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THE ADVANCED STELLARATOR CONCEPT BEYOND W7-X: MOTIVATION AND OPTIONS FOR A BURNING PLASMA STELLARATOR

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From W7-X to a HELIAS fusion power plant: On Engineering Considerations for Next-Step Stellarator Devices

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Abstract

With the recent start of operation of Wendelstein 7-X (W7-X), the helical-axis advanced stellarator concept (HELIAS) has gained renewed attention. In particular a discussion has been started about a research strategy leading from W7-X to a commercial HELIAS fusion power plant. In order to bridge the respective gap in physics and technology between those devices, concepts for an intermediate-step burning plasma stellarator are under discussion. However, recent studies focused on the development of conceptual designs for such devices including mainly detailed physics considerations. Extending this discussion, in this work engineering and technology considerations for next-step HELIAS devices are discussed.

Keywords: HELIAS, W7-X, intermediate-step stellarator, DEMO, engineering, technology

1. Introduction

The European consensus for the realisation of commercial fusion energy includes the construction of a demonstration fusion power plant, often simply referred to as 'DEMO' [2]. Although DEMO will be a smaller scale prototype plant, it must prove that a workable solution exists for all physics and technology questions [3].

Following in the line of ITER, the current conceptual design of such a DEMO plant concentrates on the more advanced tokamak concept. However, with the recent start of operation of Wendelstein 7-X, the helical-axis advanced stellarator line (HELIAS) has gained new attention. Considering that W7-X will demonstrate the success of optimised stellarators, the future of the HELIAS concept must be discussed including the option for next-step HELIAS devices which are to follow W7-X. In particular the motivation and boundary scenarios for an intermediate-step stellarator which may bridge the gap from W7-X to a stellarator power plant have been discussed recently [4]. However, the latest advancements of the stellarator concept concentrated on physics issues while DEMO relevant technology development focused on tokamak aspects, thus leaving a gap in discussion regarding stellarator-specific engineering and technology considerations. Therefore, this paper attempts to start a more detailed discussion for engineering and technology aspects of next-step HELIAS devices. Issues and advances regarding the magnet system and blanket are discussed in section 2 and 3, respectively. A more general long-term outlook is given in section 4 where the ECRH system is taken as an example before the work is summarised in section 5.

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2. HELIAS Magnet System

2.1 Optimisation of Equilibrium and Modular Coils

The design and optimisation of a suitable magnetic configuration is one of the key research areas of advanced stellarators. Although stellarators exhibit a fully 3-D shaped, helical-structured plasma topology which leads to much higher 'neoclassical' transport than in axisymmetric devices, the intrinsic 3-D topology offers also positive aspects. In particular, the overall 3-D configurations space of possible stellarator designs is very large and in fact much larger than the configuration space spanned by axisymmetric devices which are limited to two dimensions. While initially the optimisation of stellarators was a necessary requisite in order to improve the confinement at high temperatures, it is now becoming an advantageous freedom as more and more aspects can be integrated in the optimisation process. Although this freedom is somewhat limited by engineering constraints it is nonetheless a powerful tool for resolving plasma related issues 'by design' rather then relying on favourable operation regimes.

Wendelstein 7-X is a prototype of such an advanced stellarator and included already a considerable number of optimisation criteria, which are listed below [5]:

- Good flux surfaces of the vacuum magnetic field
- Low Shafranov shift, good MHD stability and a stiff equilibrium up to a plasma beta of 4–5%
- Good energy and particle confinement, i.e., small neoclassical transport losses (drift optimisation, very low bootstrap current)
- A suitable divertor concept for controlled particle and energy exhaust (e.g., island divertor)
- Good confinement of fast particles (i.e., alpha particles)
- Feasibility of the modular coil set (i.e., low curvature)

Consequently, one of the high-level goals of W7-X is to assess, demonstrate and verify the success of stellarator optimisation.

Already in the first operation phase of W7-X the existence of good flux surfaces could be proven by flux surface measurements [6]. The successful construction and operation of the W7-X modular magnet system also implies the general feasibility of a non-planar coil system. But as W7-X started with a limiter configuration, the demonstration of the island divertor concept and the improved confinement at high performance as well as the investigation of high beta effects can only be approached within the next operation phases.

Despite these promising results, it has been recently realised by theoretical studies that the fast ion confinement in W7-X is restricted to a narrow region in the plasma centre [7]. In addition a volume averaged beta of around 3–4% is needed to realise the fast ion confinement in W7-X due the required diamagnetic effect.

