Nuclear design issues of a stellarator fusion power plant with breeder blanket in comparison to tokamaks

U. Fischer¹, L. V. Boccaccini¹, G. Bongioví¹, A. Häußler¹, F. Warmer ²

¹Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany
²Max Planck Institute for Plasma Physics (IPP), Greifswald, Germany

E-mail contact of main author: ulrich.fischer@kit.edu

Abstract. Nuclear design issues of a stellarator power plant, based on the HELIAS (helical-axis advanced stellarator) concept, are presented and discussed in this paper. Specific analyses have been performed in the area of neutronics and structural mechanics utilizing ad-hoc developed simulation models. The neutronics analyses show a favourable breeding performance suggesting that tritium self-sufficiency can be achieved. The shielding performance suffers in regions where minimum space is available. It is concluded, however, that the shielding requirements can be fulfilled with suitable measures. The structural analyses confirm the viability of the segmentation strategy proposed previously for HELIAS 5B. More realistic simulations are recommended, however, using real 3D nuclear heating and temperature distributions and taking into account the internal blanket structure. As general outcome of this work, it is concluded that a HELIAS type power plant can be devised which satisfies the nuclear performance requirements. More detailed analyses are required to proof this for an engineering design which takes into account all stellarator-specific constraints including the maintenance.

1. Introduction

The European Roadmap to the realisation of fusion energy considers the stellarator concept as a possible long-term alternative to a tokamak Fusion Power Plant (FPP). A corresponding R&D programme is conducted by the EUROfusion consortium to advance the stellarator concept with the scientific exploitation of the W7-X experiment. The aim is to optimise the stellarator performance, prove the feasibility for steady-state operation, and, on such a basis, study the prospects of a power producing plant based on the helical-axis advanced stellarator (HELIAS) configuration. An important issue towards this goal is the nuclear performance of a HELIAS type FPP equipped with a tritium breeding blanket. This work addresses this issue based on the achievements of the blanket development conducted within EUROfusion’s Power Plant Physics and Technology (PPPT) programme on a tokamak DEMO and results obtained for HELIAS in the area of neutronics and structural mechanics.

2. The HELIAS concept for a stellarator FPP

The helical-axis advanced stellarator concept (HELIAS) is considered as one of the most promising stellarator concepts and has therefore been the focus of the European stellarator research activities. The first fully optimized stellarator based on this concept is the Wendelstein 7-X (W7-X) experiment which started operation in 2015 in Greifswald, Germany. W7-X is a large-aspect-ratio stellarator with a five-fold symmetry, i.e., consisting of five identical field periods [1]. The name “HELIAS” is usually used to refer to the power plant designs of this stellarator line. The recent HELIAS-5B concept [2] is a direct extrapolation of W7-X to power plant parameters designed for 3000MW fusion power. HELIAS-5B has a major radius of 22m, an average minor radius of 1.8m, a plasma volume of about 1400 m³ and offers a magnetic field in the range of 5-6 T in the centre of the plasma generated by 50 non-planar field coils. Due to this high number of field coils, the maintenance access is constrained by the available space. FIG. 1 shows the HELIAS-5B configuration including large ports which fit in between two field coils. Blanket segments have to be exchanged through such ports and thus must fit to the port dimensions.
3. Nuclear design considerations

The operation of a FPP utilizing D-T fusion reactions to produce power relies on the availability of the tritium fuel. To this end the plasma chamber needs to be covered by a breeding blanket producing sufficient tritium to ensure self-sufficiency. This requires a technically mature design of a tritium breeding blanket suitable for a HELIAS type FPP, the proof that tritium self-sufficiency can be achieved, and the blanket can be maintained in such a configuration. Sufficient shielding performance must be also demonstrated for the assumed HELIAS FPP conditions.

The approach adopted in this work is based on the utilization of the Helium-Cooled Pebble Bed (HCPB) blanket, developed as solid breeder blanket for the tokamak DEMO [3]. The HCPB blanket requires minimum space for the tritium breeding as compared to other blanket concepts. This is very advantageous for application in the HELIAS FPP with the limited space available for a breeding blanket, as indicated in FIG. 2.

