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STEP INBOARD SYSTEM – ARCHITECTURE AND TECHNOLOGY DEVELOPMENT OVERVIEW

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Abstract

The STEP programme has the mission to deliver a UK prototype fusion energy plant, targeting 2040 and a path to commercial viability. The design of inboard build is a fundamental challenge for the STEP Prototype Plant (SPP). The spherical tokamak geometry of the SPP means the inboard radius drives the size, and hence overall cost, of the machine. The STEP Inboard System protects the central column magnets from heat, particle, and neutronic loads while using this captured energy for power generation. Innovative designs and novel technologies are needed for the STEP Inboard System to fit these spatial constraints, particularly considering the complex integration with the magnet systems. The architecture of the inboard system consists of three distant radially-layered regions: i. Vacuum Vessel & Low-Pressure Shield, ii. High-Pressure Shield, and iii. Inboard First Wall.

1. INTRODUCTION

The STEP programme has the mission to deliver a UK prototype fusion energy plant, targeting 2040 and a path to commercial viability. The STEP Prototype Plant (SPP) aims to demonstrate net energy, fuel self-sufficiency and a route to commercial operations with respect to component lifetime and maintenance durations [1]. The design of the SPP, shown in Fig. 1 and listed Table 1., include specific features which drive the design of the STEP Inboard System such as a high aspect ratio, superconducting magnets, no inboard breeding, upper and lower magnet joints, and a vertical maintenance strategy [2,3].

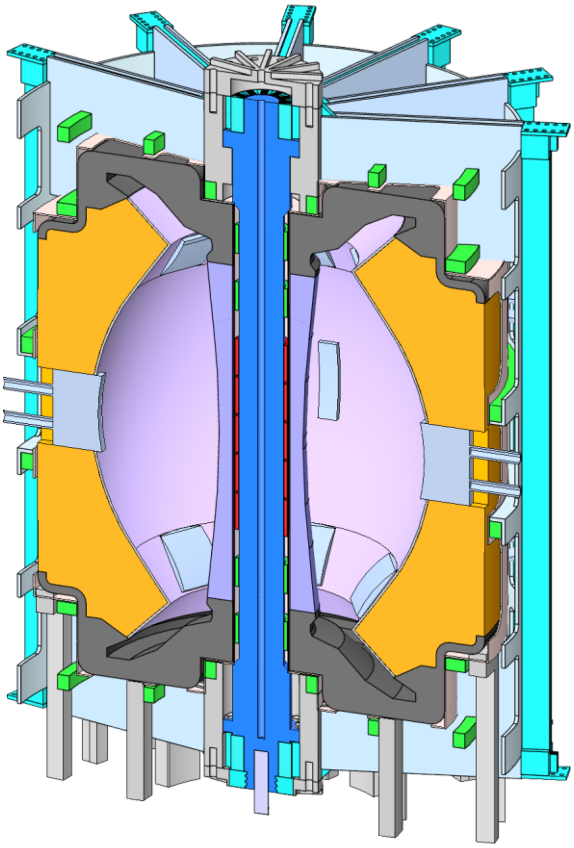

FIG. 1. STEP Prototype Powerplant

TABLE 1. STEP PARAMETERS

Parameters	Values
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
Outboard First Wall, Blanket, Outboard Limiter Coolant	CO_2 / He
Blanket Coolant Outlet Temperature	600°C

The STEP Inboard System is located in the centre column of the SPP and consists of the vacuum vessel and in-vessel components on the inboard, but does not include the inboard magnets. The primary functions are to:

— Handle to the steady-state and transient heat loads from the plasma;
— Protect the inboard magnets from neutron and gamma radiation from the plasma;
— Create the vacuum barrier in combination with conditions;
— Provide useful heat for power generation;
— Use minimal radial space.

The STEP programme has taken an iterative approach to the overall machine where designs are selected, rapidly assessed and then continued or modified depending on the outcome. The assessment is applied both locally by technical experts in specific areas and holistically balancing the competing needs of the different systems. This approach means the SPP design is continually changing so the design presented here may change as a result of future design iterations [2].

