreme heat, particle and structural loads onto the divertor. In addition to these loads, the divertor must also manage the wider powerplant requirements relating to safety, net power generation, tritium breeding and plant availability. The STEP programme is pursuing a spherical tokamak design to take advantage of reduced capital costs compared to conventional tokamak designs. The smaller radius of a spherical tokamak increases the exhaust challenge due to the reduced area available to dissipate heat and manage particle exhaust. Integrating the divertors into a prototype powerplant, as opposed to an experimental device, introduces additional challenges such as the need to demonstrate commercially relevant component lifetimes in a fusion environment, enable efficient maintenance for viable plant availability and introduces constraints such as ensuring manufacturing cost and levels of activated waste are reduced as far as practicable. The successful realisation of the STEP exhaust system is therefore reliant on understanding and balancing a wide range of trade-offs, such as balancing heat load management and compatibility with the power generation cycle when specifying coolant temperatures. The paper will present an overview of the current STEP Exhaust System architecture and ongoing/upcoming technology development activities.

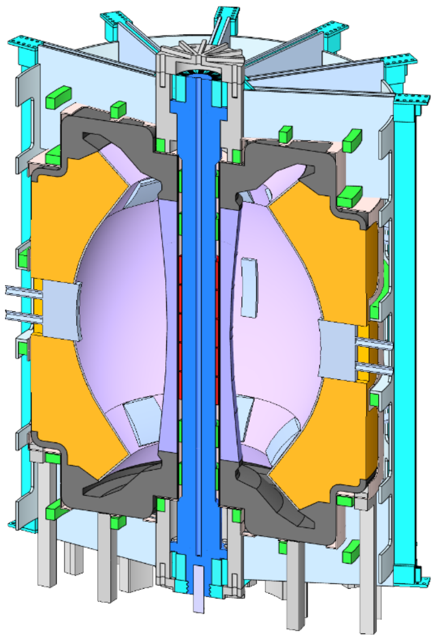


FIG. 1. Section view of STEP tokamak machine (Exhaust system in dark grey).

TABLE 1. STEP key technical parameters

Parameter	Value
Fusion Power	1.6-1.8 GW
Net electric power	100-200 MWe
Inboard Build	1.9 m
Major Radius	4.275 m
Magnetic Field	3.0 T
Plasma current	20-25 MA
Elongation	~3
Triangularity	~+0.5
Plasma Edge	Edge Pedestal
HCD Mix	EC + EB
Primary Divertor Configuration	Dynamic DN
Secondary Divertor Config (Inboard)	Flat Top: X Type Ramp Up: Perpendicular
Secondary Divertor Config (Outboard)	Extended Leg
TF Conductor Type	REBCO
Primary Maintenance Access Route	Vertical
Remountable Toroidal Field Coils	12 TF coils (3 Remountable joints per TF)
Peak Steady State Divertor Heat Flux	<20 MW/m^2
Tritium Breeder Material / Breeding ratio	Li_2O / >1.0
Centre Column / Divertor Coolant	D_2O & H_2O / CO_2 & H_2O
OB First Wall, Blanket, OB Limiter Coolant	CO_2 / He
Blanket Coolant Outlet Temperature	600°C

2. STEP EXHAUST SYSTEM – ARCHITECTURE

The main functions of the STEP exhaust system can be grouped into three categories. Firstly, managing heat and particle loads, addressed through the so-called 'zoned' approach PFCs with three different PFC concepts, i.e. Monoblock, Brushblock, Tile on Heat Sink (ToHS) PFCs. Secondly, shielding superconducting magnet coils, whereby a gas cooled divertor cassette with embedded W alloy materials is used to balance wider needs. Lastly, extracting impurities, achieved through optimised wall shape and purposely positioned vacuum pumping duct. This section mainly focuses on how the main functions are achieved with the current system architecture.

