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Abstract:

The paper discusses various options for a blanket of the Helias reactor HSR22. The Helias reactor is an upgrade version of the Wendelstein 7-X device. The dimensions of the Helias reactor are: major radius 22 m, average plasma radius 1.8 m, magnetic field on axis 4.75 T, maximum field 10 T, number of field periods 5, fusion power 3000 MW. The minimum distance between plasma and coils is 1.5 m, leaving sufficient space for a blanket and shield. Three options of a breeding blanket are discussed taking into account the specific properties of the Helias configuration. Due to the large area of the first wall (2600 m^2) the average neutron power load on the first wall is below 1 MWm^{-2}, which has a strong impact on the blanket performance with respect to lifetime and cooling requirements. A comparison with a tokamak reactor shows that the lifetime of first wall components and blanket components in the Helias reactor is expected to be at least two times longer. The blanket concepts being discussed in the following are: the solid breeder concept (HCPB), the Dual-Coolant Pb-17Li blanket concept and the water-cooled Pb-17Li concept (WCLL).

1. Introduction

The Helias reactor is an upgrade version of the Wendelstein 7-X stellarator. This is an optimised stellarator with reduced Pfirsch-Schlüter currents and reduced neoclassical transport. In order to provide sufficient stability limits for reactor purposes the number of field periods is five. The criteria guiding the scaling of this configuration to reactor dimensions are the following:

- The magnetic field should be low enough to allow for NbTi-superconducting coils.
- The plasma volume should be large enough to allow for a fusion power of 3000 MW.
- Plasma confinement should be good enough for ignition.
- Minimum distance between plasma and coils should be large enough to accommodate a breeding blanket and a shield.

These criteria lead to the following dimensions of the Helias reactor

Table 1-1: Main Parameters of the Helias Reactor

Major Radius R	22	[m]
Minor Radius a	1.8	[m]
Elongation	1.000	[]
Plasma Volume	$1.407\ 10^{03}$	[m^3]
Iota(0)	0.85	[]
Iota(a)	0.99	[]
Magnetic Field on Axis	4.750	[T]
Max. Field on Coils	10.00	[T]
Line Average Density	$2.126\ 10^{20}$	[m^{-3}]
Electron Density n(0)	$3.040\ 10^{20}$	[m^{-3}]
Electron Temperature T(0)	15	[keV]
Av. Electron Temperature	4.965	[keV]
Beta(0)	15.67	[%]
Average Beta	4.241	[%]
Diamagnetic Current	40.95	[MA]
Diamagnetic Flux	1.025	[Vs]
Fusion Power	3000	[MW]
Neutron Power	2400	[MW]
Area of First Wall	2600	[m^2]
Average Neutron Wall Load	0.92	[MW m^{-2}]
Peaking factor of Wall Load	1.7	
Av. Radiative Wall Load	≤ 0.23	[MW m^{-2}]

The aspect ratio is larger than 10, which is large compared with tokamak devices. Since the fusion power depends on plasma parameters and plasma volume, the plasma surface and hence the surface of first wall are larger than in equivalent tokamaks with the same fusion power. Writing the area of the plasma surface or the surface of the first wall in terms of volume and major radius yields

$$F \approx \pi 2^{3/2} \sqrt{V} \sqrt{R} \qquad \text{Eq. 1-1}$$

At fixed plasma volume (or volume inside the first wall) the area grows with the square root of the major radius. As a consequence, the power load on the first wall decreases with the square root of the major radius, if the fusion output is kept fixed. A large aspect

ratio stellarator has a smaller neutron wall load and smaller radiative power load than a low aspect ratio tokamak. On the other side the larger area of first wall requires a larger blanket and a larger shield than in a tokamak reactor.
In the following the geometry available for a blanket and several options of a blanket will be discussed in detail

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2. Available Space

The following figures show coil and plasma cross sections in the Helias reactor. The plasma equilibrium has been computed for $\langle\beta\rangle = 5\%$, which is sufficient to provide a fusion power of 3000 MW. It is expected that the plasma is unstable at higher beta values.
The smallest distance between plasma and coils occurs in the horizontal plane z=0 at the inboard side. The distance there is 1.5 m, which is sufficient to accommodate a breeding blanket and a shield. On the outboard side the distance is 2 m, and in the divertor region this distance is about 2.5 m. Since the configuration is not axi-symmetric the cross section varies toroidally, the three characteristic shapes are shown in Fig. 2.1 to 2.3. One field period extends toroidally from 0 to 72°, $\varphi = 0°$ and $\varphi = 36°$ being symmetry planes.

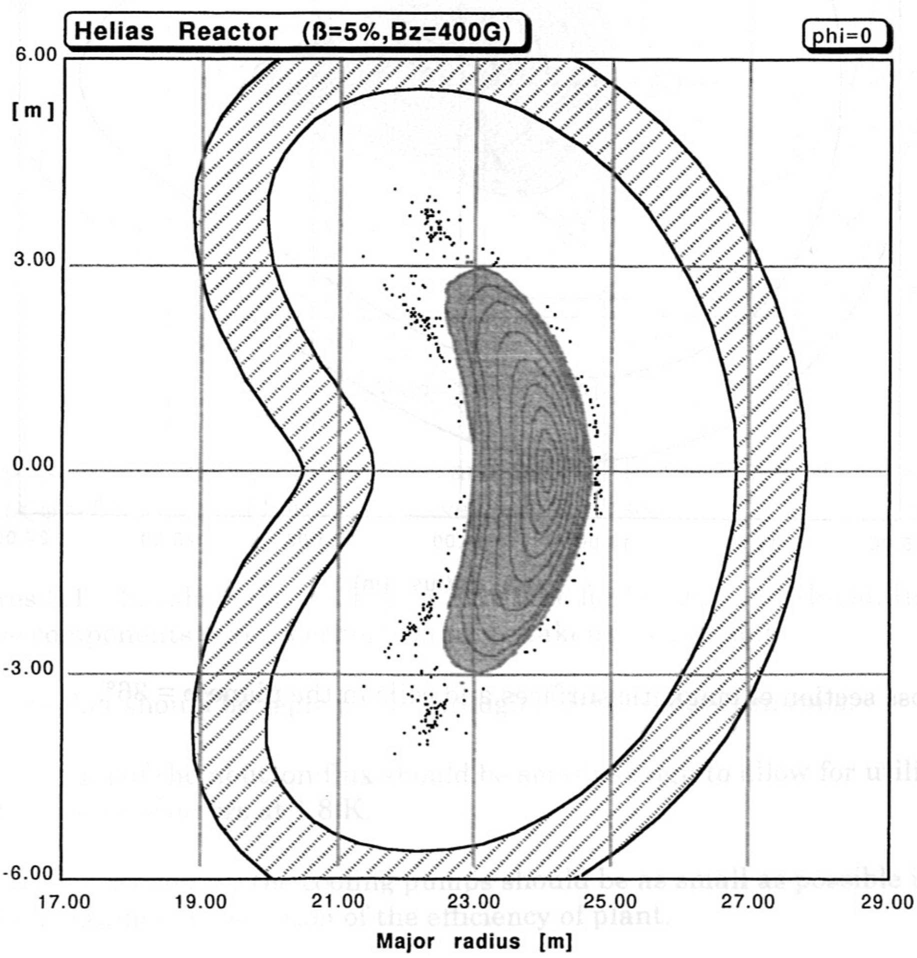

Fig. 2-1: Cross section of magnetic surfaces and coils in the plane $\varphi = 0°$.

The hatched region marks the region enclosing the modular coils and casings. The radial width is 1 m, which is the height of the coils including the steel casing. The region of closed magnetic surfaces is displayed in gray-scale, the average radius of the magnetic surfaces is 1.8 m and the total plasma volume amounts to 1400 m^3. Outside the last magnetic surface a region with open field lines exist which will be utilised for divertor action. In the following figures the intersection points of these field lines are also indicated in the figures.