Therefore, the optimisation of stellarator configurations has recently been more focused on improving the fast particle confinement in next-step HELIAS devices. While HELIAS-type configurations with improved fast particle confinement could be found, so far it remained a challenge to find a suitable accompanying modular coil set. Initial results required therefore an additional set of more complicated 'modulated toroidal coils' [8]. However, for next-step HELIAS devices and reactor-relevant configurations it is essential to keep the coil geometry as simple as possible, not only to save costs but also to reduce the forces acting on the coils.

In order to achieve this, the respective codes belonging to the stellarator optimisation framework ROSE and ONSET [9, 10, 11] have recently been upgraded. The major upgrades and achievements (first two points refer to ROSE and third point to ONSET) are the following:

- From now on the coil complexity and form can be directly considered during the optimisation of the magnetic configuration (plasma shape). Consequently, simpler and more suitable coil sets can be found. At this stage ROSE can include the complexity of either modular or saddle coils in the configuration optimisation.
- Also the vacuum and finite-beta equilibrium can be investigated at the same time during the optimisation procedure. This is important as a finite-beta has impact on the transport coefficients as well as through the diamagnetic effect on the fast particle confinement. In particular, now the finite-beta equilibrium can be optimised while at the same time evaluating the vacuum magnetic well.
- The optimisation scope of ONSET has been extended to include a free-boundary calculation of the equilibrium and an evaluation of the resulting VMEC solution along the lines of what is done by ROSE. Although this process is more time consuming, it can help to adjust sensitive parameters more accurately.

A preliminary coil set resulting from the upgraded version of the optimisation framework can be seen in Fig. 1. This tentative 5-field-period design has a higher number of coil-types (6 different non-planar coils compared to 5 in W7-X). The higher number of coils (in total 60 compared to 50 in W7-X) helps to considerably reduce the fast particle losses. It should be noted, however, that this coil set represents a preliminary result and further optimisation and analysis is ongoing.

What has not yet been considered in sufficient detail for the optimisation of magnetic configurations and their coil sets is the turbulent transport. In fact, turbulent transport was not considered at all during the design of W7-X. However, recent advances in gyrokinetic simulations have found that the magnetic topology has a profound effect on turbulent modes. E.g. in W7-X the trapped-electron-modes are partly stabilised [12] and the ion-temperature-gradient modes are localised on a flux-surface [13]. Following this realisation, initial attempts were made to control and reduce turbulent transport by appropriate shaping of 3-D magnetic configurations [14, 15]. Although these initial attempts are indicating that the turbulent transport can indeed be reduced by tailoring of the magnetic field, a more detailed understanding of turbulent transport in stellarators is necessary. Moreover, it needs to be checked in future studies to what degree turbulent optimisation is compatible with the existing optimisation criteria and coil design. If successful, turbulence reduction may become an important criteria in the integrated configuration and coil optimisation for advanced stellarators in the future.

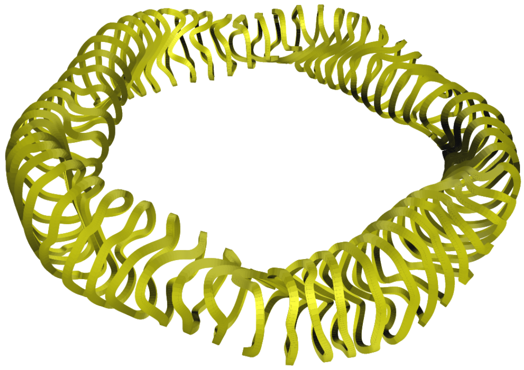

Figure 1: Tentative modular coil set of a 5-field-period HELIAS-type configuration with improved fast particle confinement and 6 different coil types totaling 60 coils.

2.2 Coil Technology

Early HELIAS power plant studies concentrated mainly on the NbTi low-temperature-superconductor (LTS) technology which was the state-of-the-art at the time. However, on these early designs the use of NbTi limited the magnetic field on-axis to about 4.5 T while often still requiring super-critical helium cooling at 1.8 K. However, with the rapid development of superconductor technology and in particular the maturing of Nb_3Sn as superconducting material, nowadays higher magnetic fields seem feasible. With the higher current density which can be used in Nb_3Sn, the field on-axis in a HELIAS device can be increased up to 6 T while employing 'normal' helium cooling at 4.2 K. This equals a magnetic field of about 12.5 T near and in between the coils. Although Nb_3Sn is more sensitive to strain than NbTi, Nb_3Sn technology is still compatible with the complex 3-D shape of the modular coils. In fact, the curvature of the coil excursions decreases with increasing coil size and is therefore smaller than in W7-X making it easier to wind the coils while reducing strain.