2.1. Wall shape

The STEP divertor design features an up-down symmetric Double-Null (DN) configuration to reduce heat flux (HF) seen by PFCs. However, control system will not be able to maintain a perfect double null, but will rather oscillate up and down, with amplitude and frequency of the oscillations tuned. The dynamic DN assumes variation in power sharing between upper and lower. As shown in Fig. 2, the plasma design has gone through multiple iterations, driven by a combination of increasing fidelity in divertor modelling and iteration of overall machine parameters (e.g. major and minor radii) [2]. Engineering constraints have also driven many aspects of the integrated divertor design. For example, maintenance considerations have placed spatial constraints on the allowable radius of the inner and outer strike points to allow space for component removal, while manufacturing constraints have led to straight surfaces being preferred to curved surfaces in the wall shape where possible. Maintaining the field line angle at the divertor targets within acceptable limits to avoid damaging PFCs is a key driver in the development of the plasma scenario and coilset design. The divertor plasma will operate in detached mode, maintaining heat and particle loads within engineering limits. With pronounced detachment, the peak heat loads on PFCs during steady-state operation will be below 10 MW/m2 and electron temperatures below 5 eV, which is crucial to limit surface material erosion.

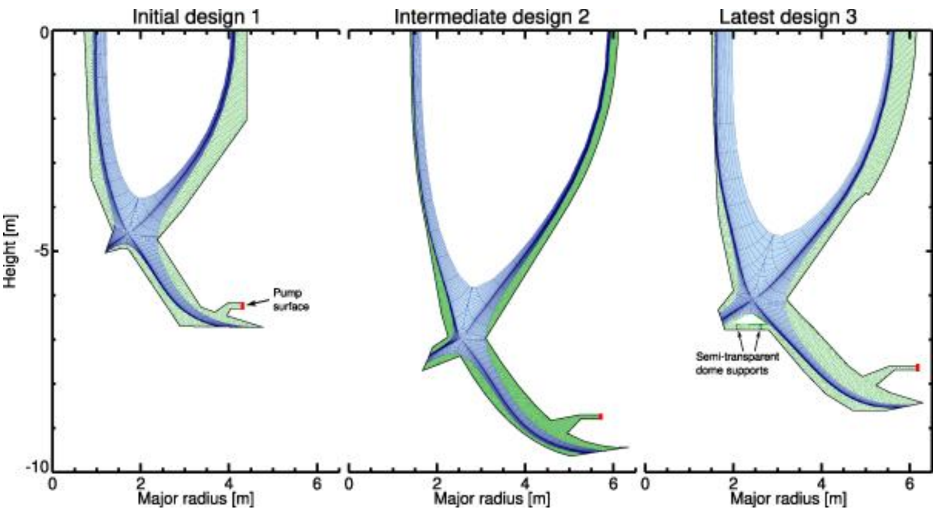


FIG. 2. STEP exhaust design iterations (Reprinted from [2] with permission.)

The latest STEP exhaust wall shape has been shown in Fig.3, the key features include:

— **Inboard**: short inboard legs approaching an X-divertor. STEP as a spherical tokamak, with a smaller major radius, has less physical space to distribute the power, particularly on the inboard side. If STEP adopted a conventional divertor design, the unmitigated heat flux would be three times higher compared to EU DEMO concepts;
— **Outboard**: a tightly baffled super-X divertor/Extended leg. In MAST Upgrade, Super-X has already demonstrated at least a tenfold reduction in the heat on divertor materials [3]. If STEP adopted a conventional divertor design, the unmitigated heat flux would be twice as high as EU DEMO concepts;
— **Dome**: A 'dome' structure is included between the inboard and outboard legs, facilitating transport of neutral particles to the outboard legs, where the vacuum pumps are located, while minimising core plasma pollution. Previous study shows reduction of dome furthers detachment and free neutral exchange between the inboard

and outboard legs [4]. However, although removal of the dome could reduce the power loading of the targets by a factor of 2 for the same detachment state, a factor 10 higher pumping speed is necessary to keep the Helium density in the core at the same level and to maintain the same fuel throughput. Therefore, the dome will ultimately involve a trade-off between pumping efficiency and detachment access;

— **Pumping**: Vacuum pump duct opening is positioned to achieve a balance of achieving sufficient pumping speeds and allowing sufficient pressure to build up near the outboard target to ensure detachment;

— **Shielding**: The divertors protect the superconducting magnet coils located above and below the tokamak, an essential function to demonstrate a pathway to a commercially relevant machine lifetime. The space required to integrate sufficient shielding material places a constraint on the design of the divertor wall profile.