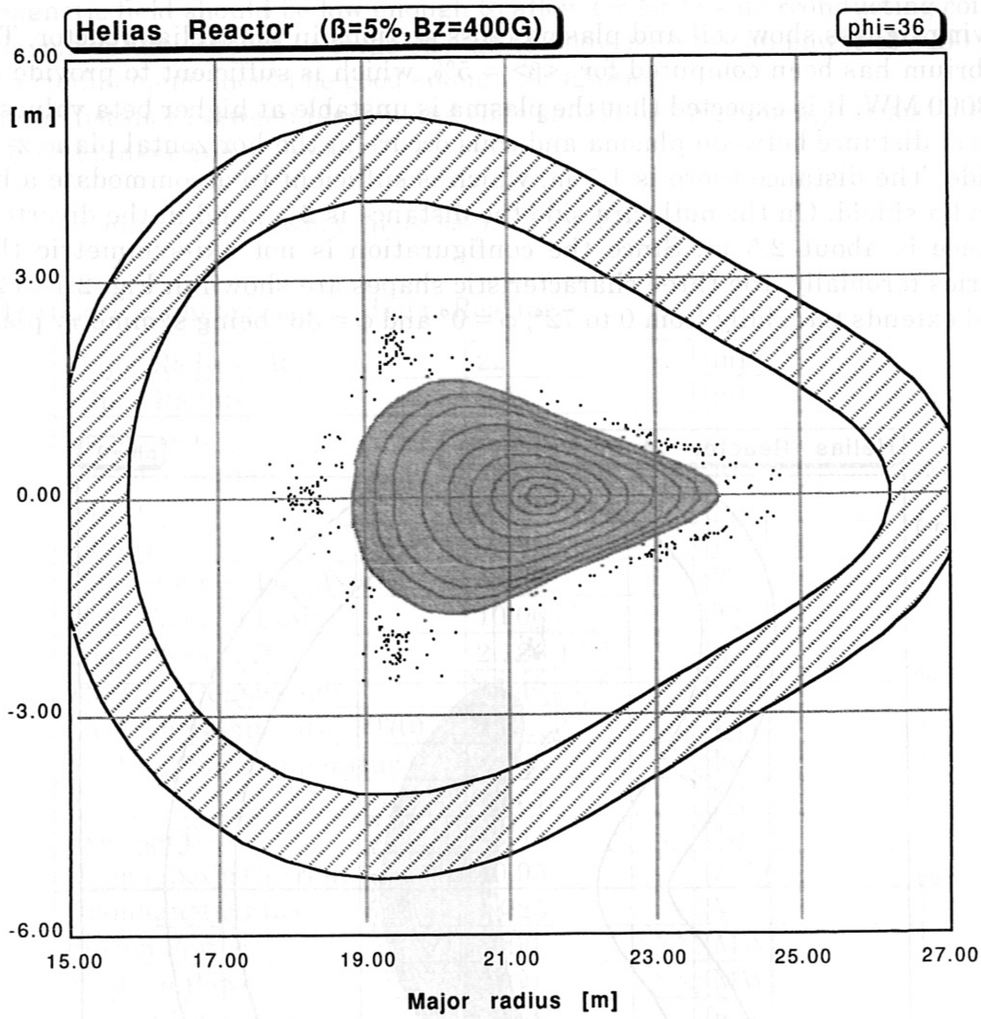

Fig. 2-2: Cross section of magnetic surfaces and coils in the plane $\varphi = 36°$.

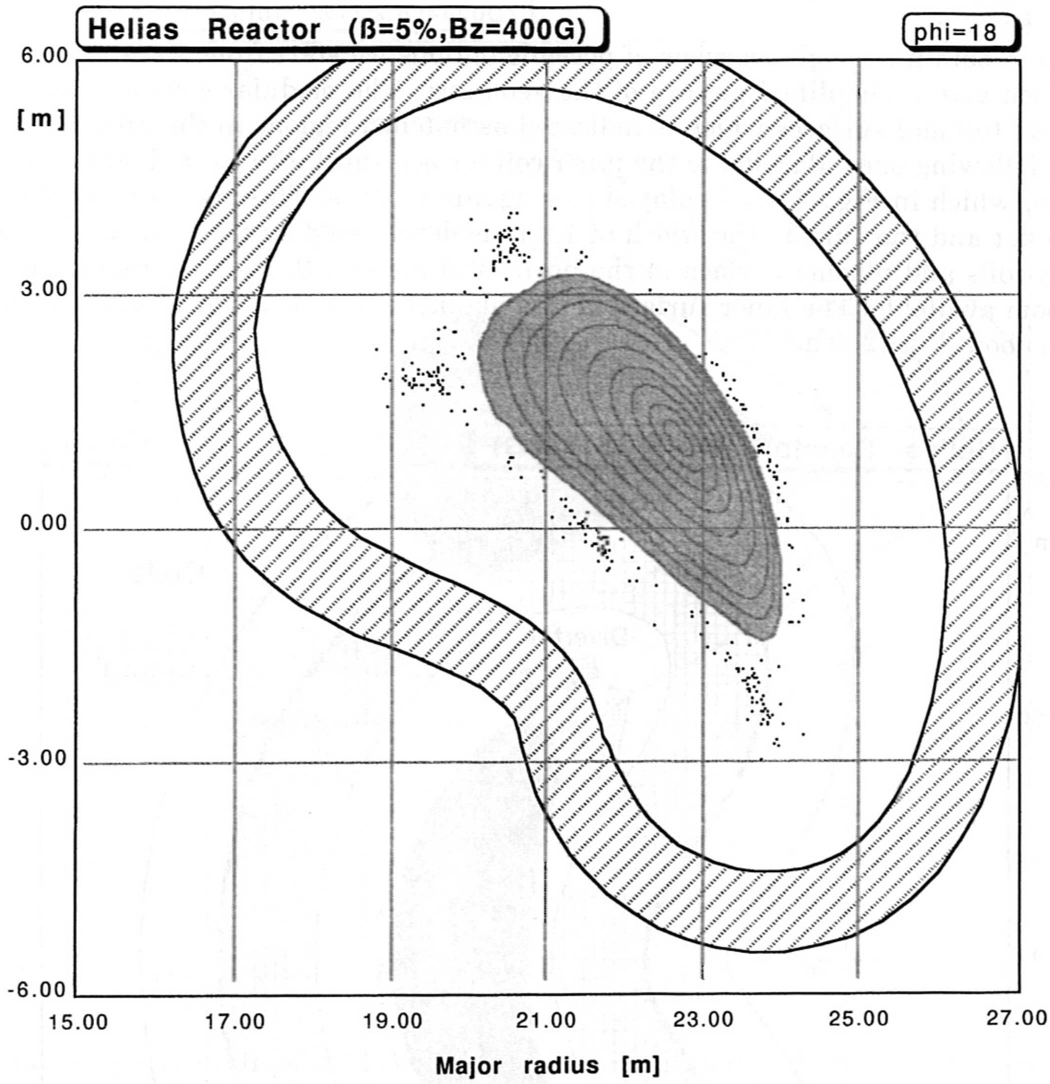

Fig. 2-3: Cross section of magnetic surfaces and coils in the plane φ = 18°. This shape also exists at φ = 54°, top and bottom, however, are reversed.

These figures 3.1 - 2.3 show the total space available for blanket and shield. In order to design these components several criteria must be taken into account:

The blanket should be replaceable through portholes between coils.

Attenuation of the neutron flux should be small enough to allow for utilisation of NbTi superconductors at 1.8 K.

The power needed for the cooling pumps should be as small as possible in order to avoid a significant reduction of the efficiency of plant.

3. Blanket Concepts

In order to obtain a rough overview of possible options we start from a system of nested tori, which give a simplified picture of the geometry. The modular coils are located between two toroidal surfaces which is indicated as hatched regions in the previous figures and the following ones. Parallel to the inner coil torus a third torus at a distance of 1.2 m is drawn, which in Fig. 3.1 is displayed in a square pattern. This region is available for the blanket and the shield. The width of 1.2 m is determined by the narrowest distance between coils and plasma surface in the horizontal plane z=0. In other regions there is more room available. The inner surface of this blanket region is the first wall the area of which is about F = 3200 m^2.

