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RECENT DEVELOPMENTS IN HELIAS REACTOR STUDIES

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Abstract: The Helias ignition experiment is an upgraded version of the Wendelstein 7-X experiment. The magnetic configuration is a 4-period Helias configuration (major radius 18 m, plasma radius 2.0 m, B = 4.5 T), which presents a more compact option than the five period configuration. Main effort has been focussed on two versions of the 4-period configuration: one option is the power reactor HSR4/18 providing at least 3 GW of fusion power and the second option is the ignition experiment HSR 4/18i aiming at a minimum of fusion power and the demonstration of self-sustaining burn. The design criteria of the ignition experiment HSR 4/18i are the following: The experiment should demonstrate: a safe and reliable route to ignition, self-sustained burn without external heating, steady-state operation during several hundred seconds, reliability of the technical components and tritium breeding in a test blanket. The paper discusses the technical issues of the coil system and describes the vacuum vessel and the shielding blanket. The power balance will be modelled with given profiles and the ignition conditions will be investigated using current scaling laws of energy confinement in stellarators. The plasma parameters of the ignition experiment are: peak density $2\text{-}3\cdot10^{20}$ m^{-3}, peak temperature 11-15 keV, average beta 3.6%, fusion power 1500 – 1700 MW.

2) Introduction

The Helias reactor (HSR) is an upgraded version of the Wendelstein 7-X experiment [1], which is under construction in the city of Greifswald, Germany. Recent studies have been focused on a 4-period Helias configuration (HSR 4/18: major radius 18 m, plasma radius 2.1 m, B = 4.4 T), which is a more compact option for a Helias ignition experiment (HSR4/18i) than the five period configuration HSR 5/22 [2]. HSR4/18 [3] and HSR5/22 [4] are designed as power reactors producing 3000 MW of fusion power. For this reason the magnetic field is 5 T and in the range 10–10.3 T on the coils. A stellarator ignition experiment is the first step into the nuclear phase of stellarator research as a competative fusion power plant. It is an experiment, which – after successful operation of Wendelstein 7-X and other stellarators – is the natural next step in the stellarator line. Since many features of plasma behaviour are common to stellarators and tokamaks, use can be made from many results and achievement of tokamak research. However, the stellarator geometry introduces new features, which cannot be extrapolated from tokamak experiments. In particular, the absence of a toroidal current leads to the absence of disruptive instabilities and the associated methods of control and disruption mitigation. Since in stellarators there is no need to drive a toroidal current, neither transformer coils nor poloidal field coils are necessary.
The ignition experiment should

- Achieve extended burn without external heating and current drive and with a duration being sufficient to reach stationary condition.
- Demonstrate the reliable performance of the refueling processes and the control of the

operational point.
- Demonstrate sufficient and reliable divertor action and control of plasma impurities.
- Demonstrate the feasibility and availability of a large modular magnet system, which is essential for a stellarator power reactor.
- Demonstrate the capability to replace damaged components (divertor elements) by remote control

Basically the objectives of the stellarator ignition experiment are those of the ITER-FDR concept, and unlike to ITER FEAT, however, self-sustained burn without external heating ($Q \to \infty$) and without current drive is envisaged. It turns out that the fusion power in the stellarator ignition experiment, as will be described later, is nearly the same as in ITER FDR, namely 1500 MW. This implies the same neutron production rate and thus many results, which have been elaborated during the ITER design phase, can be taken over in designing the stellarator ignition experiment. This concerns the performance of the shield and the activation of the structural material. The principles of safety are the same in both devices.

3) Design criteria

The size of the ignition experiment is determined by two requirements: it should by large enough to accommodate a shielding blanket and secondly it should be large enough to meet the ignition condition $<n>\tau_E >2\cdot 10^{20}$. The confinement time will be extrapolated on the basis of present stellarator experiments. For the reason of keeping costs and mechanical stresses of the coil system as low as possible the magnetic field on the coils is kept below 9 T, which allows one to use NbTi-superconductors. As has been shown at the LCT-coil the magnetic field in this NbTi-coil can be pushed to 11 T, however He-cooling at 1.8 K is needed. In order to allow for a more conventional cooling at 4.2 K the magnetic field in the Helias ignition experiment has been reduced to 8.5 T. This leads to an average field of 4.4 T on the magnetic axis. The magnetic field configuration of the Helias ignition experiment is basically the same as in the power reactor HSR4/18 [3]. However, since a tritium-breeding blanket is not foreseen, the coils can be made smaller and arranged closer to the plasma than in HSR4/18. By this method the magnetic field can be reduced to 8.5 T on the coils.