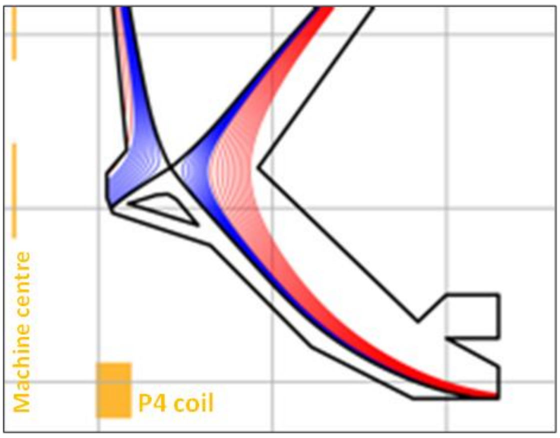


FIG. 3. STEP exhaust wall shape

2.2. Subsystems

To fulfil various functions of the exhaust system, facilicate integration with the tokamak machine and meet the needs of maintenance and inspections, the STEP exhaust system has been divided into several subsystems.

2.2.1. Zoned-approach PFCs

Heat and particle loads vary spatially throughout the divertor under a range of normal and off-normal operating conditions [5]. To achieve a balance of performance and reduced complexity, the STEP divertor uses different PFC designs at different locations. All PFCs are water-cooled with tungsten armour as the plasma facing material, but with different cooling configurations (Fig. 4). Heavy water (D_2O) rather than light water (H_2O) has been selected as D_2O offers Tritium Breeding Ratio (TBR) improvements due to the reduced absorption of scattered neutrons compared to H_2O.

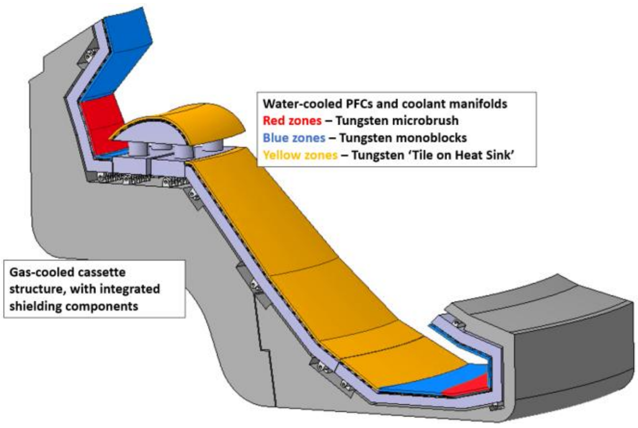
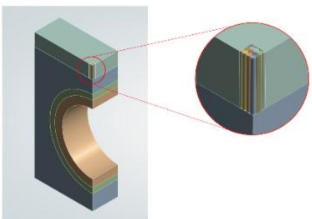
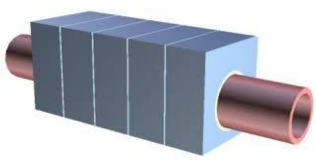
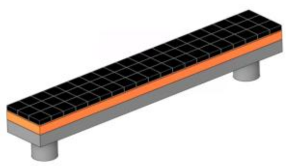


FIG. 4. One segment of Divertor (colour-coded) with PFCs and Cassette

The PFC section of the Divertor designs is divided into three ‘zones’ to allow a tailored approach. Key parameters of the zoned-approach PFCs have been summarised in Table. 2, which includes the HHF loads, key features, materials, coolant, design rationales and an illustration of the concept.