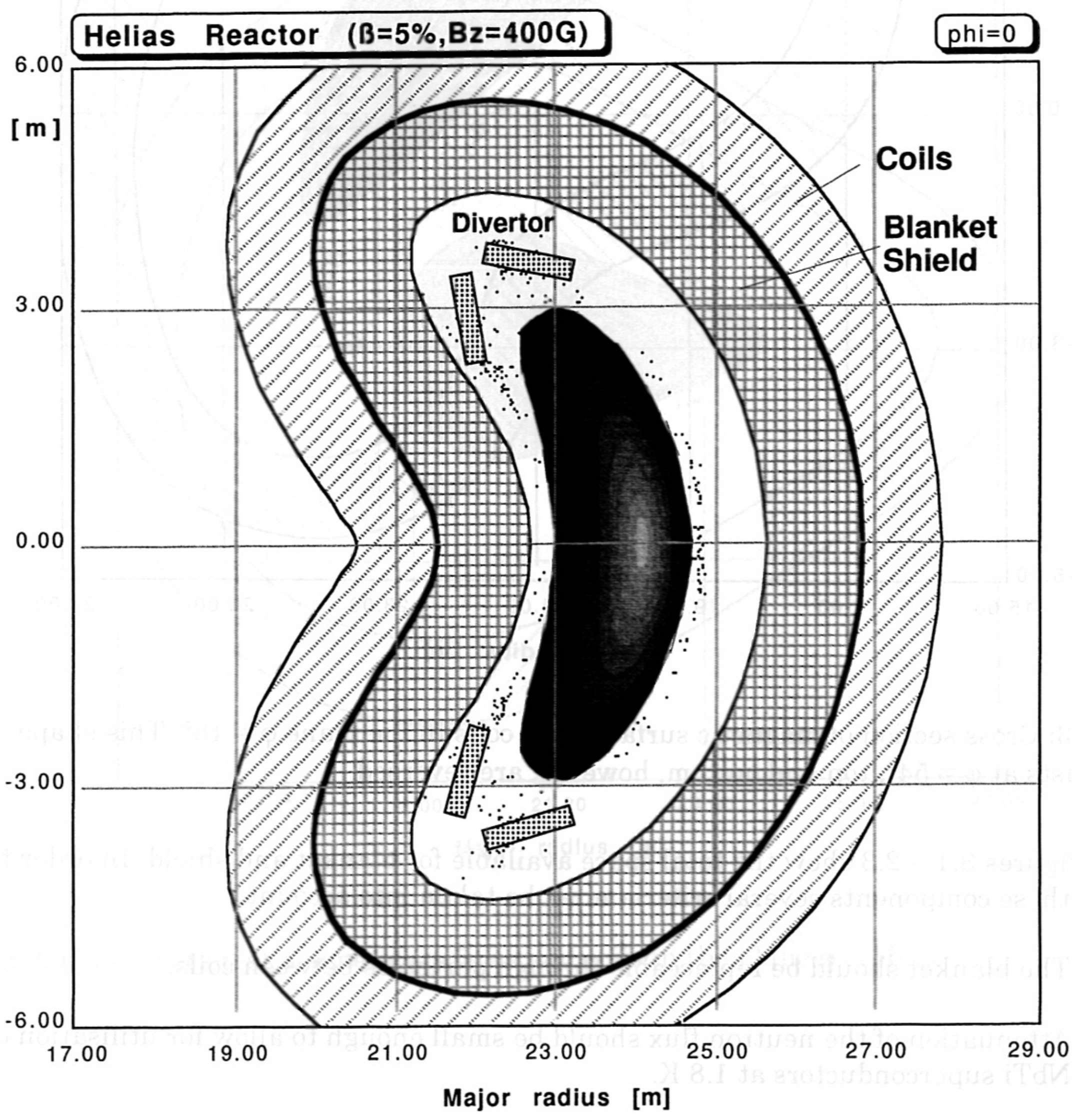

Fig. 3-1: Cross sectionof plasma, coils and blanket in the plane $\varphi = 0°$. A simplified concept of divertor plates is also shown. The width of the blanket region is 1.2m. The distance between first wall and the last closed magnetic surface is 0.3 m. Here the area of the first wall is 3200 m^2.

The average neutron wall load is 0.75 MWm^{-2}. A detailed computation of the neutron wall load on the first wall is shown in the next section.

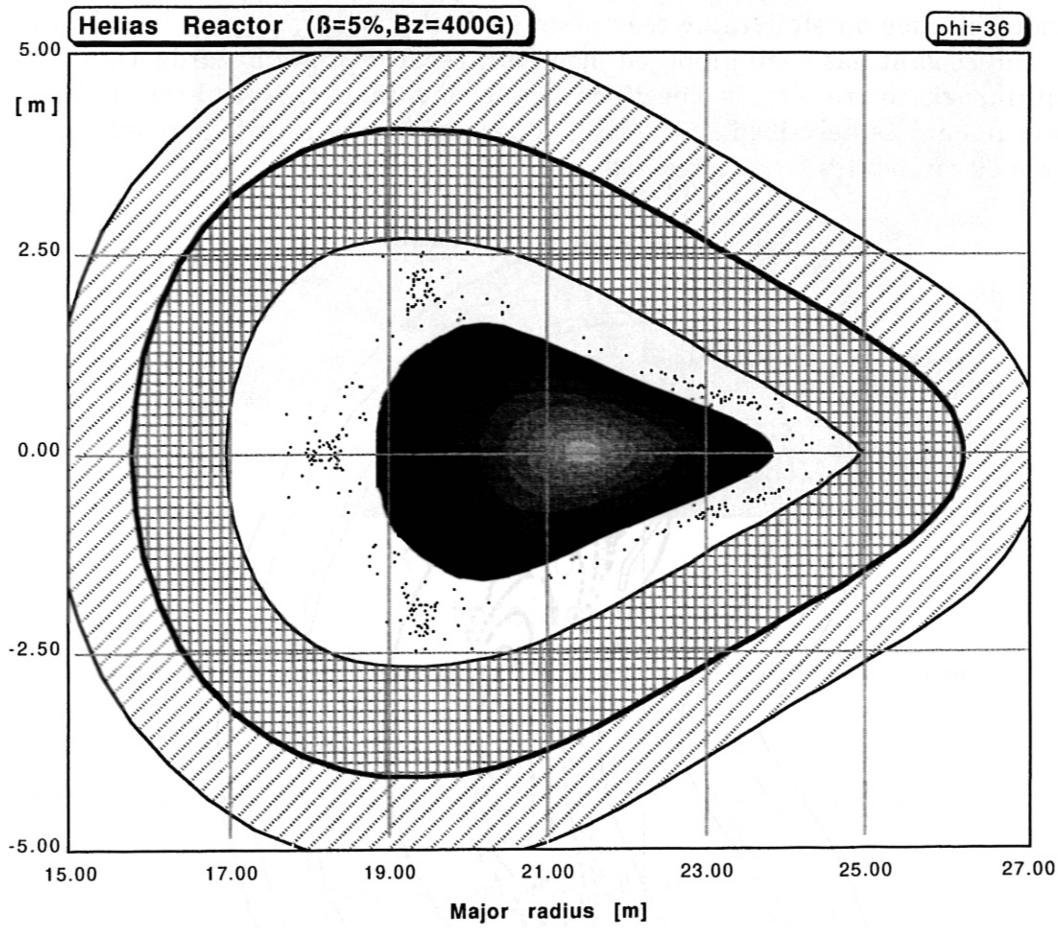

Fig. 3-2: Plasma cross section, coils and blanket in the plane φ = 36°.

The option displayed in Fig. 3.1 is one where the first wall has the largest distance to the plasma and only 1.2 m for blanket and shield everywhere. This concept offers maximum space for the divertor target plates. On the other hand, optimising the space for blanket and minimising the area of the first wall leads to an area of F = 2600 m^2. In this case the average neutron wall load would be 0.92 MWm^{-2} and the peak value about 1.6 MWm^{-2}. In the following considerations these data will be used to estimate the amount of blanket and shield. The peaking factor is 1.7.

Comparing this stellarator with a tokamak reactor SEAFP[1] with a first wall of 1080 m^2, the average neutron wall load would be 2.3 times larger: 2.1 MWm^{-2} in average and 3.2 MWm^{-2} peak value (peaking factor 1.5). The radial height of the blanket is determined by atomic processes and therefore independent of the configuration. In first approximation the volume of the breeding zone in the Helias reactor is 2.6 times as large, and, since the deposited power in the blanket is the same, this leads to a 2.6 times smaller power density in the blanket of HSR. This feature will have an impact on all blanket concepts being envisaged for stellarators.

In the following several concepts, which have been designed for tokamak reactors, will be discussed:

Helium-cooled ceramic breeder blanket (HCPB)
Helium-cooled Pb-17Li-blanket (Dual-Coolant Concept)
Water-cooled Pb-17Li-blanket (WCLL)

[1] W. Daenner et al., SEAFP Task M6, Intermediate Report, April 1993

In previous studies on stellarator reactors[2,3] a blanket concept using liquid lithium as breeder and coolant has been proposed, however, because of the hazards associated with liquid lithium there are serious objections against the use of this material. Following the safety arguments as described in the SEAFP report, liquid lithium blankets will not be considered as candidates for a Helias reactor.

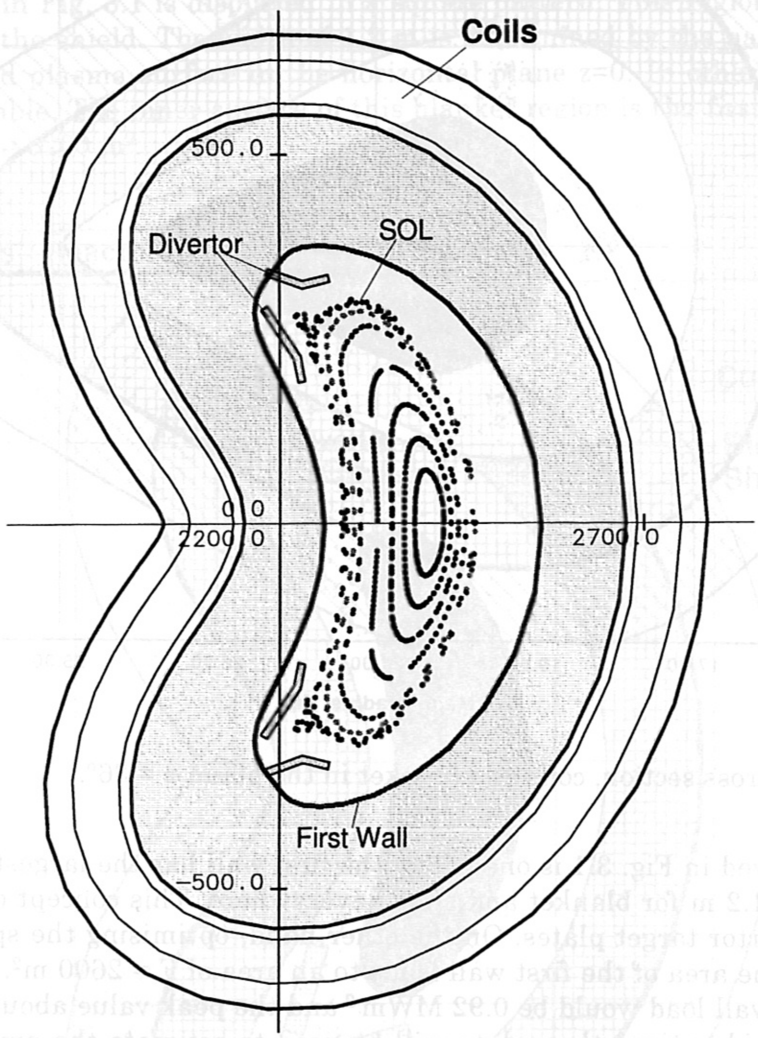

Fig. 3-3: Minimisation of first wall. Cross section of coils and magnetic surfaces in the plane $\varphi = 0°$. The area of first wall (is 2600 m^2, the minimum distance between first wall and coil case is 1.2 m. Average beta = 5%.

3.1. Neutron Wall Loading

The following figure displays the neutron wall loading in one period of the first wall in toroidal and poloidal coordinates. The contour lines are the lines of equal power density in MWm^{-2}. This power load has been calculated in 3D starting from given temperature and density distributions. The resulting pressure distribution is used to compute the finite beta equilibrium shown in Figs. 2.1 and 2.2.

[2] L.A. El-Guebaly, I.N. Sviatoslavsky, M.E. Sawan, L.J. Wittenberg, G.L. Kulcinski, Report FPA-87-4 (1987), Fusion Power Associates

[3] L.A. El-Guebaly, Report UWFDM-986, 1995, University of Wisconsin

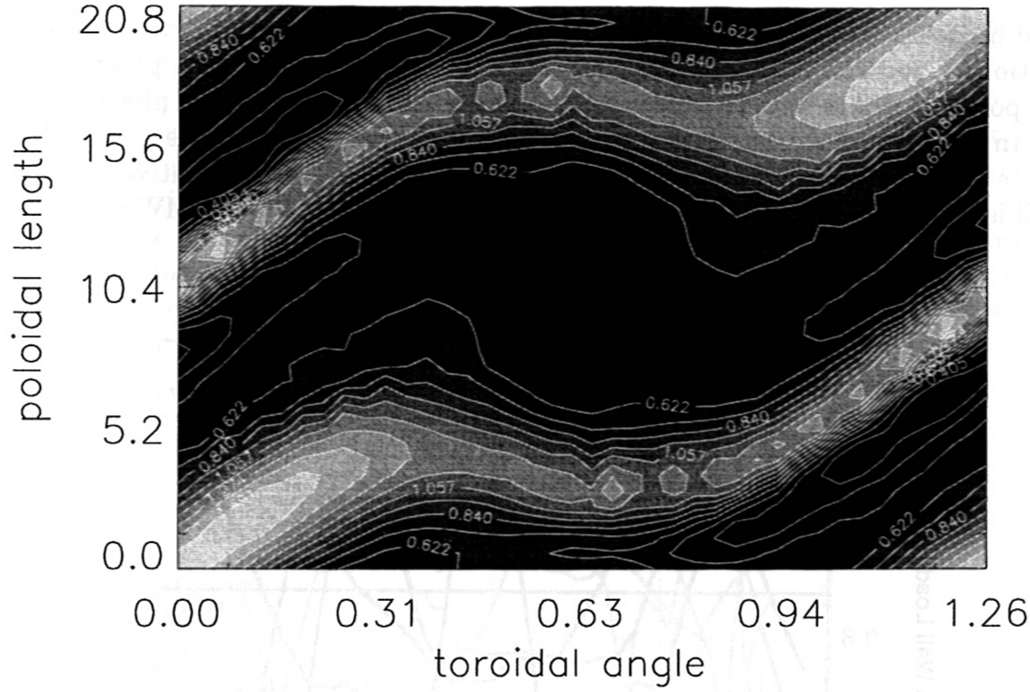

Fig. 3-4: Contour lines of equal neutron wall loading on one period of the first wall from $\varphi = 0°$ to $\varphi = 72°$.

The poloidal distribution of the neutron wall loading is shown in Fig. 3.5.

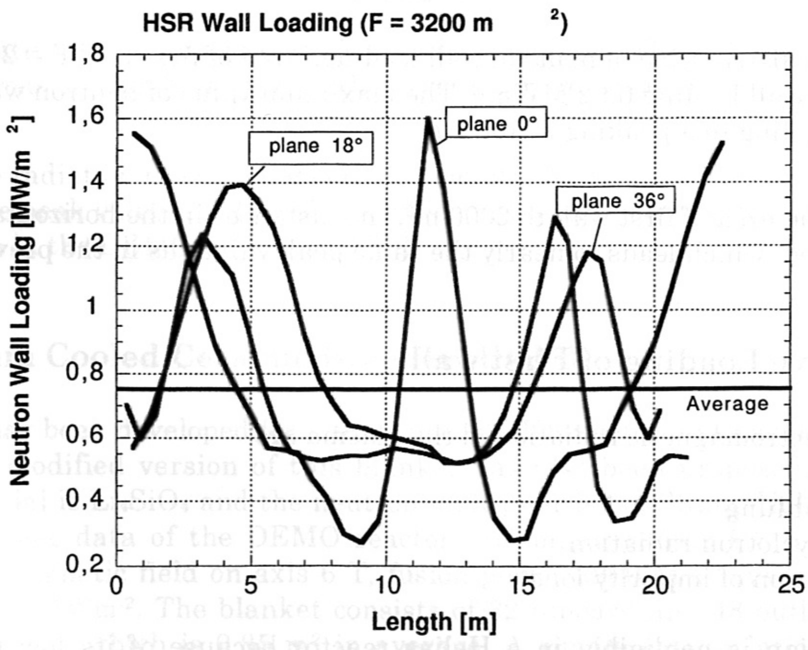

Fig. 3-5: Poloidal distribution of Nneutron wall loading in three characteristic planes: $\varphi = 0°$, $\varphi = 18°$, $\varphi = 36°$. Total neutron power 2400 MW. F = 3200 m^2. Average power density 0.75 MWm^{-2}.

The peak power load onto the first wall is 1.6 MWm^{-2}, leading to a corresponding peaking factor of 2. Some fraction of the alpha heating power of 600 MW is transferred to the divertor target plates by convection and thermal conduction and the remainig part is radiated onto the first wall by bremsstrahlung, cyclotron radiation and impurity radiation.

It should be noted that the power load in the plane at 36° (Fig. 3.2) is very similar to the distribution of neutron power in the proposed ITER device[4], there the peak value of the neutron power is 1.2 MWm^{-2}. Many results obtained during the design phase of the ITER device can be applied to the Helias reactor. This refers mainly to the power deposition and the temperature distribution in the blanket and the shield. Radiative loading of the first wall is appreaciably smaller than in the ITER device which is 0.5 MWm^{-2}.

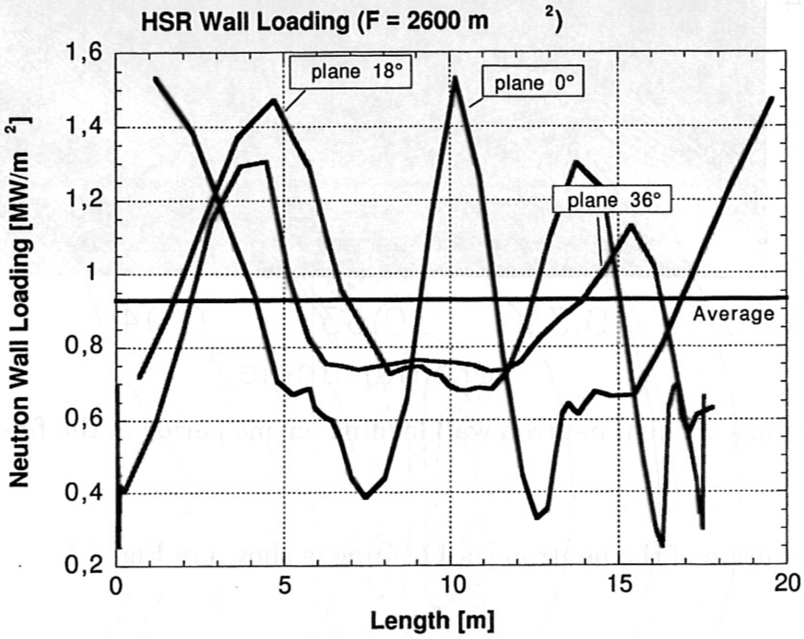

Fig. 3-6: Poloidal distribution of neutron wall loading. Area of first wall F = 2600 m^2. Average neutron wall loading 0.92 MWm^{-2}. The maximum value of neutron wall loading is 1.56 MWm^{-2} leading to a peaking factor 1.7.

When reducing the area of first wall to 2600 m^2, the distances in the horizontal plane z = 0 are kept fixed, which leads to nearly the same peak values as in the previous case.

3.2. Radiative Loading of First wall

The sources of electromagnetic radiation in the plasma are

Bremsstrahlung
Electron cyclotron radiation
Line radiation of impurity ions

Cyclotron radiation is negligible in a Helias reactor because of its low temperature. Losses by Bremsstrahlung are about 80 MW, impurity radiation is certainly higher, however, precise estimates are difficult to make since a prediction of the impurity content is very uncertain. If all alpha-heating power is converted to radiation and transfered to the wall, the average radiative wall load is 0.23 MWm^{-2} and the peak power would be 0.39 MWm^{-2}. However, this completely detached plasma is unlikely to occur, since there is always a finite particle flux onto the target plates, which also carries some energy onto

[4] ITER Nuclear Analysis Report, p. 25, G 73 DDD 1)8-06-17 W 0.2

the plates. 300 MW of all radiative power would result in an average surface load of 0.12 MWm^{-2} ($F = 2600\ m^2$) and a peak value of 0.2 MWm^{-2}.
The peaking of radiative power load depends on the radial profile of the local emissivity. Bramsstrahlung scales with the square root of the temperature and is peaked in the plasma center. If impurity accumulate in the plasma center, local emissivity is similar to the neutron emissivity and the peaking of the radiation on the first wall is not much different from the peaking of neutrons. In case of a radiative layer the emission is localized to the boundary leading to a more equalized illumination of the first wall. The following figure shows the radiative power load of a radiation source, which is localised to the boundary region.

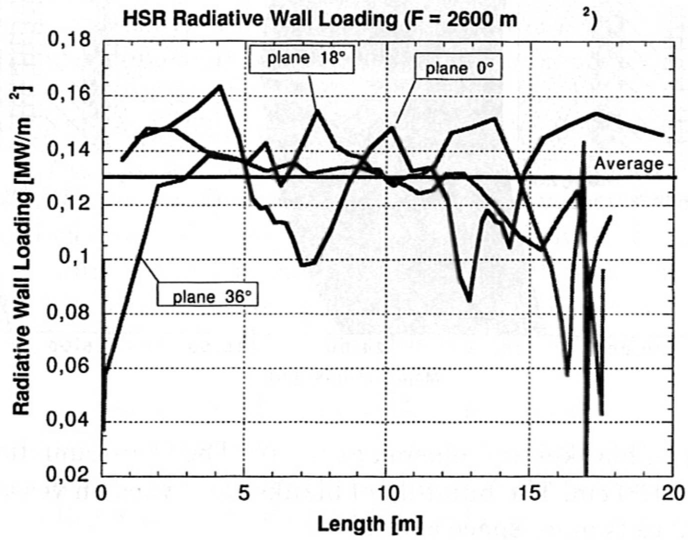

Fig. 3-7: Radiative surface load by 340 MW radiation power. The radial profile of the emissivity increases with the square of the average plasma radius.

In case of edge radiation the peaking is rather small. The average power density is 0.13 MWm^{-2} and the peak value is 0.16 MWm^{-2}, which is a factor of two to three lower than the peak value in the DEMO reactor or ITER.

3.3. Helium Cooled Ceramic Breeder (HCPB)

This concept has been developed for a tokamak DEMO reactor and has been described in KfK 5429[5]. A modified version of this blanket has also been proposed for ITER[6]. The breeding material is Li_4SiO_4 and the neutron multiplier is beryllium. Helium is used as a coolant. The basic data of the DEMO reactor are: major radius 6.3 m, minor plasma radius 1.8 m, magnetic field on axis 6 T, fusion power 2200 MW, average neutron wall loading 2.2 MW MWm^{-2}. The blanket consists of 32 inboard and 48 outboard segments, the cross section of which is 0.85 m^2 in average. A simple extrapolation to the Helias reactor with a 3.5 times larger toroidal circumference would require 112 inboard segments and 168 outboard segments. On the other hand the average length of the segment is 9.3 m, whichis smaller than in the tokamak case. Assuming a toroidal width of 1 m the 280 segments would cover the area of 2600 m^2. Since the neutron wall load on the inboard and outboard side is about the same, the radial height of the blanket segments can be made about equal on the inboard and outboard side. We assume an

[5] Mario Dalle Donne et al. *European DEMO BOT Solid Breeder Blanket*, KfK report 5429 (1994)
[6] ITER Breeding Blanket Test Modules, Design Description Document, WBS 5.6 A

average blanket height of h = 0.85 m and a breeding zone of 0.4 m. The total volume of the blanket is V = 0.85 x 2600 = 2210 m^3 and the volume of the breeding zone is 0.4 x 2600 = 1040 m^3.

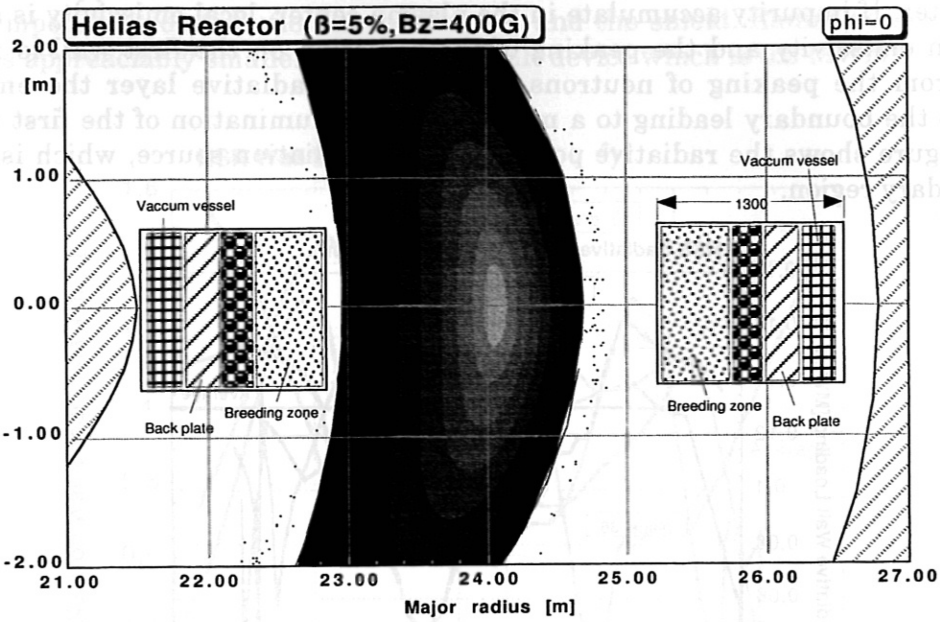

Fig. 3-8: Cross section of blanket and plasma at φ = 0°. The minimum distance between plasma and coil case is 150 cm. The build-up of blanket and vaccum vessel is 130 cm. In the outboard region there is more space available.

Table 3-1: Parameters of blanket and shield in the HCPB blanket

Major radius	22 m
Minor radius	1.85 m
Elongation	1
Magnetic field	5 T
Fusion power	3000 MW
Neutron wall loading	0.92 MWm^{-2}
Peaking factor (inboard/outboard)	1.7
Maximum surface heat flux	0.4 MWm^{-2}
Coolant Helium	
Temperature (inlet/outlet)	250/450 °C
Average pressure	8 MPa
Breeding Zone	
Breeder Li_4SiO_4, Multiplier Be pebbles	
Number of segments (inboard /outboard)	280
Length (inboard /outboard)	10 m
Total mass of breeder	201 tons
Total mass of Beryllium	843 tons
Mass of structure	1907 tons
Total mass of breeder zone	2951 tons
Weight per segment	10.5 tons
Shield/Manifolds	
Total mass	4131 tons
Total (blanket/shield)	7082 tons

The back part of the blanket consists of cooling pipes and manifolds. Assuming another 850 tons of steel for these elements yields a total weight of 3000 tons and 11.7 tons per blanket segment. The large amount of beryllium (843 tons) is certainly an overestimation since in those regions where the neutron flux is very small a breeding zone can be omitted. Another option with a reduced amount of beryllium is discussed in the SEAFP report. In this concept only the front region of the breeding zone is filled with beryllium and the rear zone with 60% ^{6}Li. Extrapolating this version to a Helias reactor requires 700 tons of beryllium, which is still a factor of 2 more than in the blanket of the tokamak DEMO reactor. A drawback of this large amount of beryllium is its capability to accumulate a large amount of tritium over the lifetime of the blanket, in case of the DEMO blanket this was about 1278 g in 300 tons of beryllium. Therefore, in the HCPB blanket of a Helias stellarator a total amount of 2 to 3 kg of tritium is expected at the end of life.
The total neutron power deposited in the blanket is the same as in the equivalent tokamak, which implies that due to the larger volume of the blanket the power density is lower by more than a factor 2. This implies that also the flow velocity of the coolant can be smaller by a factor of two and hence the pressure drop along the cooling pipes. Since the dissipated power scales quadratically with the flow velocity, the power to drive the helium coolant is reduced by a factor of more than four. However, since the volume in the Helias case is larger than in the tokamak, this power will again grow proportional to the volume. The DEMO blanket of the tokamak reactor needs a power of 175 MW to drive the helium blowers (Ref. 1). In the Helias reactor this power would be more than a factor two smaller.
Another consequence of the reduced volumetric power is a reduction of the maximum temperature in the breeding zone. All temperature gradients are reduced proportional to the power density, which in case of the HCPB blanket leads to a maximum temperature of 522 °C in the beryllium bed and 632 °C in the breeder. These data are obtained by rescaling the results presented in Ref. 1. According to Fig. 7.2 in Ref. 1 the tritium inventory in beryllium is nearly independent of temperature in the region between 450 °C and 600 °C. Swelling of beryllium is another issue, and as shown in Fig. 7.1 in ref. 1 the swelling rate decreases with temperature in the region below 500 °C. Since the thermal conductivity increases during swelling, a further reduction of the temperature below 500 °C, and, as a consequence, a swelling rate less than 8.5 % may be expected.
The reduced volumetric power density requires separate optimisation of the blanket structure as designed for the DEMO reactor. Since the neutron wall load in the Helias reactor is very similar to that envisaged in the ITER device, the design of the ITER test blanket is applicable also to the Helias reactor. Various alternatives have been considered for ITER, the neutronic analysis is described by Fischer and TsigeTamirat[7]. Due to the high ^{6}Li enrichment the energy multiplication factor in the blanket is about 1.3, which has a beneficial impact on the thermal efficiency of plant. Thermohydraulic calculations of the ITER test blanket[8] have verified the reduction of temperature as discussed previously.
The impact of disruptions and of vertical displacements on the blanket is a matter of concern for tokamaks. Eddy currents cause strong forces on the segment boxes and the resulting stresses can go beyond acceptable limits[9]. Since the optimised stellarator has neither an induced toroidal current nor a bootstrap current, this issue does not exist in the Helias reactor.

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[7] U. Fischer, H. Tsige-Tamirat, *Proceedings of the 20th SOFT*, Marseille 1998, Vol. 2, p. 1157

[8] G. Dell'Orco et al. *Proceedings of the 20th SOFT*, Marseille 1998, Vol. 2, p. 1305

[9] L.V.Boccaccini, M. Dalle Donne, *Proceedings of the 16th SOFT*, London 1990, Vol. 1, p 792

3.4. Dual Coolant Liquid Pb-17Li Blanket

Using liquid Pb-17Li as a breeding material and neutron multiplicator leads to the concept of a self-cooled or helium cooled blanket. An assessment of these blanket concepts is described in a report by FZK[10]. Helium cooling is used to cool off the heat deposited on the first wall while the volumetric power deposition is carried away by convection of the liquid metal. The critical issue of this concept is the large power loss by eddy currents in the walls of the cooling ducts, which can be minimised by coating the walls with insulating layers or by employing insulating inserts. The main data of the DEMO blanket are

Demo	Inboard	Outboard
Toroidal width [mm]	663	1064
Radial width [mm]	856	1286
Breeder zone [mm]	547	1020
Shielding structure [mm]	309	246
Iter		
Toroidal width [mm]		1026
Radial height [mm]		750

In extrapolating this concept to a Helias reactor we assume that the segments which contain the breeder material and the cooling channels are attached to the shielding structure between blanket and coils. The segments are demountable and will be replaced during the maintenance process. We start from an average segment with the length of 10 m and a cross section of 0.8 m² (0.8 x 1.0 m². This leads to the following set of parameters.

Table 3-2: Parameters of a dual-coolant blanket

Number of Segments	280
Av. Length of Segment [m]	10
Volume of segment [m³]	8
Pb-17Li (80%) [m³]	6.4
Spez. Weight(300°C)	9,5
Mass of segment [tons]	60.8
Steel (15%) [m³]	1.2
Spez. Weight	7,8
Mass [tons]	9.4
Weight of Segment [tons]	70.2
Void [5%]	
Total Mass [tons]	19656
Mass of Pb-17Li [tons]	17024
Structure [tons]	2632

The radial height of the blanket is the average of the inboard and outboard blanket proposed for the DEMO tokamak. The volume occupied by the breeding blanket is roughly 2.6 times the volume of this zone in the DEMO tokamak. Given the same fusion output this implies that the power density in the blanket is 2.6 times smaller leading to a smaller velocity of the coolant liquid in the tubes. Since the pressure drop in the cooling

[10] S. Malang et al. *Development of Self-Cooled Liquid Metal Breeder Blankets*, FZKA 5581, (1995)

system is proportional to this velocity this pressure drop in one blanket segment will be reduced by a factor 2.6.

The amount of Pb-17Li listed above is the mass in the blanket region. In addition to this the breeder material in the external ducts and the storage tanks has to be included.

3.4.1. MHD-Losses

The main objection against a liquid metal blanket is the pressure drop along the cooling channel caused by the MHD-losses. This pressure drop results from the $\boldsymbol{j}\times\boldsymbol{B}$ - forces in the cooling pipes. Extensive efforts have been made to compute these MHD-losses in tokamak geometry and to compare these with experimental findings[11] (see also ref. 4). These results can also be applied to stellarator geometry. In detail, however, the currents in the ducts depend on magnetic field and the flow velocity the impact of stellarator geometry on this effect needs to be studied. For this reason a short overview will be given on the basic relations and by applying integral methods, some general relations, which are independent of the specific geometry, will be derived. The aim is to figure out the scaling on magnetic field and volume of the blanket. The velocity of the coolant in the cooling channels is determined by the force balance

$$0 = -\nabla p + \boldsymbol{j}\times\boldsymbol{B} - \nu\Delta\boldsymbol{v} \;\; ; \;\; \nabla\bullet\boldsymbol{v} = 0 \qquad \text{Eq. 3-1}$$

ν is the viscosity of the liquid metal. Electric current and flow velocity are combined by Ohm's law

$$-\nabla\Phi + \boldsymbol{v}\times\boldsymbol{B} = \eta\boldsymbol{j} \qquad \text{Eq. 3-2}$$

Φ is the electric potential. The energy balance is obtained by multiplying the force balance Eq. 3.1 by $\boldsymbol{v}$ and integrating over the volume of the cooling channel. Using the boundary conditions

$$\boldsymbol{v}_n = 0 \;\; ; \;\; \boldsymbol{j}_n = 0 \qquad \text{Eq. 3-3}$$

on the boundary of the tubes leads to

$$\delta p\bar{v}F = \int_{V_0} \nu\left(\nabla\boldsymbol{v}\right)^2 d^3\boldsymbol{x} + \int_{V_0} \eta_0\boldsymbol{j}^2 d^3\boldsymbol{x} + \int_{V_1} \eta_1\boldsymbol{j}^2 d^3\boldsymbol{x} \qquad \text{Eq. 3-4}$$

F is the cross section of the tube at the pump, δp is the pressure drop at the pump and $\bar{v}$ the average velocity of the coolant in the outer duct. Because of the boundary conditions Eq. 3.3 the electric field drops out from the energy balance. The first term on the right hand side is the dissipated power by viscous forces, and this term is small in large tubes, where the effect of viscosity is restricted to a small layer at the wall. V_0 is the volume of the coolant in the tube and V_1 is the volume of the tube. η_0 is the resistivity of the liquid breeder and η_1 the resistivity in the wall of the tube. Ohm's law shows that the plasma current scales with $\sigma_0 v_0 B_0$, where v_0 is a reference velocity, B_0 a reference field and σ_0 the electric conductivity.

[11] L. Barleon, L. Bühler, H. Kreuzinger, L. Lenhart, H.J. Mack, S. Malang, Fusion Technolpgy 1990, Proc. 16th SOFT, Vol. 1, p. 881

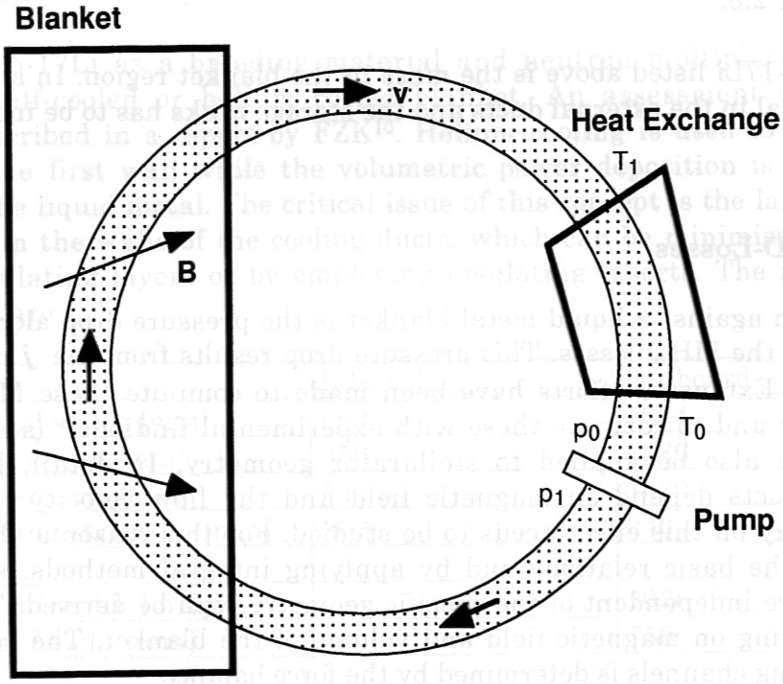

Fig. 3-9: Scheme of one segment of the Pb-17Li-cooling system. T_1 is the temperature at the inlet of the heat exchange system and T_0 the outlet temperature. In the FZK- concept these data are: $T_1 = 425$ °C, $T_0 = 275$ °C. $\delta p = p_1 - p_0$ is the pressure drop at the pump.

The velocity is written as $\boldsymbol{v} = v_0 \boldsymbol{u}$, where $\boldsymbol{u}$ is a dimensionless velocity. From Ohm's law it follows that the current density is proportional to $v_0 B_0$: $\boldsymbol{j} = \sigma_0 v_0 B_0 \boldsymbol{J}$. The dimensionless current results from Ohm,'s law

$$-\nabla\Psi + \boldsymbol{u}\times\boldsymbol{b} = \sigma_0 \eta \boldsymbol{J} \quad ; \quad \boldsymbol{b} = \frac{\boldsymbol{B}}{B_0} \quad ; \quad \Psi = \frac{\Phi}{v_0 B_0} \qquad \text{Eq. 3-5}$$

Inserting this into Eq. 3.4 yields a scaling of the pressure drop

$$\delta p = C_0 \sigma_0 v_0 B_0^2 \frac{V}{F} \qquad \text{Eq. 3-6}$$

$V/F = L$ is a measure for the length of the cooling channel. C is a geometry factor and calculated by

$$C_0 = \frac{\sigma_0}{\overline{u}\,V}\left(\int_{V_0} \eta_0 \boldsymbol{J}^2 d^3\boldsymbol{x} + \int_{V_1} \eta_1 \boldsymbol{J}^2 d^3\boldsymbol{x}\right) \qquad \text{Eq. 3-7}$$

The dissipated power is given by Eq. 3.4

$$P = C_1 \sigma_0 v_0^2 B_0^2 V \qquad \text{Eq. 3-8}$$

The second geometrical factor is

$$C_1 = \frac{\sigma_0}{V}\left(\int_{V_0} \eta_0 \boldsymbol{J}^2 d^3\boldsymbol{x} + \int_{V_1} \eta_1 \boldsymbol{J}^2 d^3\boldsymbol{x}\right) \qquad \text{Eq. 3-9}$$

$V = V_0 + V_1$ is the total volume of the cooling channel including the walls. The factors in front of these scaling laws are determined by the geometry of the cooling system. If the cooling channels are insulated by non-conducting inserts, the second terms in Eqs. 3.7 and 3.9 are negligible.

It is obvious from Ohm's law that a flow in parallel direction to the magnetic field does not lead to currents and hence power dissipation. In the inhomogeneous field of stellarator, however, it seems to be rather difficult to align the cooling channels parallel to magnetic field lines.

The coefficients C_0, C_1 depend on the details of the current density, which is the result of Ohm's law (Eq. 3.5) and the specific boundary conditions. In case of an insulating wall all currents are confined in the volume V_0, whereas in case of conducting walls the circuit is closed over the walls. In the latter case the dissipated power is appreciably larger than in case of insulating walls. Detailed computations of this feature are given by Malang et al. (ref. 4). If the walls are insulated the dominating contribution to the dissipated power is coming from the Hartmann layer at the boundary and the coeffients C_0, C_1 scale inversely with the Hartmann number M.

$$C_1 \propto \frac{1}{M} \quad ; \quad M = aB_0\sqrt{\frac{\sigma}{\rho\nu}} \qquad \text{Eq. 3-10}$$

The velocity of the coolant is determined by the need to transfer the energy deposited by the neutrons in the blanket to the heat exchange system. The equation governing the temperature in the cooling tube is

$$\rho c_p \boldsymbol{v} \bullet \nabla T + \nabla \bullet \boldsymbol{q} = Q \quad ; \quad \boldsymbol{q} = -\chi \nabla T \qquad \text{Eq. 3-11}$$

ρ is the density, c_p the specific heat, χ the heat conductivity and Q the power density in the liquid metal. Convection is the dominating mechanism to carry the energy to the exchange system (large Peclet number). Assuming good thermal insulation of the ducts results in the boundary condition $q_n = 0$ on the boundary of the wall of the tube.
Integrating Eq. 3.11 over the entire loop yields:

$$\rho c_p \left(T_1 - T_2\right) \iint \boldsymbol{v} \bullet d\boldsymbol{f} = \iiint_{V_0} Q d^3\boldsymbol{x} = P_{in} \qquad \text{Eq. 3-12}$$

P_{in} is the total power deposited in the blanket segment. Writing the flux of the coolant as $v_0 F \overline{u}$ - F is the cross section of the tube at the intlet to the heat exchange system and v_0 a reference velocity – we find the scaling of the velocity in the form

$$v_0 = \frac{P_{in}}{\rho c_p \overline{u} F \delta T} \qquad \text{Eq. 3-13}$$

Let be N the number of blanket segments and P_{tot} the total deposited neutron power in the blanket we can also write this equation as

$$v_0 = \frac{P_{tot}}{\rho c_p \overline{u} N F \delta T} \qquad \text{Eq. 3-14}$$

This equation shows the expected result: the velocity of the coolant liquid decreases as the number of blanket segment increases. Combining Eq. 7 and Eq. 11 shows that the MHD losses scale as

$$P = C_1 \frac{\sigma_0}{\left(\rho c_p \delta T \overline{u}\right)^2} P_{tot}^2 B_0^2 \frac{V}{N^2 F^2} \qquad \text{Eq. 3-15}$$

The total pumping power of all segments is

$$P = C_1 \frac{\sigma_0}{\left(\rho c_p \delta T \bar{u}\right)^2} P_{tot}^2 B_0^2 \frac{V_{tot}}{N^2 F^2} \qquad \text{Eq. 3-16}$$

Since the total volume of the breeding zone grows linearily with major radius and since the number of segments also grow linearily with major radius, we obtain the result expected above: the power needed to compensate the MHD losses decreases with the volume and hence with major radius.

3.4.2. Helium Cooling of First Wall

In the extreme case of a detached plasma where all alpha-heating power (600 MW) is dumped onto the first wall in form of radiation, the power load would be 0.23 MWm^{-2} in average and 0.4 MWm^{-2} peak power density. Using more realistic assumptions the radiative power onto the first wall is assumed to be 400 MW, which yields 0.154 MWm^{-2} average power density and 0.26 MWm^{-2} peak power density. This estimate assumes volume radiation, where the peaking factor is similar to the peaking factor of neutron emission. In case of boundary radiation the peaking factor is smaller. In comparison, the specifications of the tokamak DEMO reactor are: Average heat load on the wall 0.4 MWm^{-2} and 0.5 MWm^{-2} peak value[12].

The average radiation power to be removed in the Helias reactor by one blanket segment is P = 600/280 = 2.14 MW in the extreme case of a detached plasma and 1.43 MW in the more realistic case.

3.4.3. Comparison to a Tokamak Reactor.

In comparing a tokamak reactor and a Helias reactor with the same fusion output it is obvious that the main difference originates from the volume of the breeding zone and the area of the first wall. Since the radial height of the breeding zone is fixed, the volume is roughly proportional to the area of the first wall. This surface is about 2.6 times larger than the first wall of a tokamak power reactor. Accordingly the surface load per square meter by radiation is reduced by a factor of 2.6 leading to a smaller power to be removed on each blanket segment. The helium flow rate and the pressure drop can be reduced appreciably.

As shown above, the pressure drop in the Pb-17Li-coolant is reduced compared with the equivalent tokamak leading to a pumping power, which is roughly by a factor of 2.6 smaller than in the tokamak reactor. Furthermore, since the magnetic field in a Helias reactor is smaller, this also reduces the MHD-losses. In case of conducting walls this reduction is quadratic in the magnetic field and in case of insulated ducts it scales linearily with the magnetic field. The following figure shows the magnetic field in a poloidal plane of a Helias reactor.

The maximum field in the inboard blanket is 7 T and 6.4 T on the first wall. On the outboard segment the maximum field is 5 T and the minimum value 4T. The magnetic field on axis is 5 T. On the inboard side of tokamak reactor the magnetic field is about 9 T. The influence of the magnetic field is only linear in the case of insulating walls, therefore the reduction of the cooling power by the reduced magnetic field is small but not negligible.

[12] Nuclear fusion Project, Annual Report Oct. 1991 – Sept. 1992, KFK 5099, p. 124

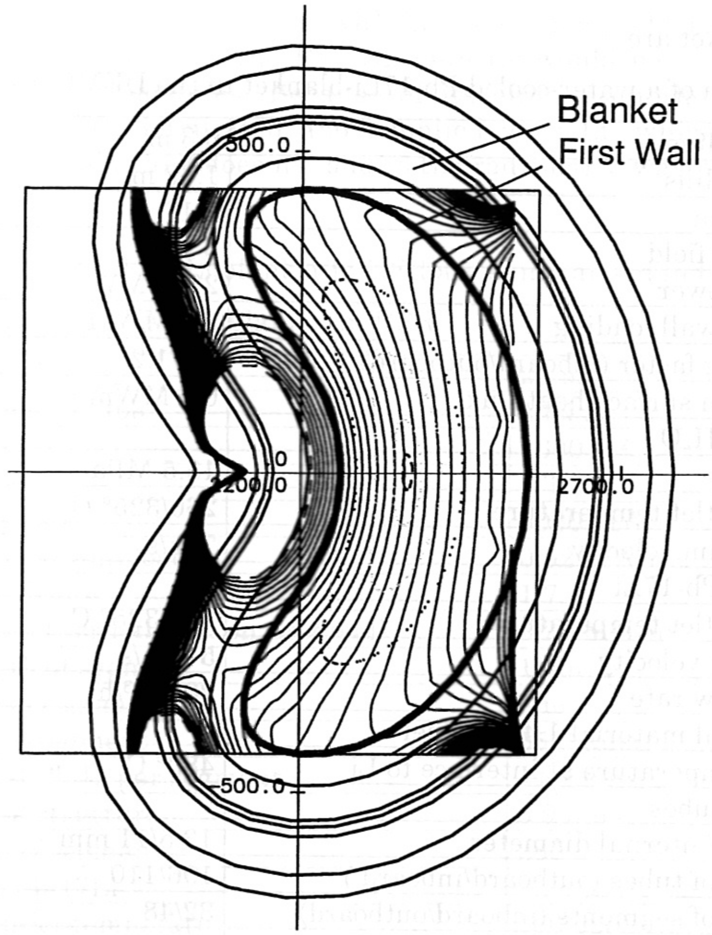

Fig. 3-10: Magnetic field of the Helias reactor in the plane φ = 0°.

With respect to the internal structure of the blanket modifications of the proposed tokamak blanket are not necessary. The main difference to the tokamak concept is the non-axisymmetry and the curved shape of the inboard blanket. As seen in Fig. 3.4 the height of the blanket is 10 m; the equivalent height in the DEMO tokamak is about 14 m, which will reduces the length of the cooling tubes in the Helias case.

3.5. Water-Cooled Pb-17Li Blanket

This concept has been developed for the DEMO tokamak[13],[14] and for the ITER tokamak[15]. The Pb-17Li-blanket cooled by water is also an attractive candidate as a blanket in the Helias reactor since it avoids the large amount of MHD-losses. It can be adapted to the Helias reactor with minor modifications. These modifications mainly concern the geometrical shape of the blanket segments, which have to be aligned to the 3D-geometry of the first wall. A tritium breeding rate of 1.19 has been computed for the DEMO blanket[16] Furthermore, the smaller local power deposition has its bearing on the flow velocity of the water and the liquid breeder. Since the cooling is performed by the water, the velocity of the liquid breeder is rather small, thus avoiding the large MHD-losses of the self-cooled Pb-17Li concept.

[13] Ph. Labbe, L. Giancarli et al., 16th SOFT, London 1990, Vol. 1, p. 983

[14] L. Giancarli et al. Fusion Technology 21, (1992), 2081

[15] Design Description Document DDD 5.6 C

[16] L. Petrizzi, L. Giancarli, 18th SOFT, Karlsruhe 1994, Vol. 2, p. 1369

The data of the blanket are

Table 3-3: Main data of a water-cooled Pb-17Li-blanket in the DEMO are

Major radius	6.3 m
Minor radius	1.82 m
Elongation	2.17
Magnetic field	6 T
Fusion power	2200 MW
Neutron wall loading	2.2 MWM^{-2}
Peaking factor (inboard/outboard)	1.4/1.2
Maximum surface heat flux	0.5 MWm^{-2}
Coolant H_2O	
Pressure	15.5 MPa
Inlet/outlet temperature	260/325° C
Maximum velocity	7 m/s
Breeder Pb-17Li	
Inlet/outlet temperature	260/325° C
Average velocity	5 mm/s
Total flow rate	300 m^3/h
Structural material 1.4914 steel	
Max. Temperature at interface to Li	480° C
Coolant tubes	
External/internal diameter	13.5/11 mm
Number of tubes (outboard/inboard)	196/110
Number of segments (inboard/outboard)	32/48
Av. neutron power per segment	22 MW
Weight of segment (inboard)	53 tons
Weight of Pb-17Li	46 tons
Weight of structure	7 tons
Weight of segment (outboard)	75 tons
Weight of Pb-17Li	65 tons
Weight of structure	10 tons
Total weight of segments	5296 tons
Total weight of Pb-17Li	4592 tons
Weight of structure (steel)	704 tons
Length of segment (inboard/outboard)	14/12 m
Operation time (70 dpa)	20000 hours (2.3 y)

Deatails of this concept are described in a paper by L. Giancarli et al.[17]. In extrapolating this concept to the Helias reactor the main difference arises due to the large major radius of HSR which needs 280 segments from which 28 have different shapes. The average length of the blanket segments in HSR is 10 m, which makes the segments a little smaller than the DEMO segments. Since the power deposition in the HSR blanket is reduced the velocity of the coolant and as a consequence the pressure drop in the cooling ducts is also reduced. The flow velocity of water can be reduced by the factor 2.6 compared with the tokamak reactor.

The flow velocity of the Pb-17Li eutectic alloy is limited from below by the requirement to minimize the permeation of T into the water coolant. A typical number is 5 mm/s, which for the reason mentioned is not different in the Helias reactor. A drawback of this

[17] L. Giancarli, Y. Severi, L. Baraer, P. Leroy, J. Mercier, E. Proust, J. Quintric-Bossy, *Fusion Technology* **21** (992) 2081

concept is the heavy mass of the breeder with imposes some problems on the structure. However, in case of maintenance the Pb-17Li eutectic would be removed prior to the exchange of the blanket modules. Because of its circulation the liquid breeder is continuously reprocessed, detritiated and supplied with Li, which implies that the breeder material is reusable and does not add to the radioactive waste after shut down of the reactor.

The data of the water-cooled Pb-Li blanket in HSR are summarised in the following table

Table 3-4: Main data of the WCLL blanket in HSR

Major radius	22 m
Av. plasma radius	1.85 m
First wall	2600 m^2
Magnetic field	6 T
Fusion power	3000 MW
Av. neutron wall loading	0.92 MWm^{-2}
Peaking factor	1.7
Maximum thermal load on first wall	0.23 MWm^{-2}
Av. length of segment	10 m
Volume of segment	8 m^3
Weight of segment	55.6 tons
Weight of Pb-Li per segment	44.7 tons
Weight of structure	6.9 tons
Number of segments	280
Breeder Pb-17Li	
Inlet/outlet temperature	260/325° C
Average velocity	5 mm/s
Coolant H_2O	
Pressure	15.5 Mpa
Inlet/outlet temperature	260/325° C
Maximum velocity	?
Av. nuclear power in segment	8.6 MW
Number of different shapes	28
Total weight (with breeder)	14450 tons
Weight of Pb-17Li	12525 tons
Weight of structure	1920 tons
Volume of segments	2240 m^3
Operation time (70 dpa)	47800 hours (5.5 y)

The lifetime of the blanket in the DEMO reactor is mainly determined by the material of first wall (martensitic steel MANET) and the neutron flux through the wall. The limits in the DEMO reactor are 20000 hours or 70 dpa in the structural material. In HSR this number of displacements per atom (dpa) is reached after 47800 hours of operation. This extrapolation is based on the averaged neutron wall load where the differenc is a factor 2.4. Since the in tokamaks the peaking factor is less the 1.7 (1.5 in SEAFP) an extrapolation on the basis of peak neutron wall load yields a factor 2.06 in lifetime. The lifetime of the components with maximum power load would 41200 hours. In a power reactor, where 150 dpa are envisaged the liftime would be 88400 hours (about 10 FPY).

4. Conclusions

In designing the blanket for the Helias reactor several features of the stellarator turn out to be of great importance

- The absence of disruptive instabilities and vertical displacement events.
- A large area of the first wall
- A large volume of the blanket region.
- A relatively low magnetic field.

The absence of disruptions implies absence of eddy currents in the blanket and therefore absence of dynamic mechanical forces on the structural elements. Furthermore, erosion of plasma facing components and first wall by disruptions does not occur. Due to the large area of the first wall (typically $F \geq 2600\ m^2$) the average neutron wall loading is below 1 MWm^{-2}, which in comparison to a tokamak power reactor (P_N = 2400 MW, F = 1000 m^2) is roughly 2.6 times smaller. In a first approximation the lifetime of first wall and blanket segments is a factor two larger - or more - than in the equivalent tokamak reactor. The details depend on the specific design of the blanket, on the shape of the plasma and the first wall. Since the magnetic field (in case of NbTi-superconductors) is slightly smaller in the Helias reactor, MHD-losses in liquid metal blankets will be reduced.

A major difference between both reactor concepts exists with respect to the neutron power density in the blanket. Since the radial build-up of the blanket is determined by nuclear processes, breeding, scattering and slowing down, this dimension is independent of the specific configuration. For this reason the volume of the breeder zone is proportional to the area of the first wall. Accordingly the power density in the blanket is reduced leading to relaxed requirements on the cooling system. Independently of the specific blanket concept this property leads to a reduced velocity of the coolant and a reduced pressure drop in the cooling pipes. The power to drive the coolant scales with the square of the velocity, which leads to a large reduction of the pumping power per segment. On the other hand, the number of segments is by more than factor two larger in the Helias reactor, which implies that the cooling power is only reduced inversely proportional to the blanket volume.

The number of segments in HSR22 is 280; in one field period there are 56 blanket segments. Because of the symmetry of the stellarator configuration there are only 28 segments with different shapes. This poses some problems to the maintenance procedure, which have not been analysed in detail so far. The lifetime of first wall and blanket segments is determined by the maximum allowable neutron fluence, which depends on the material and composition of the blanket. In this respect there is no difference between stellarators and tokamaks. The lifetime grows inversely to the neutron flux. Using this simple argument the lifetime of blanket components in the Helias reactor is 2.6 times larger than in a tokamak reactor. Due to a stronger peaking of the neutron wall loading in the Helias reactor this improvement factor may be reduced to two. Although during maintenance 280 blanket segments have to be replaced, the accumulated nuclear waste over the lifetime of the reactor (e.g. 30 years) is about the same in both reactor types. Details of the maintenance procedure have not yet been elaborated.

From the technical point of view all three blanket concepts are applicable to the Helias reactor. Every concept has its merits and drawbacks, and only an integrated design including shield, vacuum vessel and maintenance procedure can prepare the basis to select a specific concept.