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Helias Reactor Studies

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Abstract

The Helias reactor is an upgraded version of the Wendelstein 7-X experiment. The magnetic field has 5 field periods and the main optimization principle is the reduction of the Pfirsch-Schlueter currents and the Shafranov shift, which has been verified by computations with the NEMEC and MFBE-codes. The modular coil system comprises 50 coils, which are constructed using NbTi-super-conducting cables. The basic dimensions are: major radius 22 m, average plasma radius 1.8 m, magnetic field on axis 5 T, maximum field on the coils 10 T.

Forces and stresses in the coil system have been investigated with the aid of the ANSYS code, which found maximum stress values of about 650 MPa in the coil casing. Helias configurations with 4 and 3 field periods have been constructed by starting from the 5-period case and by eliminating one or two periods while the shape of the coils is kept nearly invariant. In a first survey several blanket concepts, developed for the DEMO tokamak, have been adapted to the Helias geometry, in particular, the solid breeder concept developed by FZK (Karlsruhe) has been extrapolated to the Helias geometry identifying the drawbacks and advantages of this concept. Furthermore, the liquid breeder concept using Li17-Pb83 and water-cooling is an interesting alternative for the Helias reactor. Maintenance of blanket and plasma facing components is possible through the portholes between modular coils.

Numerical simulations of the start-up phase of the Helias reactor using the TOTAL-P code have confirmed the zero-dimensional modeling of the fusion plasma with the aid of empirical scaling laws.

1. Introduction

The dimensions of a Helias reactor are determined by the following requirements:

- The magnetic configuration is an optimized Helias configuration
- There must be sufficient space for blanket and shield
- The magnetic field is small enough to allow for NbTi-super-conducting coils
- Plasma confinement must be sufficiently good to provide ignition.
- Alpha-particle confinement should be sufficiently good
- Thermal fusion power should be about 3000 MW.

These requirements have led to the following parameters of the Helias reactor

Table 1:

Major radius [m]	22
Minor radius [m]	1.8
Plasma volume [m^3]	1407
Iota(0)	0.84
Iota(a)	1.00
Magnetic field on axis [T]	5
Max. field on coils [T]	10

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Number of coils	50
Magnetic energy [GJ]	100
Av. Line density [10^{20} m^{-3}]	2.12
Max. temperature [keV]	15
Max. beta [%]	15.7
Av. beta [%]	4.24

In the following, results of Helias reactor studies will be discussed. These encompass computations of the plasma equilibrium. MHD-stability of these equilibria has been studied using the ballooning mode code JMC and the global-mode code CAS3D. Further physics studies have been concentrated on alpha-particle confinement, neoclassical transport of the thermal plasma and modeling of the start-up phase and the steady-state phase of the fusion plasma using the TOTAL_P code [1]. Engineering studies have been focussed on the coil system and the optimization of the coil shape in order to reduce the maximum magnetic field in the coils. A further activity has been devoted to a selection of blanket concepts.

2. Physics Studies

The vacuum field of the Helias reactor differs from the standard field of Wendelstein 7-X by a higher toroidal mirror, which is necessary to improve the confinement of the alpha particles. The rotational transform in the plasma center decreases with increasing plasma pressure and low-order rational surfaces, which in the vacuum field do not exist, show up in the finite beta case. In order to avoid the existence of the 5/6 resonance the shear of the vacuum magnetic field has been reduced as much as possible thus avoiding the 5/6 resonance up to a beta value of 4.2%.

A series of equilibria has been computed using the NEMEC code and the MFBE code [2]. The MFBE code allows one to compute the magnetic field outside the last magnetic surface, which is the indispensible basis of the divertor design. The magnetic surfaces of the vacuum field are shifted slightly to the inside by an appropriate vertical field, which has been built into the modular coils. The reason is the requirement to center the plasma column in the finite beta state at $\langle\beta\rangle$ = 4–5% with respect to the first wall. This would lead to equal neutron wall loading at the inboard and the outboard side. Although the plasma equilibrium in the Helias configuration has been optimized, the residual Shafranov shift can grow to 0.5 m.