- **‘High’ Heat Fux regions**: Where (quasi) steady-state loads of up to 20 MW/m^2 may be seen during ramp-up stage, Monoblock-like design has been chosen. Monoblocks are a mature technology, having been developed over many years for ITER, although neutron irradiation impacts are still to be fully assessed;
- **‘Medium’ Heat Fux regions**: The DN configuration with extended leg has resulted in a large surface area. Therefore, where surface heat flux is not expected to exceed 10 MW/m^2, larger PFCs with reduced heat flux handling capability can be used. This will reduce the total part count significantly and is expected to increase reliability. These PFCs will consist of tungsten tiles on a Cu alloy heatsink;
- **Strike points regions**: The technology choice for the strike points is driven in part by the need to withstand Type-I Edge Localized Modes (ELMs), addressing a key risk that ELM-free plasmas cannot be guaranteed through all stages of operation. Tungsten ‘micro brushes’ are used at the strike points; a technology which offers increased resilience to repeated transient events such as ELMs compared to alternatives such as tungsten monoblocks due to the removal of surface cracking and subsequent crack propagation as a failure mode. This will increase the allowable time for the control system to react and mitigate ELMs if they occur. The micro-brush is a less mature technology than monoblocks and so technology development, particularly regarding resilient manufacturing processes, is required in order to achieve this performance benefit. To improve the robustness and maximise learning from Monoblock development, Brushblock PFC, i.e. Microbrush on a Monoblock, has been baselined as strike point PFC technology to replace Liquid Metal Armour (LMA) concept. Initially LMA was selected primarily due to the potential for handling transient events such as ELMs, thanks to the enhanced vapour shielding effect compared to solid armour options. Despite promise of technology, LMA was deselected during subsequent design iterations due to integration complexity and very low maturity.

TABLE 2. Key parameters of the zoned-approach PFCs

Locations	Strike Point	High Heat Flux zones	Medium Heat Flux zones
Steady-State (flat-top) HF	10MW/m^2	10MW/m^2	10MW/m^2
Steady-State (ramp-up) HF	20MW/m^2	20MW/m^2	10MW/m^2
ELMs HF	0.1-1MJ/m^2	N/A	N/A
Key features	Brushblock	Monoblock	Tile on Heat Sink
Materials	W wire/W/Cu alloy	W/Cu alloy	W/Cu alloy/Steel
Coolant	D_2O	D_2O	D_2O
Rationale	Crack resilient during Type-I ELMs.	ITER proven technology.	Significantly reduced part count.
Concept illustrations			

2.2.2. Gas cooled Cassette

The divertors provide shielding to protect the superconducting magnet coils located above and below the tokamak. The shielding function is mainly supplied by the “Cassette” subsystem, which also structurally support PFCs. In contrast to conventional water-cooled austenitic steel cassette design, the STEP divertor cassette adopts Eurofer97 steel [6], which is a Reduced Activation Ferritic Martensitic (RAFM) steel, to reduce intermediate level waste (ILW). The use of Eurofer97 is facilitated by using gas coolant that allows the material to operate within an acceptable temperature range (350 to 550°C) to mitigate the risk of low temperature embrittlement under irradiation. Since using separate coolant flow loops for the PFCs and the cassettes in the divertor has been strongly recommended for several reasons, primarily because the high flow rates required in the PFCs would result in prohibitive pumping power requirements if the coolant was pumped through the entire cassette structure, this will not significantly complicate the cooling systems considering gas loop has been used elsewhere in the machine. Instead, a gas cooled cassette will not only provide useable high grade of heat to the power generation cycle but offer better safety implications in case of an event Loss of Coolant (LOCA) and slightly improved TBR. However, this leads to reduced shielding performance compared to water cooling, which can be mitigated via the addition of tungsten-based shielding materials within the cooling channels. The increased complexities pose significant challenges to the manufacturing, which should be demonstrated in early stage. The required thickness of shielding components constrains the allowable location of the plasma facing surface affecting the shaping of the dome and introducing a need to move the plasma facing surface of the outboard divertor leg closer to the separatrix. Despite reduced heat removal capability, gas is still able to remove the volumetric heating but with a cost of higher pumping power. Initial studies show the increased pumping power requirement is negated by the increased power generation due to higher grade of heat. Although this design foresees additional challenges comparing with water cooled concepts, future commercial Fusion powerplant will need to minimise the ILW, this could work as a technology demonstration.

2.2.3. Vacuum pumping duct

The vacuum pumping duct is positioned to achieve a balance of achieving sufficient pumping speeds and allowing sufficient pressure to build up near the outboard target to ensure detachment. A dome is included between the inboard and outboard divertors, allowing transport of neutral particles towards the outboard divertor minimising transport into the core plasma. The area underneath the dome needs to be semi-transparent to allow sufficient flow of neutrals, adding a constraint on the available space for coolant pipes and structural supports.