TABLE 1: MAIN DATA OF HELIAS CONFIGURATIONS HSR4/18i, HSR4/18 and HSR5/22

	HSR 4/18i	HSR4/18	HSR 5/22	
Major radius	18	18	22	[m]
Average minor radius	2.1	2.1	1.8	[m]
Plasma volume	1524	1524	1407	[m^3]
Iota(0)	0.86	0.83	0.84	
Iota(a)	0.97	0.96	1.00	
Average field on axis	4.4	5	4.75	[T]
Maximum field on co ils	8.5	10.3	10.0	[T]
Number of field periods	4	4	5	
Number of coils	40	40	50	
Magnetic energy	76	98	100	[GJ]

Reducing the magnetic field to the lowest possible value alleviates the engineering problems associated with forces and stresses; however, the lower limit is set by the need for sufficient plasma confinement and MHD-stability.

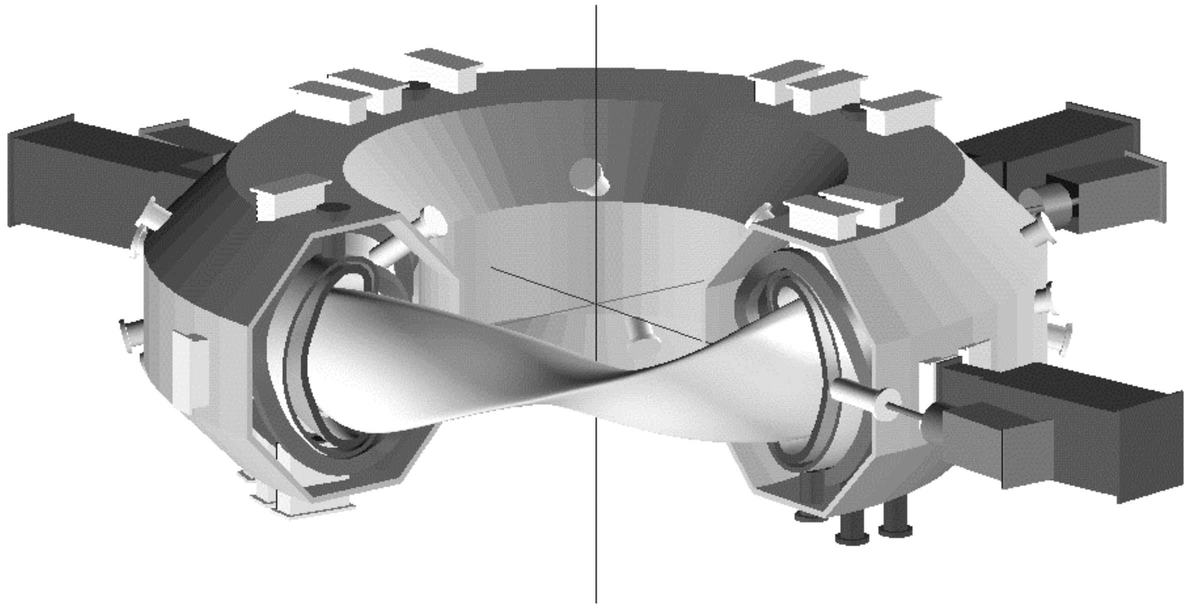

FIG. 1: Magnetic surface, modular coils and cryostat of the 4-period Helias reactor HSR4/18i

2. Coil system

The modular coil system of HSR4/18i comprises 10 coils per field period, which are constructed using NbTi-superconducting cables. Efforts have been made to reduce the maximum magnetic field in the coils to B = 8.5 T by shaping the winding pack trapezoidally.