Due to the high aspect ratio of the HELIAS concept (large major radius R, small minor radius a, $A = R/a > 10$), the modular coils are small compared to the tokamak toroidal field coils. In fact, it has been found in the 'HELIAS 5-B' engineering study [16] (R = 22 m, A = 12.2), that the modular coils have about the same size as the toroidal field coils of ITER [17]. That means that even for a HELIAS power plant the modular coils can be directly built by industry and transported to the construction site by conventional means. This is more efficient and saves cost as large winding machines are not required on site.

Recently, also the high-temperature-superconductor (HTS) technology has made very rapid developments and may become an attractive alternative in the future [18]. The possible operation at higher temperature does not only save cryogenic power but also provides a higher coil stability due to the higher heat capacity of the material. The currently investigated HTS candidates are so-called YBCO materials (Yttrium barium copper oxides, a class of crystalline chemical compounds). Initially limited to flat and small 'tapes', recently conductor concepts have been developed and demonstrated based on stacked tapes [19]. These conductors could in principle be used to form complete coils, however more tests are required in the next years.

Although such a HTS technology would allow to employ higher magnetic fields, this is not yet foreseen for the HELIAS concept. This has several reasons. On the one hand, already at about 6 T the forces and stresses in a HELIAS power plant would be on the order of 650 MPa and a further increase of the magnetic field would bring the stresses close the material limits. On the other hand, a higher magnetic field increases the confinement, but at the same time reduces the plasma beta at fixed fusion power. However, as discussed in the last section a beta of around 4% is required in order to confine fast particles with the help of the diamagnetic effect. One could compensate for this by increasing the fusion power, but the increase in neutrons would at the same time reduce the lifetime of the components. Still, the higher current densities which could be achieved by HTS could be helpful especially in narrow regions on the the inboard side of a HELIAS. Thus, the coils size could be reduced or coil optimisation improved.

A specific issue of a HELIAS device is the non-uniform distribution of forces. Due to the complex 3-D shaping of the coils, the forces can point inward, outward as well as in toroidal direction. In the end, the detailed distribution of the forces depends on the shape of the coils and the coil set in total. In order to overcome this issue, bolted plates have been proposed as intercoil support-structure which in an initial study seemed to be promising to accommodate the corresponding forces and stresses [16].

3. Stellarator-specific Blanket

3.1 Blanket Technology

In stellarators, the space between the plasma and the coils is very narrow and imposes a limiting constraint on the design, see Fig. 2. Among the different blanket concepts, the Helium-Cooled-Pebble-Bed (HCPB) [20] is therefore one of the most favourable blanket concept for a HELIAS power plant due to the reduced breeding blanket thickness required to achieve tritium self-sufficiency in comparison to other blanket concepts. The HCPB has originally been developed for tokamak fusion reactors, but it is currently anticipated that the technology can be readily adapted to stellarators, although it needs to be proved by detailed studies in the future.

The breeding material is an Li_4SiO_4 ceramic enriched with ^6Li followed by beryllium pebbles, which serve as an efficient neutron multiplier providing a high tritium breeding ratio (TBR). The blanket is anticipated to be organised in segments with cross-sections of about 0.85 m^2. During maintenance, the segments can be exchanged individually and by remote handling through the large ports which are available in a HELIAS [16]. However, as the surface area of a high-aspect ratio stellarator is very large, the corresponding number of blanket segments is also very high (on the order of 300). This fact clearly poses a challenge in terms of remote replacement as the downtime of such a power plant should be kept as low as possible from an economic point of view.

The cooling system uses helium with an inlet temperature of about 300°C and an outlet temperature of 500°C at about 8 MPa pressure using EUROFER (a ferritic martensitic steel) as structural material. With a conventional Brayton or Rankine cycle, one can expect a thermal power conversion efficiency above 35% [21, 22]. An independent helium purge gas loop at low pressure (about 0.2 MPa) is used to extract the tritium from the ceramic breeder and from the beryllium.