3. STEP EXHAUST SYSTEM – TECHNOLOGY DEVELOPMENT

Successful realisation of the STEP Exhaust System requires addressing many technical challenges. Key challenges include characterisation of relevant materials, development of reliable manufacturing and joining methods and HHF testing of PFCs. Recent technology developments have addressed some of these challenges and increased confidence in the engineering design. Results of these manufacturing trials will be presented regarding tungsten microbrush manufacture, tungsten tile to Cu alloy joining and additive manufacture of steels for the divertor cassette. An overview of planned development activities over the coming years will also be presented.

3.1. Brushblock PFC manufacture

As described in section 2.2.1, Brushblock PFC concept has been baselined for STEP divertor system. The low maturity of this technology means early-stage R&D is required to improve the confidence level. Micro-structured tungsten has been studied to demonstrate its superior ability to handle repetitive transient thermal loads or thermal shocks [7-9]. The possible paths towards the large-scale production of W wires (W_W) assembly, including techniques for realizing feasible joints with Cu, steel or W, have been presented [10]. In that study, selective laser melting (SLM) was used as the W-W_W joining technique, looking promising although not HHF tested. To further develop the Brushblock PFC concept, reliable joining of tungsten brush to tungsten substrate with more mature manufacturing methods has been identified as a key technology challenge.

To address this issue, an initial manufacturing trial has been carried out. Multi-stage approach has been adopted in the trial, it starts with “bristle pack” manufacturing which includes machining of the W wires. Optical microscopy, Scanning Electron Microscopy (SEM), and Energy-Dispersive X-ray Spectroscopy (EDX) were used to characterise components. Ultrasonic testing trials have showed some promising results, although improvement is required. By demonstrating that bristle pack can be successfully joined to W block at PFC relevant scale, the manufacturing trials show promising results regarding bristle pack manufacture (Fig. 5) and its bonding to tungsten substrate (Fig. 6).

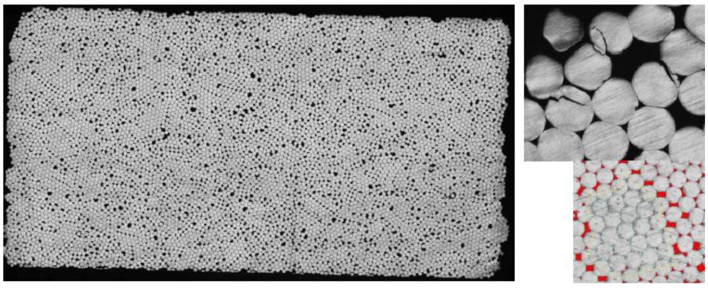


FIG. 5. Bristle surface post-joining (left) and close-up under optical microscopy.

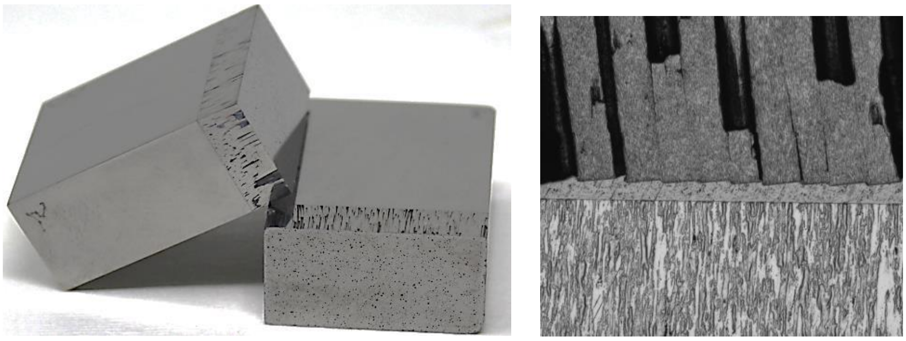


FIG. 6. Brushblock initial demonstrators (left) and SEM imaging of the Ww to W substrate joint (right).

Follow-on HHF testing is planned over course of 2025, focussing on steady-state and representative transient loads. To enable the steady state test, a heatsink tube will be joined to the demonstrator once a hole is drilled in the W substrate. Further development planned in future years is to further improve bristle packing density, uniformity and repeatability, refine joining process, and develop Non-Destructive Testing (NDT) method. Then produce Brushblock component level mock-ups and go through performance test which includes HHF test, fuel retention etc.

3.2. ToHS PFC experimental study

As described in section 2.2.1, ToHS PFC will cover majority of the Divertor surface area to significantly reduce the total part count. Although it is labelled as "Medium Heat Flux zones", the design surface heat flux will be up to 10MW/m^2 to provide enough margin to accommodate various tolerances and uncertainties. The global structural integrity of the component is dependent on the quality of the W-Cu alloy joint, since a failure of a single tile could trigger a cascade failure of neighbouring tiles [11]. In the ITER R&D activities, the eligibility of this concept was demonstrated for a medium heat flux range (5–11 MW/m^2), although the design heat flux for ITER divertor dome is 5 MW/m^2. Moreover, STEP, as a prototype plant, will see significantly higher neutron damage than ITER does. Limited data shows that even low level of neutron-induced degradation may substantially reduce the HF handling capability [12], in contrast to more robust Monoblock concept. With joint embrittlement after the neutron irradiation, dominant failure mechanism may change from a progressing ductile fracture to an abrupt brittle one, resulting in a spontaneous detachment of the complete tile [13].

To address the concern above, a ToHS PFC mock-ups manufacturing activity has been launched to validate the manufacturing process and technology performance. Similar to the Brushblock PFC manufacturing trial, this effort also features a two-stage approach. In the coupon trial stage, dozens of disc type W/Cu alloy joints (Fig. 7) have been produced to investigate a range of joining options, with variants of joining methods, joining process parameters, and compliant layer selections etc.

Both STEP Divertor and Inboard First Wall (IFW) adopt a ToHS design, but with different tile and heatsink geometries and loading conditions. Facing similar challenge, this technology development programme has been scoped such that test results can inform the design of both systems. Therefore, two types of demonstrators have been produced according to the differences of two systems. The combinations of these variances have been purposely designed to reflect the outcome of supporting analyses and simulations. The ToHS PFC manufacture is in its final stage as time of writing, where the demonstrator mock-ups are subjects to further HHF test. The final results will be used to inform the ToHS PFC tile sizing design activities and potential investigation on neutron irradiation effects.

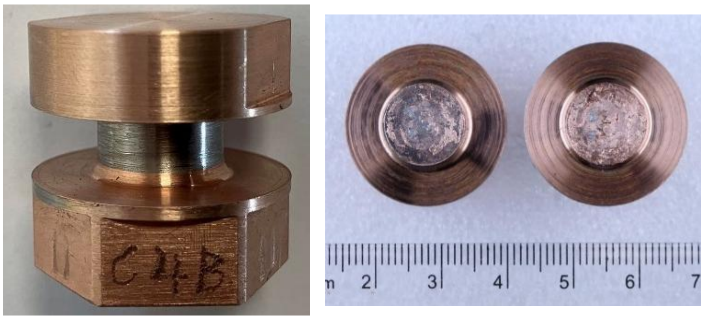


FIG. 7. Torque test results of W/Cu alloy joint.

3.3. Manufacturing trial for the divertor cassette section

As described in section 2.2.2, the cassette bodies are to be manufactured from Eurofer97 steel, which is a RAFM steel. The cassette body features numerous interconnected internal chambers across a complex geometry. The nominal envelope of the cassette body measures metres scale in each dimension. This will not only make handling challenging, but also complicate the limited inspection access. Finally, the cassette must be manufactured to extremely demanding specified tolerances, which will make distortion control from significant thermal input from welding and subsequent post weld heat treatment a key area of concern. All these will present a series of significant manufacturing challenges for the cassette design.

The preferred manufacturing route for the cassette body has yet to be defined, however traditional manufacturing methods all present challenges. For example, the addition of the shielding material to the internal chambers means that machining from solid wrought product is not feasible. This was the method that was adopted for the cassette bodies for ITER, which was complex and resulted in up to 80% material wastage.