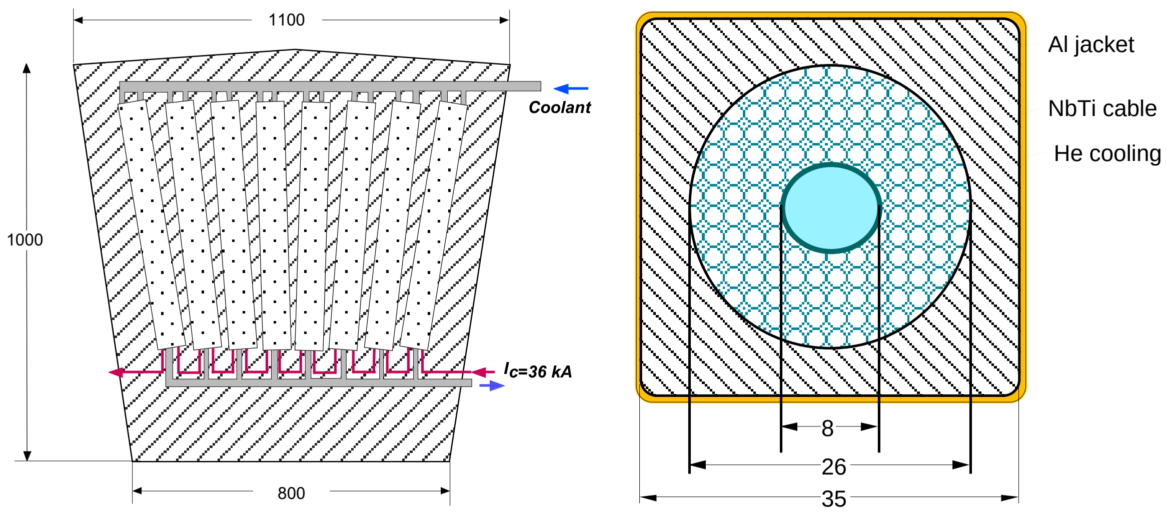

FIG. 2: Left: Cross section of coil, right: NbTi-cable

The superconducting cable is designed following the experience of Wendelstein-7X. However, the linear dimensions are two times larger and in contrast to the cable of W-7X a central cooling tube exists. The winding pack consists of 8 double pancakes with the cooling inlet on the plasma side of the coils, where the nuclear heating has a maximum. The number of turns in a layer is 288.

The ANSYS-code is the main tool to perform structural analysis of the coils and in the support system. For this purpose a support system has been designed, where the coils are connected to each other forming a support ring. Sufficient space is left between the coils to give access for maintenance. Because of the large virial stresses the coils are equipped with a steel casing and stiffening ribs between the pancake-type winding layers. Taking into account the symmetry properties of the magnetic field the stress analysis can be restricted to 5 coils in one period. In the power reactor HSR4/18 [3] the highest mechanical stresses in the steel casing are about 750 MPa. Inside the winding pack the non-linear behaviour of the imbedding has been taken into account leading to a maximum compression stress of 70 MPa in the windings. Because of the reduction of the magnetic field in HSR4/18i (from 10 T to 8.5 T) the mechanical stresses of the ignition experiment are smaller. In this case one could also reduce the support system of the coils and go to the maximum stresses. Furthermore, a coil protection system has been investigated, which in the case of quench leads to a rapid switch-off of the magnetic field and the transfer of the magnetic energy into 20 dump resistors. Eight identical coils are connected in series and fed by one power supply. The detailed layout of the electric circuit has been designed using the SIMLPORER code [5], which is able to simulate complex power systems.