Due to the high aspect ratio, the average neutron wall-load in a HELIAS is about half that of a tokamak of the same fusion power. This also means that the power density in the blanket is about a factor two lower. Without changing the blanket design, the flow velocity of the coolant could be reduced, which would reduce the associated pumping power quite strongly. This advantage is somewhat reduced by the fact that the total blanket volume can be up to a factor two higher in a HELIAS.

A final decision on the most suitable blanket concept for a HELIAS power plant can only be given after detailed numerical investigations (next section) and experimental tests as e.g. foreseen with the ITER Test Blanket Modules.

3.2 Neutronic Analysis

Following the arguments of the last section, a collaboration within IPP and KIT has recently been started for the numerical investigation and assessment of stellarator-specific aspects of the blanket design. The purpose of this activity is the development and preparation of a toolbox which allows a systematic study of 3-D neutronic simulations within HELIAS geometry. Ultimately this work shall lead to a detailed knowledge basis of the neutronics properties in next-step HELIAS devices and should consequently push the development of a stellarator-specific breeder blanket.

As the 3-D field structure of the HELIAS leads to a 3-D distribution of the neutrons, the consequences for the first-wall and blanket must be assessed in detail. The 3-D neutron wall load will especially require an adapted design and optimisation of the blanket design.

In the first step a 3-D neutron source has been established and adopted to the general Monte Carlo N-Particle Code (MCNP) [23, 24]. A verification of the stellarator-specific source subroutine was carried out and has shown good agreement of the normalised frequency of source points as sampled in MCNP and the emission probability as given by the plasma physics calculation [25].

However, due to the complex 3-D structure of the HELIAS geometry, a straightforward application of MCNP to stellarator geometry is not possible. In order to find an optimal solution

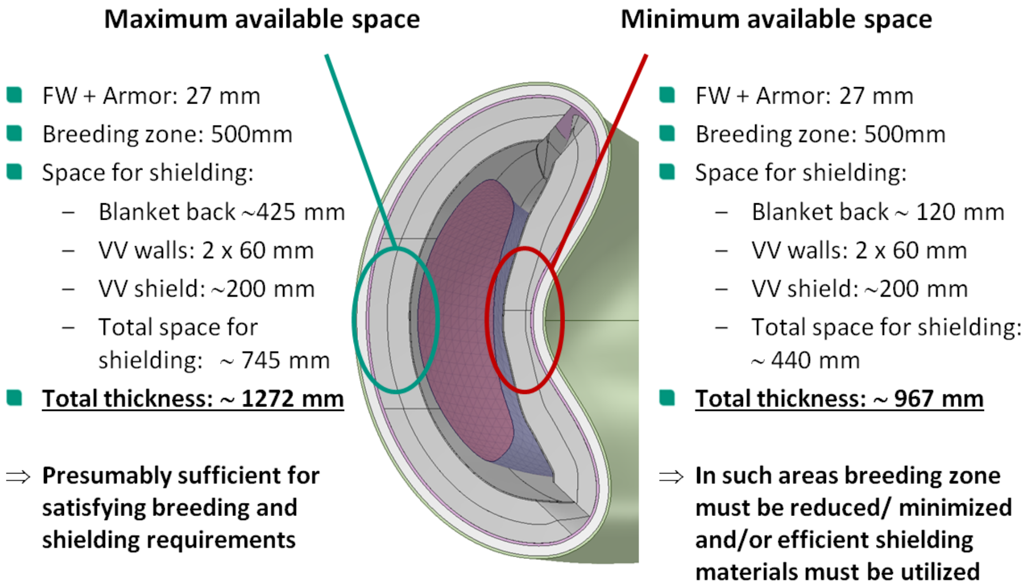

Figure 2: CAD model of a half field period of HELIAS 5-B showing maximum (outboard) and minimum (inboard) available space for breeding and shielding.

to treat the stellartor-specific geometry in MCNP, different approaches to generate a CAD based MCNP geometry have been tested [26]. The three used techniques are:

- the traditional geometry translation approach (CSG) using KIT's CAD to MCNP conversion tool McCad,
- employing faceted solids, i.e. direct tracking of particles in CAD geometry by direct usage of CAD geometry in MC codes with DAG-MCNP (direct accelerated geometry), and
- tracking of particles in an unstructured mesh geometry using newest version of MCNP6.

An exemplary test geometry with homogenous material layers has been used to test and verify all three simulation techniques. All three investigated methods give identical neutron flux profiles within the statistical uncertainty [26]. Consequently, the mentioned methods can be applied alternately in the upcoming neutron analysis of HELIAS power plant designs. In particular, the neutron wall loading, tritium breeding ratio, shielding performance, and nuclear heating of components will be investigated.

3.3 Geometry Modelling and Optimisation

A preliminary CAD model for the HELIAS 5-B reactor design was developed at IPP in order to investigate maintenance features and force distributions [16]. Due to this focus, the original CAD model was not well suited for neutronics calculations and needed substantial alterations [25].

The modifications which were done included geometry simplifications as required for neutronics purposes, the removal of geometry errors such as gaps and overlaps, the remodelling of surfaces which are described with spline functions which cannot be handled by all simulation methods, and the addition of void spaces which are not defined in CAD models but must be present in the Monte Carlo simulation model. Fig. 2 shows the current version of the CAD model as prepared for the MCNP simulations. It includes the plasma chamber, blanket modules with support structure and shield, the vacuum vessel and the field coils.

Further, Fig. 2 indicates the minimum and maximum spaces which are available in HELIAS-5B for the blanket modules and the shielding. A pre-assessment shows that in the regions of the maximum available space the breeding and shielding requirements presumably can be satisfied while in the regions of the minimum available space the breeding zone must be reduced and/or efficient shielding materials need to be utilised.

There are severe problems associated with the use of the HELIAS-5B geometry model in neutronics calculations using the MCNP code with the traditional geometry translation approach. A major problem arises from the fact that the original HELIAS-5B CAD model contains mainly spline surfaces which need to be converted to first or second order surfaces. This is a painstaking and error prone process which currently cannot be automated. This is one of the reasons why different methods were investigated to generate a CAD based MCNP geometry (see last subsection) and thus overcome the limitations posed by a geometry described by splines.

Consequently, such a manual approach is futile considering that the aim of this activity is to optimise the geometry which will inevitable lead to many design iterations. Therefore it is necessary to find a geometry description which is capable of representing the 3-D geometry accurately enough while maintaining a certain flexibility to allow for rapid design changes with minimum effort.

One solution which has been proposed is to use a regular, layered Finite Element (FE) mesh for the CAD model. For this approach ANSYS workbench has been chosen. Starting from a general spline representation it was possible to create thin FE element layers which are necessary to represent the 2 mm thick tungsten armor layer. A structural wall layer could then be added on top. The result is a FE mesh with no gaps and individual elements are represented by either linear or quadratic description. A preliminary result using a refined mesh is shown in Fig. 3.

The preliminary mesh model shown in Fig. 3 has some deviations from the original CAD model. Further improvements in the modelling and distribution of the individual elements are therefore necessary. It must also be checked to what degree further automation procedures can be implemented to speed up design iterations. In the next step it will be checked if such a mesh model can be readily integrated in MCNP simulations.

Another option which may be tested in the future is the creation of a SOLID model from the available mesh using ANSYS Finite Element Modeler. This could be an alternative approach to prepare MCNP suitable geometries.

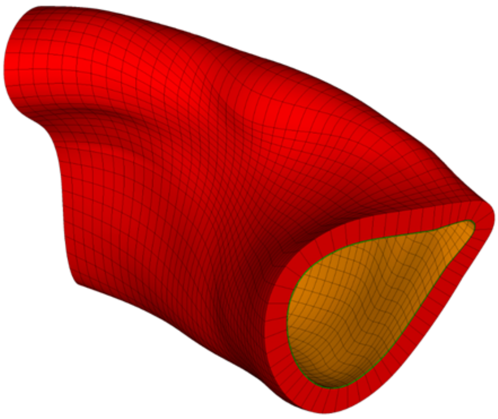

Figure 3: Mesh representation of a half period of the HELIAS 5-B blanket and first wall.

4. Relevant Technology for Next-Step HELIAS Devices

In order to proceed from today's experiments to next-step fusion devices also major steps in technology advancements must be made. The strong neutron environment and high power handling provide a technological challenge for many core components and materials no matter the device concept. Although the ITER device will contribute significantly to the integration of many technology challenges, it will not resolve these issues for a power generating DEMO-like fusion plant.

The considerations and outlook for the HELIAS-specific magnet system and superconductor technology as well as for the blanket design have already been discussed in the previous sections. Further technological challenges exist for the cooling system, remote handling, maintenance, heating, diagnostics, fuelling, tritium processing, divertor as well as the final dismantling of the device.

This list does not aspire to be complete but includes key components where considerable development is necessary to bring the technology to commercial maturity. However, many of these challenges will be addressed in the ITER and W7-X era within the next two decades. In particular will the development of certain technologies profit from the ITER and W7-X experience which is to be gained during the operation of those devices. An attempt to quantify this statement is made in Fig. 4.

Using the 'technology readiness level' (TRL) approach [27], existing technologies from today's experiments are sorted in their respective TRL category showing their level of maturity. The clear classification of a technology within TRL is by no means trivial. In particular, the assessment depends on the detailed definition of the TRL levels for each technology level. Depending then on the characteristics as well as the advantages and disadvantages of a prototype the classification within TRL may vary. The sorting made in Fig. 4 is therefore by no means a precise classification, but rather an abstract sketch. However, even for such a rough assessment the conclusions which can be drawn from the TRL classification remain intact. A more precise analysis for the tokamak concept was done in [28] and for the heliotron line in [29].

Here, TRL 1 is the lowest level representing the start of scientific research for a certain technology. An increase in TRL corresponds to the advancement of research and the technological development. Important here is in particular the step from TRL 6 to TRL 7. The experience gained from existing devices and the integration of this experience into an enhanced design can bring a technology only up to TRL 6. TRL 7 is a major step up from TRL 6 and represents the construction and test of an actual system prototype. In that sense, TRL 7 requires the construction of a next-step device which in tokamak community if often refereed to as DEMO and in the stellarator community as intermediate-step stellarator.

What we can be learned from Fig. 4 is that considerable technology development is required to bring the system components of a HELIAS power plant to maturity. While the experience from ITER and W7-X helps in certain aspects, many components require special development outside of those devices to prepare for the next-step devices. It is thus inevitable to start engineering activities for the HELIAS line.

While many of the technology challenges are similar for both tokamak and stellarator providing synergy effects during research and development, the particular details are often very specific. An example is the electron cyclotron heating system (ECRH) which is a general heating scheme in fusion devices but the specific application in tokamaks and stellarators is somewhat different. Therefore, the ECRH system for next-step HELIAS devices is exemplary discussed in the next subsection.

It should be noted that an TRL assessment gives no information about the expected time line. While some technologies may develop very rapidly through innovations others may be slow due to encountered problems or even potential showstoppers in the worst case.

What has not been included in the discussion here is the issue of the development and qualification of radiation resistant materials with a long lifetime up to many dpa. This is an extensive topic on its own, beyond the scope of this work. It is sufficient to state that material research requires a high priority and the qualification of materials is essential in obtaining an operation license for fusion devices including an adequate concept for dismantling and radiative waste.

4.1 Electron Cyclotron Heating System

Electron Cyclotron Resonance Heating (ECRH) can be used in fusion devices for plasma start-up, bulk plasma heating as well as current drive. Except for JET, nearly every major experimental plasma device is equipped with an ECRH system,

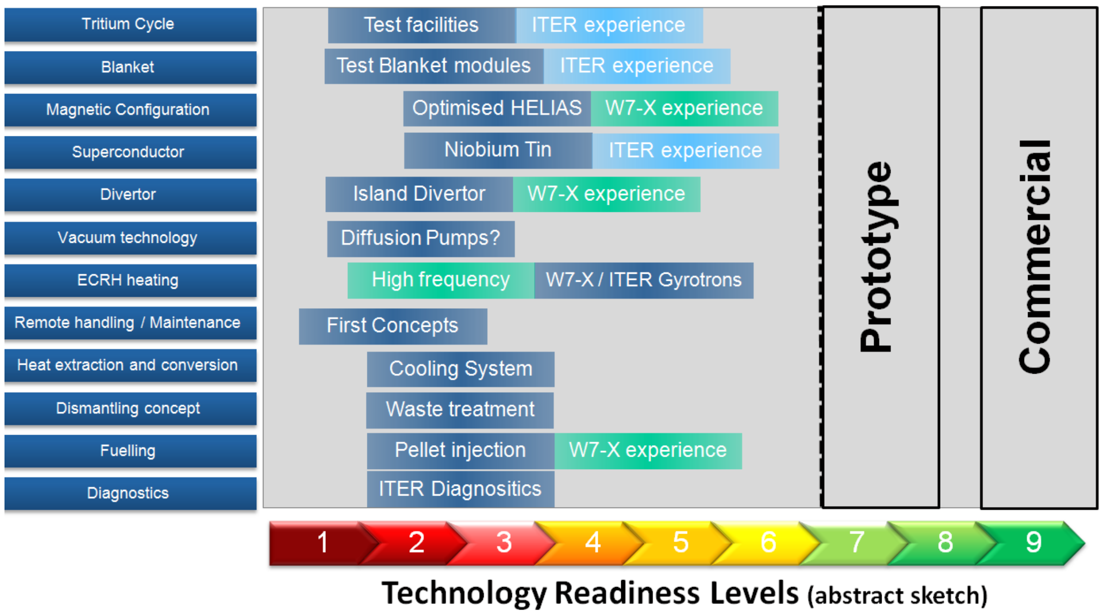

Figure 4: Abstract assessment of technology readiness levels for different technological challenges of the HELIAS-line and the expected advancement in these technologies from ITER and W7-X operational experience. Not accounting for material issues.

thus a number of different gyrotron and transmission technologies are developed since the last four decades. The most relevant gyrotrons for next-step fusion devices are the long pulse (1800 s), high power (1 MW) W7-X gyrotrons (140 GHz) and the respective gyrotrons for ITER (3600 s, 1 MW, 170 GHz). Gyrotrons for DEMO with even higher frequencies ($\sim$240 GHz) are already under investigation allowing to go to higher field and achieving optimum current drive efficiency [30].

Beyond the investigation of fixed frequency gyrotrons recently also effort was made to demonstrate the principle of a step-tunable frequency gyrotron. Such a multi-purpose gyrotron allows operation at different magnetic fields and at different optimum frequencies for heating and current drive.

For the transmission of the power from the gyrotrons to the plasma, different technologies are available. On the one hand evacuated waveguides are a traditional solution which is also foreseen for ITER, on the other hand quasi optical transmission, as employed in W7-X, may also be a promising option for a next-step DEMO-tokamak or HELIAS.

Regarding the harsh environment in a power plant like fusion devices, the currently used front steering launcher technique is no option. Alternative concepts like a remote steering launcher without moveable parts in the reactor chamber are therefore of great interest and will be tested in the future campaigns of W7-X [31].

Although all of these technologies can be employed for both tokamak and stellarator, the details of application are different. For the tokamak concept, the current drive capabilities of ECRH are of bigger interest. Moreover, high-precision ECRH is required in tokamaks to control neoclassical tearing modes (NTM) and sawtooth instabilities. For the stellarator, ECRH is mostly needed for start-up and heating with the option of controlling the bootstrap current with counter current drive. It follows from these different application goals that the specific implementation of ECRH technology is different in tokamaks and stellarators although the basic technology is the same.

Regarding next-step HELIAS devices, an EUROfusion research project at IPP has been started for the comparison of the different ECRH technologies with regard to safety and the industrial standard ‘RAMI’, i.e. reliability, availability, maintainability and inspectability. Considering the physical and technical advantages and disadvantages of each subsystem, this study shall lead to a conceptual proposal for an ECRH system for next-step HELIAS devices with considerable fusion power.

5. Summary and Conclusions

Continuing the discussion of an intermediate-step stellarator which may follow W7-X, some engineering aspects of such a next-step device were discussed in this work. In particular the optimisaiton of the magnetic configuration is an integral aspect for the design of a next-step stellarator allowing to solve plasma issues ‘by design’. The recent progress regarding the fast particle confinement confirms the potential of this method, but further aspects such as turbulence need consideration.

A next-step stellarator will definitely feature a neutron environment and it is therefore essential to progress with a detailed neutron analysis of a HELIAS in an efficient way foreseeing design iterations in the future. This activity has been recently started and a conceptual CAD design prepared. Due to the difficulties which arise from the 3-D geometry, different geometry representations were tested in MCNP which agree within statistical uncertainty. Further, a mesh model representation is currently tested which may be an option for fast and automatic geometry iterations saving engineering resources in the future.

Finally, gaps in technology development towards a HELIAS power plant have been discussed using the technology readiness method. While some specific technologies are quite advanced like W7-X type gyrotrons and ITER niobium tin superconductor, considerable efforts are required for the advancement of other components. While many synergy effects arise from the development of the same technology as the tokamak line, stellarator-specific aspects must be taken into account. A greater engineering effort is therefore inevitable to prepare the HELIAS line for its next step.

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