2. INBOARD DESIGN

The STEP Inboard System is situated within the inboard radial build between the inboard magnets and the plasma wall gap, as shown in Fig. 2. The concept design for the inboard system consists of three distant radially-layered regions: i. Vacuum Vessel & Low-Pressure (VV & LP) Shield, ii. High-Pressure (HP) Shield, and iii. Inboard First Wall (IFW), as shown in Fig. 3 [3]. The different regions of the inboard systems have distinct functions which have led to their different designs. Heavy water was chosen to as the coolant for the HP Shield and IFW increase tritium breeding within the outboard breeding zone [4,5]. The nuclear heating decreases exponentially through the thickness of shielding and the coolant operating conditions match this; the HP Shield uses high pressure coolant with high heat transfer properties to handle the high nuclear heating and the VV & LP Shield used low pressure coolant with low heat transfer properties to handle the low nuclear heat loads. The naming of the "low pressure shield" and "high pressure shield" originates from the difference in coolants used. The design of the individual components and the overall assembly of the inboard need to consider a variety of structural loads imposed on it: self-weight, thermal stresses, coolant pressures and electromagnetic loads from plasma events.

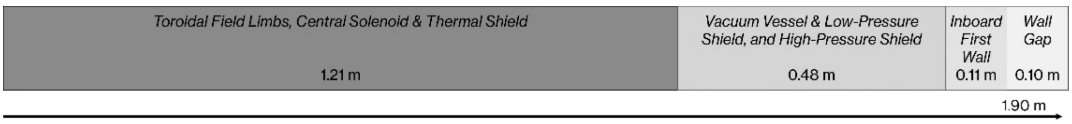

FIG. 2. STEP Inboard Radial Build

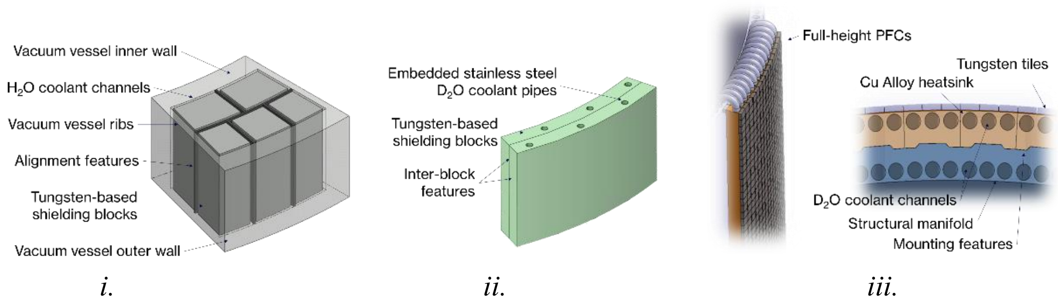

FIG. 3. Architecture of the STEP Inboard System: i. Vacuum Vessel & Low-Pressure Shield, ii. High-Pressure Shield, and iii. Inboard First Wall

2.1. Vacuum Vessel & Low-Pressure Shield

The primary functions of the VV & LP Shield are to attenuate remaining neutrons to magnet-compatible levels and form the inboard section of the vacuum containment. These two functions create a constraint as the design must accommodate shielding material within it without degrading vacuum containment performance.

The concept design for the VV & LP Shield consists of Tungsten-based shielding blocks located within the double skin 316LN vacuum vessel, as shown in Fig. 3. The internal cavity of the vacuum vessel has radial ribs to provide structural integrity, and light-water cooling channels for thermal management and for attenuating neutrons. The VV & LP Shield includes both inter-block and block-to-vacuum-vessel alignment features to minimise radiation shine paths, position the blocks, and compliance to allow for irradiation swelling and differential thermal expansion [3].

There are key challenges with the proposed VV & LP Shield concept design. The shielding performance is exponentially dependant on the density of the tungsten-based shielding material and what realistic manufacturing densities can be achieved for these large shielding blocks in the design. Tungsten-based shielding materials are brittle so the design needs to minimise the stresses on the shielding block, particularly tensile stresses. The volume of structural material and coolant needs to be minimised as these take away space for shielding material and reduce shielding performance. The inboard section of the vacuum vessel needs to form good vacuum seals with the other vacuum vessel sections to create the required ultra-high vacuum conditions. The inboard system is structurally supported at it upper and lower ends. The internal structural integrity of the VV & LP Shield is primarily provided by the vacuum vessel walls and ribs. The structural design needs to withstand self-weight loads, electromagnet loads from the plasma, thermal expansion stress and radiation swelling stresses, and transfer these loads to the structural supports.

2.2. High-Pressure Shield

The primary functions of the HP Shield are to attenuate remaining neutrons, in conjunction with the VV & LP Shield to magnet-compatible levels, and to provide useful heat via the coolant for power generation. These constraints require design which includes maximum amounts of shielding material for and the component operating at high temperature to provide high outlet coolant temperatures.

The concept design for the HP Shield consists of tungsten-based shielding blocks with full-height embedded stainless steel coolant pipes, as shown in Fig. 3. The tungsten-based shielding blocks absorb the neutron and gamma irradiation, and conducts the nuclear heat generated to the embedded pipes. The embedded pipes contain high-pressure heavy water operating at high temperature which extracts the heat for useful power generation. Inter-block features are used to structurally connecting and aligning the blocks, whilst minimising radial shine paths [3].

There are key challenges with the proposed HP Shield concept design. The structural material and coolant volume fraction must be minimised in order to maximise the shielding material content and performance, while still meeting the thermal and structural requirements. The significant self-weight of the HP Shield (100s tonnes) needs sufficient structural supports in the highly constrained regions below and above the inboard. Like with the VV & LP Shield, tungsten-based shielding materials are brittle so the design needs to minimise the stresses on the shielding block. There are several manufacturing challenges around the HP Shield concept design, including: the forming of large, high-tolerance, shaped blocks from tungsten-based shielding materials, the method of embedding steel pipes into shielding blocks and the block-to-block joining to assemble the HP Shield.

2.3. Inboard First Wall

The primary functions of the IFW are to protects other inboard components (HP Shield, VV & LP Shield, and Magnets) from heat & particle loads of the plasma and provide useful heat via the coolant for power generation. As part of the inboard build the IFW must take up minimal radial space to maximise space for shielding by the other inboard components. The IFW also needs to align with neighbouring components to form part of the continuous overall SPP first wall profile [2], shown in Fig.1. The IFW is subjected to high surface heat loads from the plasma and high nuclear heating from the radiation environment. The IFW also has to withstand the erosion by the particle bombardment from the plasma.

The IFW concept design consists of Plasma Facing Components (PFCs) attached to a structural manifold, as shown in Fig. 3. The IFW PFC uses tungsten armour tiles on a copper alloy heatsink. The IFW PFC are full-height of the inboard and are cooled via internal channels carrying high-pressure heavy water operating at high temperatures, similar to the HP Shield, to provide useful heat for power generation. There are only coolant connections at one end of the IFW so the coolant enters through the PFC and returns through the structural manifold. The structural manifold is made stainless steel and includes mounting features for the IFW PFCs [3].

The key challenges with the IFW include the manufacturability of long-length, ~10 m, PFCs, particularly with curved profiles to align with first wall profile. There is significant uncertainty in the expected thermal and structural loads on the IFW; specifically, disruption load cases which can cause extreme transient heat loads and large electromagnetic forces. These disruption load cases can be design-driving for both the thermal and structural performance of the component. Also the IFW is situated on the mid-plane near to the plasma core which is an area of high neutron and gamma irradiation. The irradiation-effects on these materials needs to be understood and the component design needs to allow for these effects (swelling, embrittlement, reduction in thermal conductivity, loss of strength) while still remaining operational.

3. RECENT TECHNOLOGY DEVELOPMENT ACTIVITIES

There has been a programme of recent technology development activities in support of the STEP Inboard System in addition to the concept design efforts. These activities have focused on the materials, manufacturing and component testing, listed below.

3.1. Materials

3.1.1. Production of stainless steel plates

Stainless steel is used as the structural material with the HP Shield and IFW. The specific stainless steel alloy was chosen for it's high operating temperature.. However, there is no established supply chain for the material. To address this, trials were undertaken for small-scale production of stainless steel plates. Compositional analysis showed that the produced stainless plates were within specification but also highlighted the challenges around meeting the specific elemental content which is required for the swelling resistant behaviour.

3.1.2. Producing tungsten-based shielding material

Tungsten-based shielding materials are used in the VV & LP Shield and the HP Shield. These materials are effective at neutron and gamma shielding due to their high density and elemental composition. However, there are challenges in forming parts from these materials and gaps in material properties. To address this, trials were undertaken to produce tungsten-based shielding material samples. The trials included composition exploration, forming trials, phase analysis, process scale-up and materials testing, including ion-irradiated samples.

3.2. Manufacturing

3.2.1. W tile on Cu alloy manufacturing trials

The IFW design includes tungsten armour tiles bonded to a copper alloy heatsink. This dissimilar joint needs to make good thermal contact and be resilient to cyclic loading while operating in a high temperature and radiation environment. Manufacturing trials were undertaken to produce tungsten tile to copper alloy joints. The joints produced were assessed by visual inspection, microscopy of cross-sections and mechanical testing. A small PFC mock-up was also manufactured using the joining methods developed.

3.2.2. Shield block manufacturing trials

The HP Shield and the VV & LP Shield both utilize tungsten-based shielding material blocks in their designs. However, there are challenges around the forming, geometric features and assembly of these blocks. Shield block manufacturing trials were undertaken to address these challenges. The trials included forming shield blocks, machining inter-block features, assembly of blocks and block to block joining. The trials identified viable methods for forming, machining and assembling shield blocks, but highlighted the difficulty in handling and joining between shield blocks.

3.2.3. Shield embedded pipe manufacturing trials

The HP shield design includes coolant pipes embedded within an assembly of shield blocks. Manufacturing methods are needed to embed the pipes into the blocks and make joints between the assembled blocks. Shield embedded pipe manufacturing trials were undertaken to address these issues. The trials culminated in producing a small-scale demonstrator consisting of multiple shield blocks including embedded pipes. The trials identified the challenge of scaling up the bonding method to large, complex geometries.

3.3. Component Testing

3.3.1. High heat flux testing of PFCs

The IFW will be subjected to high surface heat loads from the plasma. HFF testing will be used to assess the IFW PFC, includes analysis on both post-mortem and interrupted samples.

3.3.2. PFC NDT inspection trials

Inspection techniques are required for monitoring operating components and acceptance testing of manufacturing components for the IFW. Inspection trials have been undertaken using ultrasonic non-destructive testing methods on PFC mock-ups and thick tungsten-based material samples. The trials highlighted the challenges in reliably resolving small defects in thick samples.

4. FUTURE TECHNOLOGY DEVELOPMENT ACTIVITIES

The future technology development activities for the STEP Inboard System are focused on addressing the risks associated with the concept design and demonstrating TRL4 by 2029, as shown in Fig. 4. The technology demonstration plan covers materials, manufacturing and component testing.

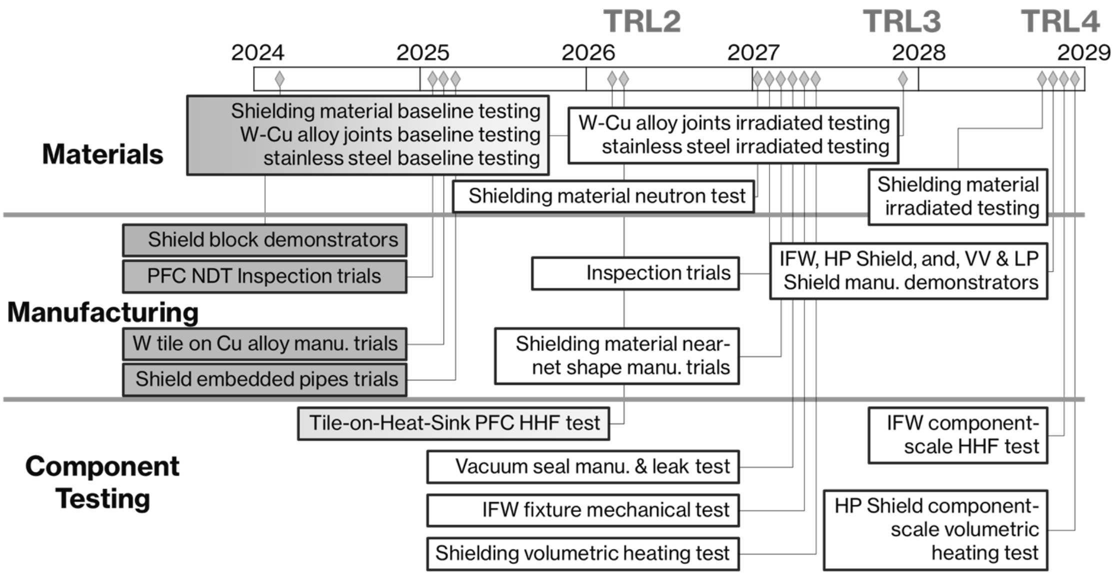

FIG. 4. Technology demonstration plan covering the materials, manufacturing and component testing activities for the STEP Inboard System with the target of achieving at least Technology Readiness Level (TRL) 4 by 2029. Shaded boxes indicate completed and in progress activities.

4.1. Materials

The material testing programme is focused on addressing gaps in material properties data needed for design and understanding the behaviour of materials under irradiation. Particularly, there are gaps around mechanical properties varying with temperature, and corrosion & chemical compatibility. Additionally, the mechanical performance of tungsten to copper alloy joints, used in the IFW, need to be assessed. Irradiation testing will be undertaken to understand how the properties of the novel materials and joints change under the radiation environment expected within the SPP. The testing will involve a combination of ion-irradiation, gas implantation and neutron irradiation. Specific neutron transmission tests will be done on the shielding materials to demonstrate their radiation shielding performance against neutronic modelling predictions. The information on the behaviours and limitations of these materials will be fed back to the design which will then be modified to best incorporate these considerations.

4.2. Manufacturing

The manufacturing programme is on the manufacturing technology challenges associated with the STEP Inboard System design. There are challenges around manufacturing large tile on heatsink PFCs for the IFW. These are addressed with manufacture trials developing tungsten tile to copper alloy joining methods which are then scaled to a larger IFW manufacturing demonstrator which will include a number of tiles, design-specific geometries & tolerances, and assembly of multiple PFCs with fixtures connecting them to a structural manifold. Trials will also address the manufacturing challenges around the tungsten based shielding material. The manufacturing methods will start small then be scales to larger HP Shield and VV & LP Shield manufacturing demonstrators which would include design-specific shield block geometries, structural connections, and assemblies. Inspection techniques will also be developed along side the manufacturing processes which can be used for assessment the as-manufactured and during operation components. The results of the manufacturing programme activities will be fed back to the design which will then be modified to account for the manufacturing practicalities.

4.3. Component Testing

The component testing programme is on demonstrating the thermal, structural and mechanical performance of the technologies within the design. High heat flux tests will be used to demonstrate the IFW heat handling performance.Volumetric heat flux testing will be used to assess the VV& LP Shield and HP Shield ability to handle nuclear heating. The tests would start on small tungsten based shielding material blocks and then move to larger HP Shield component-scale test.Vacuum seal tests will be done to demonstrate the seal design for the VV & LP Shield can achieve the ultra-high vacuum conditions and the alignment features of the sealing mechanism function as expected. Mechanical testing of the IFW fixtures will demonstrate the assembled PFC alignment tolerances can be achieved. Like with the other technology development areas, the results of the component testing will be fed back to validate the analysis results and improve the design.

5. CONCLUSIONS

The STEP inboard system has the challenging functions of protecting the inboard magnets from radiation and handling the high heat loads from the plasma and nuclear heating, while fitting within tight radial space constraints and also providing useful heat for power generation. A concept design has been produced which consists three distinct radial components: VV & LP Shield, HP Shield and IFW. Collectively they achieve the shielding functions and have specific designs in the HP Shield and IFW to handle the higher heat loads and provide high temperature coolant for useful power generation. There are risks around the proposed design which need materials, manufacturing and component testing activities to address; Specifically, the lack of materials data around stainless steel structural material and tungsten-based shielding materials, the properties of tungsten to copper alloy joints and manufacturing methods, and heat load testing of components. Several activities have already begun to address these risks and future activities to are planned to achieve TRL4 for all relevant technologies for the STEP Inboard System.

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