Numerical investigations of the MHD-stability have shown stability up to an averaged beta of 4%. This result was found by local ballooning mode analysis using the JMC-code and by global mode analysis with the CAS3D code [3]. Both methods yield instability at $\langle\beta\rangle$ = 5%. Interpolating the results of CAS3D results in a stability limit of 4.2% [4].

Low-order rational magnetic surfaces, which do not exist in the vacuum field, can exist in the finite beta plasma. This concerns the 5/6 resonance, which appears for $\langle\beta\rangle \geq 4.2\%$. Since at this point a global mode is expected, the appearance of $\iota(0) = 5/6$ is used as a definition of the MHD-stability limit. The magnetic well deepens with rising beta, which is a decisive feature in providing MHD-stability.

Drift waves in linear and non-linear approximation have also been studied taking into account the specific geometry of the Helias configuration. In particular, attention has been focussed on the effect of field line curvature and local shear on the linear growth rate of the dissipative drift waves. There is a significant difference between the Wendelstein 7-AS configuration and Wendelstein 7-X, which can be understood mainly by the difference in local shear. The larger local shear in Wendelstein 7-X, which is a by-product of the optimization process, has a stabilizing effect on the drift waves [5].

Computations of neoclassical transport in non-axisymmetric configurations using a Monte-Carlo code are lengthy and time-consuming. The reason is the complex Fourier spectrum of the magnetic field and the trapping of particles in the various magnetic mirrors of the Helias configuration. For reactor-relevant parameters the ripple-averaged kinetic equation is an appropriate tool to compute these losses. Analytical and numerical approaches to solve this equation [6] allow one to compute neoclassical losses on a much faster time scale than any other methods. The effective helical ripple in the Helias reactor depends on the plasma pressure and is in the range of 1.5–2%.

Good confinement of highly energetic alpha particles is a necessary condition for self-sustaining operation of the fusion process in a Helias reactor. In this context the following problems are of importance: Sufficient confinement of trapped alpha particles, a small number of particles trapped in the modular ripple and anomalous losses of alpha particles by plasma oscillations

Previous investigations (A.V. Zolotukhin [7]) have shown that the losses of alpha particles trapped in a modular ripple can be sufficiently reduced in case of ten

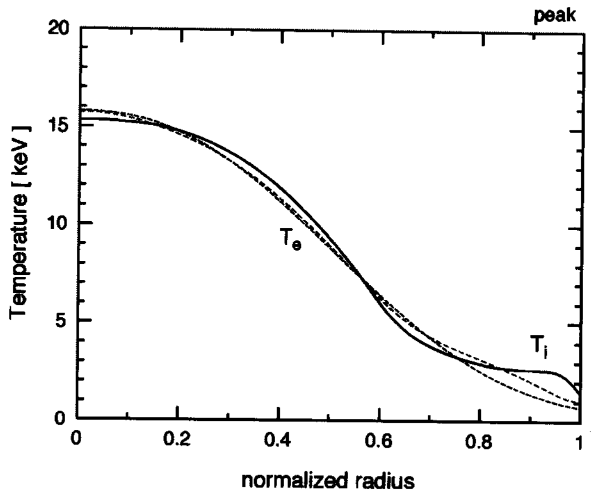

Fig. 1 Temperature profiles in HSR, result of TOTAL_P.

modular coils per period. A smaller number of coils would raise the ripple-induced losses appreciably. The majority of trapped alpha particles are trapped in the basic field period. As already shown by W. Lotz *et al.* [8], poloidal magnetic drift in the finite beta equilibrium improves the confinement of these particles such that only a small fraction of trapped particle is lost in a time shorter than one slowing down time. In the present reactor configuration, however, more than 10% of the heating power is lost by poorly confined alpha particles. Thus, further fine-tuning of the magnetic field is necessary to improve the classical confinement of alpha particles.

Fusion plasmas in the Helias reactor have been modeled using the transport code TOTAL_P, which has been developed by T. Amano and K. Yamazaki. In this code all neoclassical transport coefficients including the non-diagonal terms have been retained and, in addition, the radial electric field has been determined by the condition of ambipolarity. The code solves three coupled differential equations of electron temperature, ion temperature and density and computes the time evolution of the profiles. As one of the first results hollow density profiles have been computed, which occur at low density in conjunction with pellet refueling. Hollow density profiles are not very favorable with respect to fusion power output; possibilities to prevent these profiles need to be studied further.

3. Engineering Studies

The coil system of the Helias reactor consists of 50 modular coils but with only 5 different types. There are 10 coils in every field period and because of the stellarator symmetry only 5 coils have different shapes. Accessibility to the blanket would be facilitated by a smaller number of coils, but this would raise the modular ripple and the losses of highly energetic alpha particles. Concerning the inner structure of the coils large efforts have been made to shape the winding pack in order to reduce the maximum magnetic field in the coils. By shaping the winding pack into a trapezoidal form the maximum magnetic field on the coils could be reduced to 10 T, while the averaged magnetic field on axis is close to 5 T. There is a sufficient safety margin to the limit of a NbTi-superconductor, if forced flow cooling with super-critical Helium at 1.8 K is applied [9]. In contrast to a former design [10] the windings are wound in double pancakes consisting of 2×18 turns each. The winding process takes place on a winding mould, which also serves as a stiffening element of the coils. The dimensions of the super-conducting cable are 32×32 mm^2. 8 of these elements together with the windings are welded together. In contrast to the former concept where the winding pack is wound in layers, the advantage of the pancake technique is that the winding mould is an integrated part of the coils. The 8 elements are welded together and then enclosed by a coil casing.

Stresses in the coils depend strongly on the geometry of the support system. This support system consists of the coil casing and the intercoil support elements. In designing this support system a compromise could be reached between the need to minimize the stresses and the desire for optimum access to blanket and the plasma chamber. Stress analysis is performed using the ANSYS code. Orthotropic elastic data of the proposed cable in conduit conductor are used for these computations as approximation to the complex winding pack, which consists of super-conducting strands, copper, aluminum alloy and insulating material. The elastic data of the coil casing and the support elements are those of stainless steel, the elastic data of the winding pack are those of Wendelstein 7-X.

The maximum stress found in the coil housing is 650 MPa. Locally the deformation is larger than 0.2%, which implies that further optimization of the support system is needed to reduce the local stress maxima. First results of the winding pack analysis show that the stresses in the super-conducting windings are not higher than 40 MPa.

The dimensions of the Helias reactor are mainly determined by the need to accommodate a breeding blanket and a shield. In the narrowest region on the inboard side the distance between plasma surface and coil casing is 1.5 m, which leaves a gap of 1.3 m for blanket and shield. On other places around the torus the gap is wider, it can reach nearly 2 m. In the frame of the NET-ITER activity several blanket concepts have been developed and one of the goals in Helias reactor studies was to transfer these concepts to the Helias reactor and to establish a critical assessment of the advantages and disadvantages of the various concepts [11].

The options in the Helias reactor are:

- Helium-cooled solid breeder blanket (HCPB)
- Water-cooled Li-Pb blanket (WCLL)

Two major differences between a tokamak reactor and a Helias reactor are decisive with respect to blanket design and performance; these are the three-dimensional shape of the blanket and the large area of the first wall. In the present concept this area is 2600 m^2 which leads to an averaged neutron wall loading of less than 1 MW/m^2 (fusion power 3000 MW). The peak wall loading is 1.7 MW/m^2. As the result the lifetime of first wall and plasma facing components is larger than in a compact tokamak reactor. Compared with the DEMO tokamak reactor, where a lifetime of 2.3 years [12] (70 dpa in the structural material) is envisaged, the lifetime of plasma facing components and blanket elements can reach 4.6 years. Since in present material studies a limit of 140 dpa is considered as realistic, the lifetime of components in the Helias reactor may reach 9 years.

In contrast to earlier concepts of maintenance and repair, in which a whole period of the coil system is horizontally withdrawn, it is now proposed to replace blanket segments through portholes between the coils. In every period there are 8 big portholes available, 4 on the top and 4 at the bottom of the period. Typical dimensions of the portholes are 2×6 m^2. Also in the horizontal direction large portholes are available. The number of segments in HSR22 is 250; in one field period there are 50 blanket segments. Because of the symmetry of the stellarator configuration there are only 25 segments with different shapes. The size of one segment is roughly $1 \times 1 \times 10$ m^3 and it has been studied how these blanket segments can be installed through the portholes. The two blanket concepts discussed above differ with respect to their maintenance procedure. While the segments of the solid breeder blanket must be installed and removed fully equipped with breeder material and neutron multiplier, the segments of the LiPb-concepts can be replaced without the liquid breeder material.

Parameter studies have begun to explore alternatives to the 5-period Helias configuration [13]. The goal is to reduce the aspect ratio and to achieve a more compact device. Alternatives are 4 periods and 3 periods, which can be modeled by starting from the 5-period HSR22 (R = 22 m) and by removing one or two periods while the space for blanket and shield is kept fixed. One disadvantage of the low aspect ratio configurations is a larger ratio of the magnetic field on the coils to the average field on the magnetic axis, which leads to an increase of the electromagnetic forces on the coils. Since in NbTi superconductors the field strength on the coils is a limiting value, a reduction of the field strength in the plasma region is unavoidable. If this for reasons of confinement is not tolerable, the use of NbSn superconductors must be considered. It is also expected that the stability limit shrinks as one lowers the number of field periods.

4. Conclusions

Significant progress has been achieved in understanding the physics issues of a Helias fusion reactor. There is a chance to run the Helias reactor in stable operation at $\langle\beta\rangle = 4.2\%$ and to deliver a fusion power output of 3000 MW. Projections of present-day empirical scaling laws are not in contradiction to the requirement of a self-sustained fusion discharge; the results of the LHD experiment justify some optimism that these scaling laws can be extrapolated to the reactor regime. Dilution of the fusion plasma by thermal alpha particles is still an issue. It is closely related to the neoclassical accumulation of impurity ions in the plasma center. However, it is expected that the fusion plasma stays at the beta limit, where the onset of MHD-instability leads to increased plasma losses and thus stabilizes the thermal instability.

The divertor concept in the Helias reactor relies upon the efficacy of the island divertor. This island divertor will be installed and tested in the Wendelstein 7-X device and it is designed for a power load of 10 MW/m^2. If this is achievable under reactor conditions a wetted area of about 40 m^2 would be needed in HSR.

Because of the large area of the first wall in a Helias reactor, the initial installation of the breeding blanket is nearly two times larger than in the equivalent tokamak. However, since the neutron wall load is reduced, the lifetime of blanket components is roughly twice as long. Therefore, integrated over the lifetime of the fusion reactor (30 years), the amount of breeder material is nearly the same. This is also true for the amount of nuclear waste at shutdown of the power plant.

First steps are being made to develop an integrated concept of the stellarator power plant including the buildings and the thermal conversion cycle.

References

[1] T. Amano *et al.*, IPP-report III/246, Jan. 2000.

[2] E. Strumberger, C. Nührenberg *et al.*, *Proc. 26th EPS-Conf.* Maastricht, June 1999.

[3] C. Nührenberg, Phys. Fluids **B5** (1993) 3195.

[4] E. Strumberger *et al.*, IPP-report IPP III/249.

[5] A. Kendl and H. Wobig, Phys. Plasmas **6**, (1999), 4714.

[6] C.D. Beidler and H. Maassberg, IAEA-Technical Committee Meeting, Seeon, June 1999.

[7] A.V. Zolotukhin *et al.*, Verhandl. DPG (VI) 32, 275, P13.12 (1997).

[8] Lotz W., P. Merkel, J. Nührenberg and E. Strumberger, Plasma Physics and Contr. Fusion **34**, 1037-1052 (1992).

[9] C.D. Beidler *et al.*, *Proc 16th IAEA Conf. on Fusion Energy*, Montreal 1996, paper IAEA-Cn-64/G1-4.

[10] E. Harmeyer, N. Jaksic and J. Simon Weidner, *Proc. 19th SOFT*, Lisboa 1996, Vol. II, p. 1035.

[11] H. Wobig, E. Harmeyer, F. Herrnegger and J. Kisslinger, IPP-report III/244.

[12] M. Dalle-Donne *et al.*, KfK Report KfK 5429, Nov. 1994.

[13] J. Kisslinger, C.D. Beidler, E. Strumberger and H. Wobig, Int. Stellarator workshop, Madison 1999.