3. Magnetic surfaces

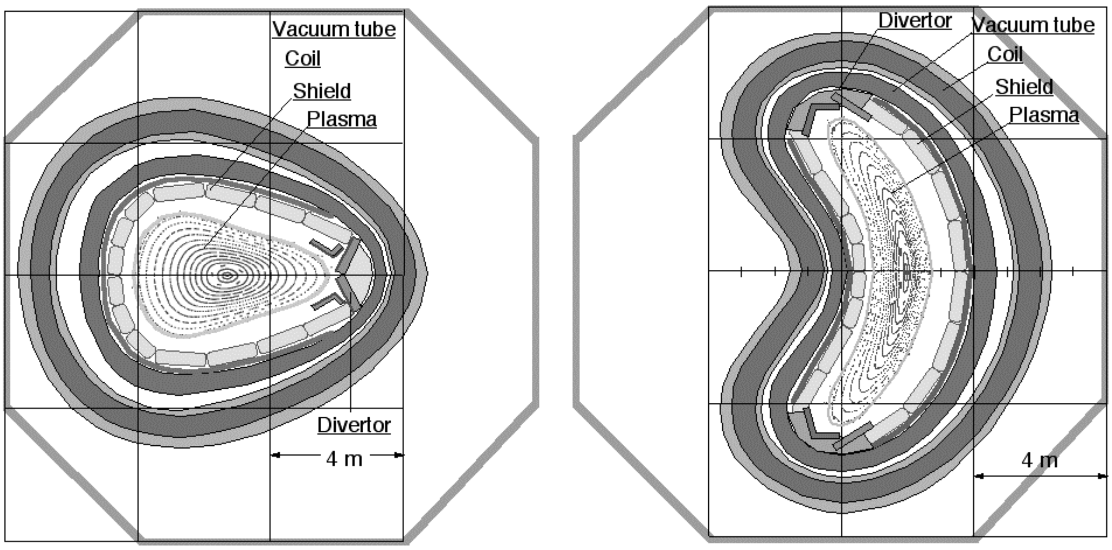

FIG. 3: Poincaré plot of HSR4/18i, <β> = 4.3%*; symmetry plane φ= 0° (right) and φ= 45° (left), units in m. The smallest distance between coils and the last closed flux surface is 1.1 m.*

Magnetic surfaces of HSR4/18i are shown in Fig. 3. The rotational transform ranges from 0.86 in the centre to 0.97 at the boundary. There is a shallow magnetic well of roughly 0.74 % which deepens with rising plasma pressure to 7.8 % at $\langle\beta\rangle = 4.3\%$. The figure shows the Poincare plots at $\langle\beta\rangle = 4.3\%$. Outside the last magnetic surface, which has an average radius of 2.1 m, the magnetic field is stochastic. This stochastic region is the remnant of the 4 islands at $\iota = 1.0$, but the basic structure of the 4/4 islands is still present and the plasma flow in this region will follow certain channels towards the divertor target plates. In the bulk plasma there are no significant magnetic islands, except for $\iota = 8/9$, where 9 small islands exist.

The reduction of the Pfirsch-Schlüter currents leads to a small Shafranov shift. This principle could be largely verified in HSR4/18i. The maximum ratio between the parallel current density and diamagnetic current density is 1.6; on average the ratio is 0.7.

4. Equilibrium and stability

Plasma equilibrium has been studied using the NEMEC and MBFE code. These computations verified the strong reduction of the Shafranov shift as the result of the optimisation. The shift of the magnetic axis at the highest beta of 4.3% is 23 cm in the triangle plane ($\varphi = 45°$) and 43 cm in the symmetry plane ($\varphi = 0°$). The magnetic surfaces in the vacuum field are shifted inward, however, due to the small but finite Shafranov shift at $\langle\beta\rangle = 4.3$ % the plasma is centered between inboard and outboard first wall as exhibited in Fig. 2.