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<th>FIG. 1: HELIAS 5B model with vessel, coils, support structure and ports [2]</th>
<th>FIG. 2: Maximum and minimum spaces available for breeding and shielding in HELIAS 5B.</th>
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3.1 Design approach for the Tokamak DEMO

The design approach for the tokamak DEMO assumes a vacuum vessel (VV) with integrated shielding function [4], sufficient to protect the superconducting Toroidal Filed Coils (TFC) over the entire DEMO lifetime of 6 full power years (fpy). The VV is thus a lifetime component to which replaceable in-vessel components such as the breeder blankets are directly attached. The breeding blankets are designed for vertical maintenance and are separated into removable segments. A torus sector of 22.5° contains 3 outboard and 2 inboard segments. The Multi Module Segmentation (MMS) scheme employs 6 to 8 modules, arranged in poloidal direction on either side of the plasma chamber, see FIG. 3. The modules are attached to a Back Supporting Structure (BSS) acting as mechanical support and hosting the main manifolds for the coolant and the Tritium carrier. The radial space available for the breeder blanket, including first wall (FW), breeder zone (BZ), manifold (MF) and the BSS, is about 80 cm and 130 cm, inboard and outboard, respectively. The blanket needs to be designed for the maximum neutron wall loadings (NWL) which amount to 1.14 and 1.33 MW/m², inboard and outboard, respectively, for the current DEMO conditions.

3.2 HCPB breeder blanket concept

A blanket module of the MMS type consists of a steel box made of the Eurofer low-activation steel and includes the U-shaped FW, a stiffening grids (SG) with breeder units (BU), a box manifold with a back wall, two caps at the top and the bottom, and the integrated BBS, see FIG. 4 for a typical configuration. The dimensions of a module amount to 509 mm x
1184 mm x 1706 mm (radial x toroidal x poloidal) at the inboard mid-plane and 840 mm x 1576 mm x 2141 mm at the outboard mid-plane. Li4SiO4 ceramics is used as breeder material with 6Li enriched to 60 at% and beryllium as neutron multiplier. Both materials are filled in the form of pebble beds in the space between the cooling/stiffening plates. High pressure He gas is used for the cooling of the breeder units, the FW and the box structure.

3.3 Preliminary design approach for HELIAS

The design approach adopted in this work, is based on the utilization of the HCPB MMS breeder blanket design with the blanket segmentation and maintenance scheme proposed previously for HELIAS 5B [2]. HELIAS 5B employs a modular 5-field-period coil set which results in five identical torus sectors of 72 degree. The blanket of a torus sector is arranged in 16 toroidal rings with varying widths. Each blanket ring is divided into five segments with typical sizes of 5 m x 1.6 m x 0.8 m (height x width x depth). Thus there are 80 blanket segments in one torus sector which are to be exchanged through a vertical port. A preliminary CAD model, developed previously at IPP [2] (see FIG. 5), was adopted as basis for the development of specific simulation models for the neutronics and structural analyses.

4. Neutronics analyses: Tritium breeding and shielding performance

The neutronic analyses aim at the assessment of the breeding capability and the shielding performance of the blanket/shield system considered for HELIAS. A suitable neutron source model has been previously developed and verified for such analyses [5]. FIG. 6 shows a resulting 2D map of the neutron wall loading (NWL) produced for the HELIAS 5B. The NWL, averaged over the entire first wall surface, amounts to 0.95 MW/m² and thus compares favourably to the average value of 1.0 MW/m² for the tokamak DEMO. There are, however, large NWL variations in poloidal and toroidal directions. The maximum NWL is at 1.95 MW/m² and thus significantly higher than in the tokamak DEMO (1.35 MW/m²). This affects the thermal-hydraulic layout of the blanket modules which have to be designed for higher and lower neutron heating loads.