The objective of the activity is to manufacture Ferritic Martensitic steels using a modified section (Fig. 8) from the divertor cassette body as the basis for a technology demonstrator. The difficulty in obtaining Eurofer97 in significant quantities and in the correct product form means that surrogate materials, such as grade 91 steel, are needed for development activity. After down-selected post welding heat treatment, a series of metallography and mechanical testing have been conducted to characterise the deposition trials following dimensional check and NDT. The mechanical properties did not vary significantly between the code/standards recommendations and test results. Dimensional inspection and Phased Array Ultrasonic Testing (PAUT) also show promising results.

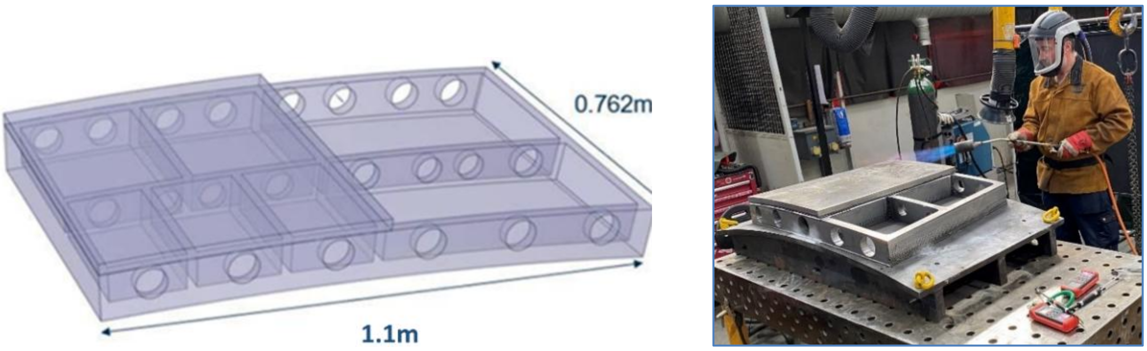


FIG. 8. CAD of Cassette modified section (left) and final demonstrator (right).

3.4. Technology development plan

To align with the schedule of the STEP programme, the Exhaust system is targeting to achieve at least Technology Readiness Level (TRL) 4 by 2029. To achieve the required TRL, the Exhaust System has many technical challenges to overcome. A technology development plan covering the materials, manufacturing and component testing activities for the Exhaust system is shown in Fig. 9 against the timeline and expected TRL. The plan focuses on small mock-ups and/or medium scale qualification prototypes where possible. These activities can be largely categorised into different groups depending on the stage/nature of the subsystem design, allowing identification of priorities and main areas of focus. For example, Monoblock has been considered as ITER proven technology, domestic supply chain development will be the focus for Monoblock PFC. In contrast, the Brushblock PFC has a lower maturity level, which needs more intense R&D activities in short term. The ongoing and upcoming ToHS PFC experimental activities will inform the design in terms of tile sizing and HF handling capability under foreseen neutron irradiation.