Since the configuration of HSR4/18i is nearly identical with the configuration HSR4/18 the same stability limit will be expected. Stability analysis using CAS3D showed that the equilibrium in HSR4/18 at $\langle\beta\rangle = 4.3$ % is unstable to global modes in the boundary regions, while at $\langle\beta\rangle = 3.5\%$ these modes are stable. There is a chance that by tailoring the pressure profile global modes can also be stable at $\langle\beta\rangle = 4.3$ %. The rotational transform in HSR4/18 slightly decreases in the finite-ß case. Drift waves in the linear and non-linear approximations were also studied, with the specific geometry of the Helias configuration being taken into account. In particular, attention was focused on the effect of field line curvature and local shear on the linear growth rate of the dissipative drift waves. In cooperation with the Institute of Nuclear Research, Kiev, Ukraine, the spectrum of shear-Alfvén waves in a Helias reactor was analysed. For this reason a simplified code has been developed. Alfvén eigenmodes residing in the continuum gaps were investigated, and possible energy losses associated with the escape of circulating and transitioning α-particles were evaluated. It was found that the largest losses can result from destabilisation of mirror-induced Alfvén eigenmodes (MAE) and helicity-induced Alfvén eigenmodes with the poloidal and toroidal mode coupling $\Delta\mu=\Delta\nu=1$ (HAE_11). Destabilization of certain Alfvén eigenmodes was shown to affect transport of the partially slowed-down alphas, thus promoting removal of the helium ash from the reactor.

The Helias reactor is expected to operate at high density (central electron density of $3{\cdot}10^{20}$ m^{-3}) and moderate temperature (central temperatures of 15 keV). Under these conditions, neoclassical theory predicts that only the so-called 'ion-root' solution for the radial electric field exists, thus requiring strong optimisation of the magnetic field spectrum to minimize losses in the stellarator-specific $1/\nu$-regime. HSR4/18 is excellent in this regard, having an effective helical ripple considerably less than one per cent over the entire plasma cross-section. The effective helical ripple of the 4-period Helias configuration is about 0.4 % while the geometrical ripple is on the order of 8-10%. At this level, $1/\nu$-losses do not provide any threat to ignition.

5. Divertor concept

The divertor concept of the Helias reactor follows the same line which has been developed for Wendelstein 7-AS and Wendelstein 7-X. Although the structure of the magnetic field outside the last closed surface is stochastic, there exists a separatrix region with the 4 islands, where a strong radial transport of the outflowing plasma arises. In HSR4/18 the structure of this inhomogeneity was investigated by Monte Carlo technique following particles along field lines subject to pitch angle scattering. Anomalous transport and radial spreading of the scrape-off layer is simulated by a collision frequency leading to a radial temperature decay length of 5 cm. Heat load on the target plates is a critical issue; for keeping the thermal load below the technical limits of 5 – 10 MW/m^2, up to 90% of the alpha-particle power must be radiated

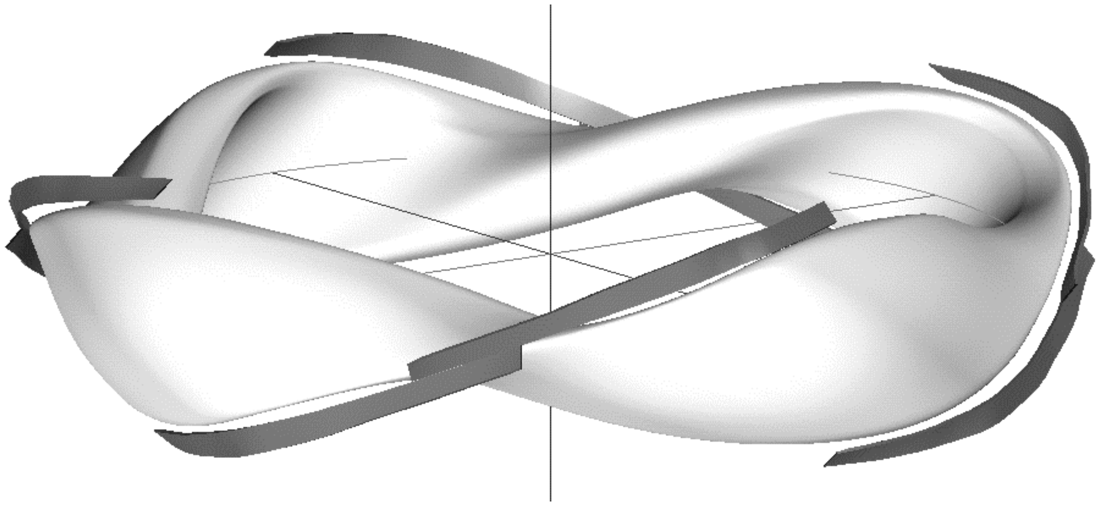

FIG. 4: Last magnetic surface and divertor target plates. Baffle plates are not shown

The divertor plates are aligned along the ι=1 islands. Monte Carlo calculations of particles have verified the efficiency of this concept. Also experiments in Wendelstein 7-AS demonstrated the advantage of the island divertor. Additional baffle plates, which are not shown in the figure, are installed in order to avoid neutral particle diffusion around the surface. Every divertor plate is 15 m long and 1 m wide. For the reason of maintenance segmentation of the target plate is necessary. The length of each segment is 2 m. Divertor segments of this size can be replaced through portholes as displayed in Fig. 1. The cross section of the vertical or horizontal ports is $1.5{\cdot}4$ m^2. However, the details of the divertor concept need further specification, in this first approach only the geometrical conditions have been considered.

6. Neutron shield

Neutronic calculations using the MCNP code are in preparation; in particular the effect of the 3-D geometry of the blanket on the activation of the structural material needs to be investigated. The real 3-D geometry of the blanket of a Helias reactor has been prepared as input data for the MCNP code and the wall loading due to the 14 MeV fusion neutrons has been calculated. The wall loading is more inhomogeneous than in a tokamak reactor which underlines the importance of a careful investigation of the 3-D geometry.

The number of neutrons emitted per second is about $5{\cdot}10^{20}$; it is the same number as in ITER FDR. This implies that neutronic calculations for ITER can be transferred also to HSR4/18i. In particular, this refers to the shielding blanket, which has been designed for ITER. This consists of modules on the order of $1.5 \cdot 1 \cdot 0.4$ m^3. The width of the elements is 0.4 m. Since the area of the first wall in HSR4/18i is about 2500 m^2 more than 1500 blanket modules are required in this device. However, the specific wall load by neutrons is 0.5 MW m^{-2} in average, which is a factor of two smaller than in ITER FDR. A peaking factor of 1.8 leads to a maximum wall load of 0.9 MWm^{-2}. The reduced neutron flux leads to a lower activation or a longer lifetime of the blanket elements.

The maximum neutron flux of 0.9 MWm^{-2} determines the width of the blanket modules, which in case of ITER FDR is 0.4 m. Because of the strong poloidal inhomogeneity of the neutron flux in HSR there are regions where the flux is only 0.2 MWm^{-2} or less. Computations are in preparation to account for this inhomogeneity and to optimise the blanket system.

7. Heating and refueling

The external heating power for reaching ignition conditions is about 50 MW, which implies that 80-100 MW should be installed in order to be on the safe side. Two options have been envisaged as heating methods: NBI heating and ECR heating. The ECRH-system is an upgraded version of the system in preparation for Wendelstein 7-X (P = 10 MW, f = 140 GHz). In HSR4/18i access is provided from the low-field side and also from the high-field side. In the triangular plane (φ = 45°) the magnetic field increases in direction of the major radius thus allowing for easy access from the high-field side. The effficiency of this heating method has been successfully verified at the Wendelstein 7-AS experiment. NBI heating is possible at various positions around the device, and in this respect no major difference to ITER exists.

The plasma parameters envisaged in the ignition experiment provide excellent conditions for pellet injection as the refuelling method. Since the lifetime of the pellet scales inversely with a weak power of the density and a strong power of the electron temperature [6,7,8] ($\tau_p \sim n^{-0.333}T^{-1.64}$) a small plasma temperature favours the lifetime of the pellet and thus the penetration length appreciably. Compared to a tokamak reactor (low density, high temperature) the lifetime of the pellet can be 4 – 4.8 times larger. Here the same velocity and the same pellet size were assumed. Thus, deep penetration of pellets and peaked density profiles are expected in the stellarator reactor. In the symmetry plane φ = 0° the distance between plasma boundary and plasma centre is less than 1 m, however in this plane pellet injection is possible only from the low-field side (see Fig. 1). Since the curvature drift reduces the penetration of the pellet it is favourable to launch the pellet from the high field side. In HSR4/18i this is possible in the triangle plane (φ = 45°), where the pellets can be injected from the outer regions of the torus. Here the flight tube is not bended, which provides no restriction to the curvature of the flight tube and the speed of the pellet.