Different approaches were investigated for the modelling of the HELIAS 5B geometry in Monte Carlo (MC) transport simulations [6]. The analyses showed the superiority of the DAG-MC (Direct Accelerated Geometry Monte Carlo) approach [7], which enables the direct tracking of particles on a faceted CAD geometry. DAG-MC is available as extension to the MCNP-5 MC code [8] which is a standard code for neutronics simulations in fusion technology.
The DAG-MC model includes a simplified representation of the HCPB blanket with homogenized material layers. The radial build, adapted from the tokamak DEMO, assumes a FW protection layer (2 mm tungsten), a FW made of Eurofer steel (25 mm), a BZ of 50 cm (breeder blanket mixture with Li$_2$SiO$_4$, 60at% 6Li enrichment, Be multiplier, Eurofer structure and He coolant), a BSS of variable thickness (~10 cm - 40 cm), and a VV of 32 cm thickness. The VV consists of an inner and an outer SS-316 shell, each 6 cm thick, and a 20 cm thick shield mixture (60vol% SS-316, 40 vol% water) in between. The thickness of the BSS is adapted to the space available in each segment. In the DAG-MC calculations, typically $10^8$ to $10^9$ source neutron histories were tracked. A dedicated approach, based on the use of weight windows meshes for the variance reduction, was applied in the shielding calculations [9].

On the basis of the DAG-MC model with homogenized breeder blanket mixture, distributed uniformly in the large sized breeder zone, a very high TBR of $1.39 \pm 0.001$ was obtained. This high value is due to the very idealistic assumptions of a homogenized breeding zone which covers nearly the entire plasma chamber and does not take into account any gaps between the breeder blankets or the structural components of the breeder elements. The result indicates, however, that it should be possible to design a realistic breeder blanket for HELIAS-5B which can fulfil the tritium self-sufficiency requirement.

A typical neutron flux distribution, obtained with DAG-MC for the bean shape section of HELIAS, is shown in FIG. 8. The neutron flux in the plasma chamber of HELIAS and the surrounding is at the same level as in the tokamak DEMO. It is thus expected that the radiation shielding requirements for the superconducting field magnets can be fulfilled. Specific analyses are required, however, to check the shielding performance in those regions where minimum space is available and the NWL is high. Such a region, with a total thickness of about 100 cm and a NWL of ~1.4 MW/m$^2$ is indicated in FIG. 9 (yellow circle).

Radial profiles were calculated in this region for several nuclear responses [9], see FIGs. 10 a and b for the neutron flux and the power density. The comparison to the radiation load limits specified for the superconducting magnet coils of the tokamak DEMO [10] shows that the shielding requirements are not met in this region. Both the neutron flux and the power density exceed the specified limits by more than one order of magnitude. This is due to the limited space available in this region for the shielding. Possible countermeasures are to increase the thickness of the shielding zone at the expense of the breeder zone, and/or utilize more efficient shielding materials, such as tungsten combined with water.
The maximum displacement damage to the steel of the vacuum vessel was shown to be around ~ 0.11 dpa (displacements per atom) per full power year (fpy). This would allow to operate HELIAS 5B for 25 fpy before reaching the lifetime limit of 2.75 dpa as specified to guarantee the structural functionality of the vessel [10].

5. Blanket segmentation scheme - structural analysis

Most of the technologies, developed for the blankets of the tokamak DEMO, can be assumed to be valid for the HELIAS FPP assuming similar load conditions. This applies also for the HCPB blanket design. Of major concern, however, is the blanket segmentation scheme and the applicable maintenance strategy. Investigations were started in this field on the basis of the remote maintenance strategy proposed previously for HELIAS 5B [2]. The first aim is to find a segmentation which allows, under a simplified steady state loading scenario, the thermal expansion of the blanket modules in HELIAS without the risk of overlapping. A CAD model of a half period HELIAS-5B torus sector with dummy blanket modules (without internal structure) and 20 mm gaps between the modules was developed to this end, see FIG. 11. This model served as basis for the development of simplified 3D FEM models reproducing the triangular and bean shape sections of HELIAS-5B, i. e. blanket rings 1 and 8, respectively, see FIG. 12.
Spatially-averaged temperatures ($T_{\text{av}}$) of 445.9, 588.0 and 328.5 °C have been estimated for the FW, the BZ and the BSS, respectively, on the basis of the HCPB DEMO blanket data [11] and used as thermal boundary conditions. For the VV, a temperature of 200 °C has been assumed. Proper sets of equivalent material properties were used to represent the typical HCPB blanket material composition in the FEM model with dummy blanket segments, based on the material fractions of the HCPB DEMO blanket [12] for the calculated average temperatures. Thus proper account is taken of the masses for the structural materials (Eurofer for the BB, AISI 316L for the VV), breeders and coolants in the components of the models. Furthermore, an equivalent Young’s Modulus equal to 10 % of the actual one has been assumed for Eurofer and AISI 316L in the dummy components in order to ensure that their displacement is comparable with that of the real structure [13].