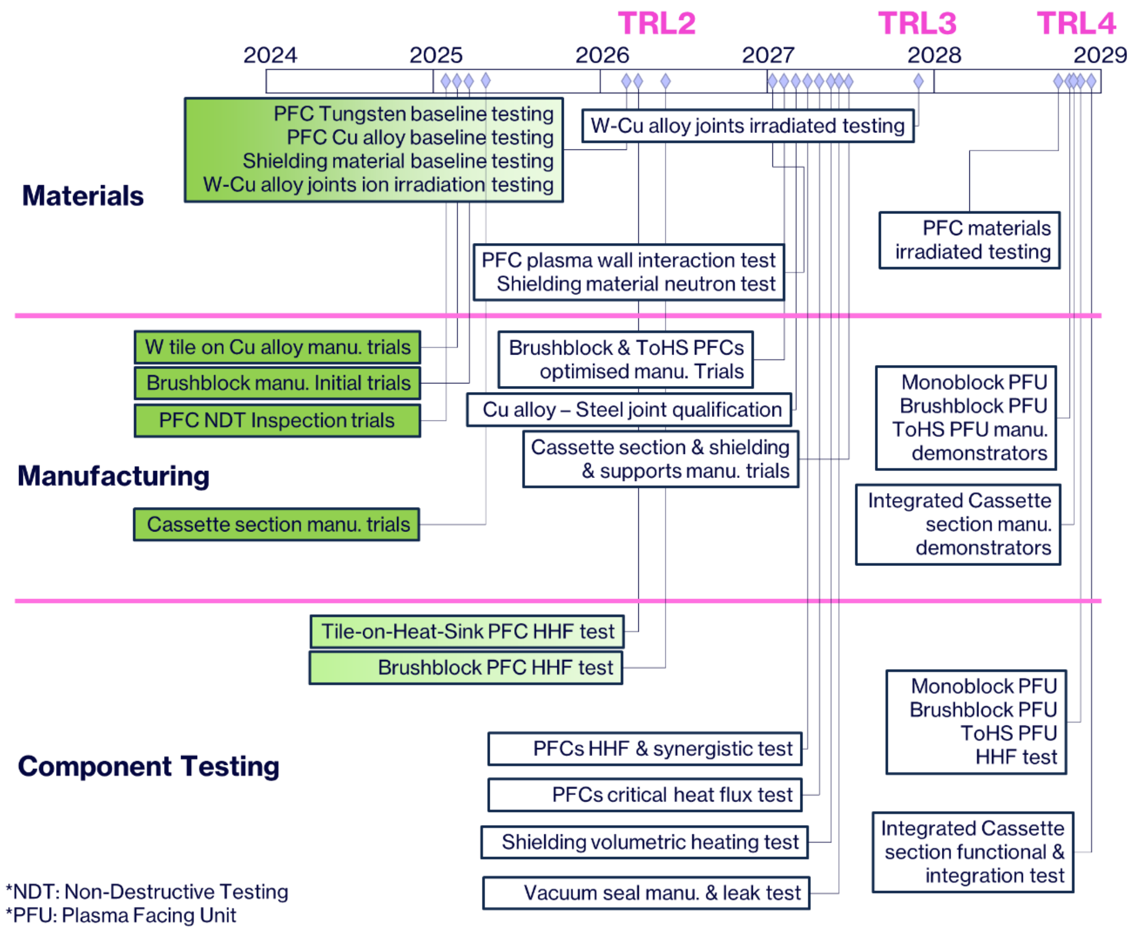


FIG. 9. Exhaust system technology development plan in upcoming years.

4. CONCLUSION

Heat and particle exhaust is a fundamental challenge for any fusion device. The spherical tokamak design has made it even more challenging for STEP Prototype Plant. The STEP Exhaust system architecture has been presented, balancing the requirements from exhaust plasma and engineering limits. After multiple iterations, the latest STEP Exhaust system features an up-down symmetric Double-Null primary configuration. The secondary configuration comprises of an X-divertor and super-X divertor on inboard and outboard legs, respectively, with an in-between “dome” structure. The STEP exhaust system has been designed using a zoned-approach PFCs to manage varied heat flux across a large surface area and a gas cooled cassette aiming to reduce ILW and provide high grade of heat.

The STEP Exhaust System has a range of technical challenges to address. Key challenges include characterisation of relevant materials, development of reliable manufacturing and joining methods and HHF testing of PFCs. From R&D for low maturity concept to supply chain development for proven technology, key technology development activities have been prioritised in a structured approach. A high-level overview of planned development activities over the coming years to support Tranche 2a objectives in 2029 has also be presented. Recent technology developments have addressed some of these challenges and increased confidence in the engineering design. The manufacturing trials regarding tungsten microbrush manufacture, tungsten tile to Cu alloy joining and manufacturing trial for the divertor cassette section have showed initial promising results. Performance tests including HHF tests are planned in 2025 to validate the adopt technologies and inform the design.

Looking ahead, as the system continues to evolve, it's foreseen that both wall shape and subsystem design will be tuned to keep striking the balances of wider measures of effectiveness of the SPP. The technology development activities during STEP Tranche 2a (2025-2029) will feed into detailed design and manufacture during 2030s.

ACKNOWLEDGEMENTS

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