8. Power balance

Since self-sustained burn depends on the balance between alpha-particle heating and energy transport, current scaling laws of energy confinement [9,10] have been tested with respect to their compatibility to ignition conditions. Neoclassical transport provides the minimum loss in any stellarator device. In the Helias configuration the effective helical ripple has been reduced to less than 1%. This implies that neoclassical losses do not provide an obstacle to ignition. The anomalous transport, which exists in all stellarator experiments, sets a lower limit to the plasma parameters and the size of the device. The power balance of the ignition experiment has been computed using given plasma profiles rather than from a transport code; the profiles have been selected in accordance with plasma profiles in existing stellarator experiments. This eliminates the uncertainties of the transport coefficients. The required confinement time is in the range of 2.5 – 3 s; three of the empirical scaling laws predict confinement times, which are larger. There is no need to invoke any improvement factors or H-mode confinement in order to meet the ignition conditions. The start-up scenario has been studied using the empirical scaling laws of confinement; the net heating power to reach ignition is on the order of 50 MW. A set of self-consistent plasma parameters is shown in the next TABLE 2.

The three cases in TABLE 2 differ in temperature and density; the average beta is nearly the same in all cases. The external heating power is zero.

TABLE 2: PLASMA PARAMETERS OF THE IGNITION EXPERIMENT.

	Case 1	Case 2	Case 3	
Electron Density n(0)	$3.04 \cdot 10^{20}$	$2.44 \cdot 10^{20}$	$2.14 \cdot 10^{20}$	[m^{-3}]
Line Average Density	$2.12 \cdot 10^{20}$	$1.71 \cdot 10^{20}$	$1.50 \cdot 10^{20}$	[m^{-3}]
Electron Temperature T(0)	11.0	14.0	15.0	[keV]
Alpha fraction f_alpha	6.6	8.2	9.3	[%]
Beta(0)	13.40	13.55	12.65	[%]
Average Beta	3.62	3.67	3.42	[%]
DT_Power (Total)	321.3	345.7	291.6	[MW]
Neutrons/s	$5.70 \cdot 10^{20}$	$6.13 \cdot 10^{20}$	$5.175 \cdot 10^{20}$	[1/s]
Bremsstrahlung	95.67	72.15	58.93	[MW]
Fusion Power	1608	1729	1459	[MW]
Internal Heating Power	215.5	261.8	222.2	[MW]
Energy Conf.Time	3.05	2.54	2.79	[s]
En. Conf.Time (NLHD2)	2.73	2.14	2.21	[s]
En. Conf.Time (ModLGS)	3.41	2.66	2.71	[s]
En. Conf.Time (W7)	3.76	3.03	3.10	[s]
En. Conf.Time (ISS95)	1.82	1.45	1.50	[s]
En. Conf.Time (NLHD1)	3.48	2.76	2.82	[s]

9. Summary and conlusions

It is feasible to design a fusion ignition experiment on the basis of an helical stellarator advanced (Helias) with the same conceptual goals as the ITER FDR experiment. The Helias Ignition Experiment concept shows clear advantages: there is no need to drive a toroidal current, neither transformer coils nor poloidal field coils are necessary, no provisions must be made for controlling disruptions.

The modular coil system based on conventional NbTi technology will lower the issue of mechanical forces and stresses. The winding pack consists of 1040 tons of superconducting NbTi cable and the total weight of the coil system including the support structure is estimated to be about 9600 t. Between plasma and coils enough space for a double-walled vacuum vessel and a shielding blanket is available.

Neoclassical transport does not present a threat to the ignition condition, these are determined by anomalous transport. Confinement times predicted by LGS-scaling, W7-scaling and NLHD1-scaling [9,10] are sufficient to meet the ignition condition. The road to ignition requires an external net heating power of 50 MW, which can be switched off after ignition. The Helias geometry offers several advantages for the refueling procedure by pellets. Based on these first approaches in considering some technical and physical issues one may conclude that the prospects of a Helias ignition experiment are very favourable.

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