On the basis of these assumptions and models, structural analyses have been performed for steady state conditions considering a reference temperature of 20 °C and imposing a uniform temperature distribution in the different components. A uniform distribution has been also assumed for the gravity loads and the mechanical constraints. To reproduce the effect of the pendulum support, typically envisaged for a stellarator machine [14], the displacement along the global Z direction has been prevented to nodes on one line. Results of the calculated displacement distributions are shown in FIG. 13 (Ring1 sector) and 14 (Ring 8 sector). Ring 8 shows the highest displacement, with a considerable displacement towards the plasma of ~60 mm.
The residual gap size has been checked in order to verify that no overlapping occurs between segments. The results show that the considered 20 mm gaps are sufficient under the assumed loads, although some narrow residual gaps are predicted e. g. between segments 2-3 in Ring 8 (~ 0.2 mm) and segments 1-2 and 3-4 in Ring 1 (~3 to 5 mm). The maximum displacement towards the divertor was obtained in Ring 8 sector for Segment 4 and 5 (~ 20 to 30 mm). Thus an overlapping with the divertor can occur if a 20 mm gap is considered between the blanket segments 4, 5 and the divertor. The maximum displacements predicted for the toroidal direction towards adjacent rings are ~ 10 mm, i. e. half of the assumed gap width, for the Ring 1 segments. Ring 8 segments show lower maximum displacements around ~5 – 6 mm.

The analyses conducted so far indicate the viability of the proposed segmentation strategy. Although overlapping of some blanket segments cannot be excluded, the results are quite promising suggesting the extension of the analyses to more realistic simulations. This refers e. g. to neutronics results providing the nuclear heating distribution as required for the calculation of the real temperature distribution, taking into account the internal blanket structure as well. With such pre-conditions fulfilled, a full thermal-mechanical analysis can be performed to assess the nuclear performance of HELIAS 5B with breeder blankets.

6. Conclusions

Nuclear design issues of a stellarator power plant, based on the HELIAS concept, were presented and discussed in this paper. The approach adopted in this work is based on the utilization of the HCPB blanket, originally developed for the tokamak DEMO, for the HELIAS power plant. Specific analyses have been performed in the area of neutronics and structural mechanics utilizing ad-hoc developed simulation models of HELIAS 5B.
The neutronics analyses showed a favourable breeding performance suggesting that tritium self-sufficiency can be achieved in a HELIAS-type power reactor. The shielding performance suffers in those regions where minimum space is available for the breeding and shielding. It is concluded, however, that a sufficient shielding performance can be achieved by increasing the thickness of the shielding zone at the expense of the breeder zone, and/or utilize more efficient shielding materials. The structural analyses confirm the viability of the segmentation strategy proposed previously for HELIAS 5B. Overlapping of some blanket segments cannot be excluded, however, with the applied modelling and simulation approach. More realistic simulations are recommended on the basis of real 3D distributions of the nuclear heating and the temperature in the blanket segments which take into account the internal blanket structure as well. As result of the analyses performed, it is concluded that a HELIAS type power plant can be devised which satisfies the nuclear performance requirements. More detailed analyses are required, however, to proof this for an engineering design which takes into account all stellarator-specific constraints including the maintenance.

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References: