MODULAR STELLARATOR REACTORS AND PLANS FOR WENDELSTEIN 7-X

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ABSTRACT

Modular stellarator reactors of the Advanced Stellarator type appear to have the potential to offer desirable reactor properties: i) a single NbTi coil system is sufficient; ii) steady-state operation is inherent and rests upon refuelling and exhaust only (no current drive system is needed); iii) there is no possibility for a major current disruption because there is no net toroidal current; iv) reactor volume, mass and magnetic field energy are comparable to those of tokamak reactors. The Wendelstein 7-X experiment will provide an integrated concept test which is needed for producing convincing predictions on the properties of this reactor approach.

I. INTRODUCTION

With the start of construction of experimental fusion reactors, like ITER, getting closer, it is now becoming urgent to check whether after ITER the presently followed tokamak approach will automatically lead to the optimum fusion reactor concept, or whether there are other, perhaps rather congenial approaches, which might offer decisively more desirable solutions. This comparison has to be made for full-scale reactor conditions and has to include all questions of relevance to reactor performance like unit size, power and particle exhaust, pulse length, power density, power loadings, plasma stability, technical requirements, maintenance approach, costing, etc.
In this paper it will be argued that the Advanced Stellarator has indeed the potential to offer a set of more desirable reactor properties than the tokamak system. Since the stellarator differs only in some prospects from a tokamak (whereas in most prospects is basically very similar to it and shares with it most of the beneficial ones of the tokamak) the comparison is best made by concentrating on the differences between the two systems. This way the comparison becomes relative and is thus much more conclusive than a point to point comparison between two independently developed and designed reactors.

The stellarator is a concept for confining toroidal plasmas with magnetic fields generated by currents exclusively outside the plasma region. The plasma contributes only by reactive, pressure-driven (in the case of the Advanced Stellarator essentially diamagnetic) currents. The stellarator shares with the tokamak the basic concept of nested magnetic surfaces for achieving confinement. However, a net toroidal plasma current, as needed in tokamaks, is not required in stellarators. Thus, stellarators without this current can be operated in steady state, without disruptions, without request for an external current drive system, and without need for more than one single coil system for generating the confining magnetic field. These properties indeed are of major importance for fusion reactors. After ignition, a stellarator reactor would work continuously on refuelling and exhaust alone. Since a net toroidal current provides free magnetic energy to drive instabilities in particular of disruptive nature, this reservoir is minimum in stellarators since they do not have this current. The stellarator is the toroidal confinement system with the smallest free energy.

In Section II, a short description of those properties of Advanced Stellarators will be given which are essential for the intended comparison of their reactor properties with those of tokamaks. Section III will define a reactor based on the Advanced Stellarator concept. Section IV will then provide an analysis of possible advantages and disadvantages of stellarator as compared to tokamak reactors.

Section V will analyse some properties of Advanced Stellarators which - before drawing final conclusions - have to be checked for their compatibility with the expected reactor performance. Section VI will deal with WENDELSTEIN 7-X, which is planned to provide an integrated concept test to yield convincing predictions on the properties of ignited plasmas in Advanced Stellarators. Section VII will then provide a short summary and conclusions.

II. SHORT DESCRIPTION OF BASIC PROPERTIES OF ADVANCED STELLARATORS

The basic stellarator concept was invented by Lyman Spitzer at Princeton in the early 1950's[1]. IPP's interest in this concept started only a few years after this date, and from then on both, stellarators at IPP and the IPP development of the Helical Advanced Stellarator (HELIAS) concept[2] have played a major role in achieving the goals that have been essential for the viability of this toroidal confinement approach. This approach has been described elsewhere[3]. Here only those points will be briefly summarized which are needed as a basis for the ensuing assessment.

The HELIAS concept is a result of a stellarator optimization with respect to all criteria which are considered indispensible for good reactor conditions. In essence these are:

- High quality of vacuum-field magnetic surfaces to yield good transport properties, i.e. avoidance of major resonances and small thickness of islands.
- Good finite-β equilibrium properties to achieve a stiff configuration with rising β at fixed coil currents.
- Good MHD stability properties to achieve $<\beta> \approx 5\%$.
- Small neoclassical transport under reactor conditions.
- Small bootstrap current to minimize the ability of the plasma to affect the confinement configuration.
- Good collisionless a-particle containment at operational values of β.
- Good modular coil feasibility for coils providing a sufficiently large distance between coils and plasma.

The achieved stiffness of the magnetic configuration results from the fact that it was possible to reduce the internal plasma currents to values which are approaching the pure diamagnetic current ($< j_{||}^2/j_{\perp}^2 > \approx 1/2$). It is important to note that mutual compatibility and simultaneous achievement of all of the above criteria have been proven. The nature of the optimization result can be characterized as follows:

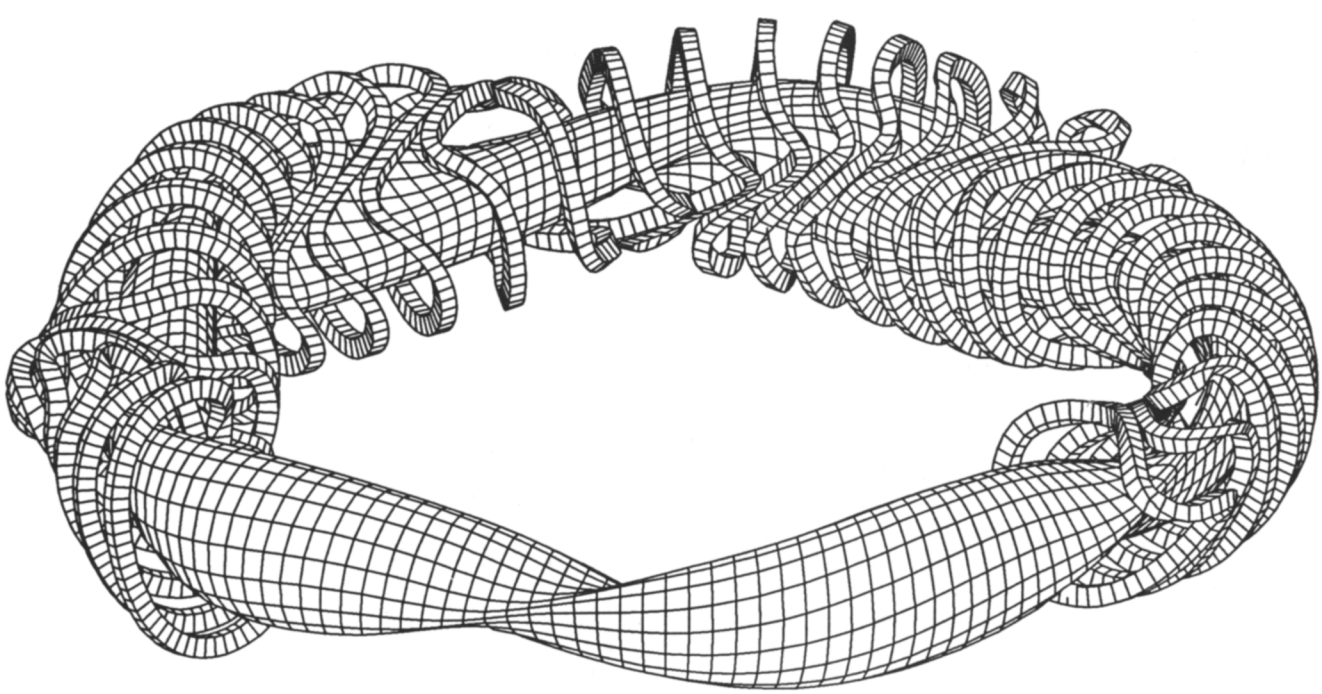

Fig. 1. Plasma and coils of Wendelstein 7-X.

The simultaneous achievement of the above set of criteria essentially determines the structure of the magnetic field strength distribution of the configuration, and its geometrical shape is then a consequence of this structure.

The resulting configuration, together with the coils producing it, is displayed in Fig. 1. It is clearly visible that the optimization makes extensive use of non-axisymmetry. In fact, most of the favourable properties are not compatible with axisymmetry. The configuration is of five-fold rotational symmetry. For fixed constraints, four-fold symmetry is not sufficient for achieving the goals quantitatively and six-fold symmetry is leading to engineering difficulties in an unnecessary fashion.

As will become evident later, the optimization according to the above criteria directly yields very desirable properties also for other properties which are essential for reactor engineering and maintenance, and also promises reduction of anomalous plasma transport.

III. A STELLARATOR REACTOR BASED ON THE HELIAS CONCEPT

A stellarator reactor has to observe a number of constraints if the properties offered by the Advanced Stellarator concept are to be optimally exploited: The magnetic field is limited to 5T in order to keep the NbTi technology available for the superconducting magnet. The configuration and the number of modular coils will be the same as for W7-X because the W7-X configuration (see later) has been designed such that by proper scaling-up the desired reactor properties would be achieved. There will also be no provision for current drive because the configuration essentially eliminates the bootstrap current and a residual one is tolerable. For given plasma pressure, the plasma temperature is selected for about maximum fusion power output. This allows high plasma density and the exploitation of the stellarator-typical increase of confinement time with increasing density together with the absence of a disruptive density limit[4]. The magnetic configuration is scaled up linearly in dimensions from R=5.5m for W7-X to R=20m. A minor radius of 1.6m has been selected.

These conditions lead to the data listed in Table I. The dimensions of blanket and shield are rough estimates resulting from the ASRA 6C stellarator reactor study[5].

Plasma parameters for this reactor are shown in Table II for two fractional concentrations of helium, 1% to simulate the start of

TABLE I
Parameters of a Helias Reactor (HSR)

Average major radius	[m]	19.5
Average coil radius	[m]	3.9
Number of coils		50
Number of field periods		5
Induction on axis	[T]	5.0
Max. induction on coils	[T]	10.7
Tot. magnetic energy	[GJ]	70
Rot. transform on axis		0.84
Rot. transform on boundary		0.97
Average plasma radius	[m]	1.6
Plasma volume	[m^3]	10^3
Surface of first wall	[m^2]	2.1×10^3
Volume of blanket (d=0.4 m, coverage 85%)	[m^3]	780
Mass of blanket (ρ=4)	[t]	3.1×10^3
Volume of shield (d=0.6m)	[m^3]	1.59×10^3
Mass of shield (ρ=5)	[t]	7.95×10^3
Volume of reactor core (Coils, cryostat, structure, etc.)	[m^3]	$\approx 10^4$
Total weight	[t]	$\approx 2.4 \times 10^4$
Winding pack		
Radial height	[m]	0.6
Lateral coil width	[m]	0.54
Max. toroidal elongation	[m]	2.0
Volume	[m^3]	9.0
Current density	MA/[m^2]	31.5
Total current	[MA]	10.4
Average force density	[MN/[m^3]	86
Max. net force	[MN]	115
Total coil volume	[m^3]	450
Total coil mass	[t]	2.8×10^3
Mass of structure	[t]	$\approx 8.4 \times 10^3$
Virial stress	[MPa]	150
Superconductor		NbTi
Temperature	[K]	1.8

burn, and 10% for a burning plasma. The data listed in Tab. II show that - for the reactor - $\langle\beta\rangle$, n, T are within the intended limits. The fusion power output is in the right range. With a value slightly above $1 MW/m^2$, the neutron wall loading is by no means excessive. The table also shows that the Lackner-Gottardi scaling (LGS) which rather well describes today's Wendelstein experiments, even without any improvement, yields a confinement time already slightly larger than the required energy confinement time, $\tau_{req.}$, whereas the LHD and the gyro-reduced Bohm (GRB) scalings indeed need up to a factor of two improvement which we however indeed expect to gain from the W7-X optimization. These three scalings have been selected because they exhibit the positive density scaling ($n^{3/5}$) found in stellarators. Fig. 2 shows how the <ß>-requirement varies with the major radius of the reactor. This figure indicates what types of modifications would be used if the target were likely to be missed by some margin.

TABLE II

Plasma Parameters of the Reactor Described in Table I

f_α [%]	1.0	10
f_{Oxygen} [%]	0.1	0.1
f_{Carbon} [%]	1.5	1.5
f_{DT} [%]	88	70
Z_{eff}	1.5	1.7
$n(0)$ [$10^{20}m^{-3}$]	3.0	4.0
$<n>$ [$10^{20}m^{-3}$]	1.33	1.77
$<n>_L$ [$10^{20}m^{-3}$]	1.93	2.57
$T(0)$ [keV]	17.0	14.0
$<T>$ [keV]	7.49	6.17
$<ß>$ [%]	4.57	4.78
E_{tot} [M]	692	725
Power output		
P_α [MW]	682	516
P_{Fusion} [GW]	3.34	2.52
$P_{Neutron}$ [GW]	2.66	2.0
Neutron wall load		
$p_{w,N}$ [MW/m^2]	1.2	0.9
Energy confinement times		
$\tau_{req.}$ [s]	1.16	1.98
τ_{LGS} [s]	1.27	2.04
τ_{LHD} [s]	0.79	1.28
τ_{GRB} [s]	0.63	1.01
$n_{DT}(0)\tau_{req.}T(0)$ [$10^{20}m^{-3}$ s keV]	52.0	78.0

IV. POSSIBLE ADVANTAGES AND DISADVANTAGES OF STELLARATOR AS COMPARED TO TOKAMAK REACTORS

The above reactor will now form the basis for the following assessment on the possible advantages or disadvantages of stellarator as compared to tokamak reactors.

The advantages of Advanced Stellarators over Tokamaks essentially rest on two simple looking but nevertheless very basic differences:

1) Except for the response plasma currents, the total confining magnetic field of Advanced Stellarators is exclusively generated by currents flowing in coils external to the plasma, and
2) the response plasma currents are dominated by the diamagnetic currents; all other plasma currents are smaller.

These simple and unimportant looking properties are not compatible with axisymmetry, and thus do not apply to Tokamaks, but they do lead to a surprisingly large number of

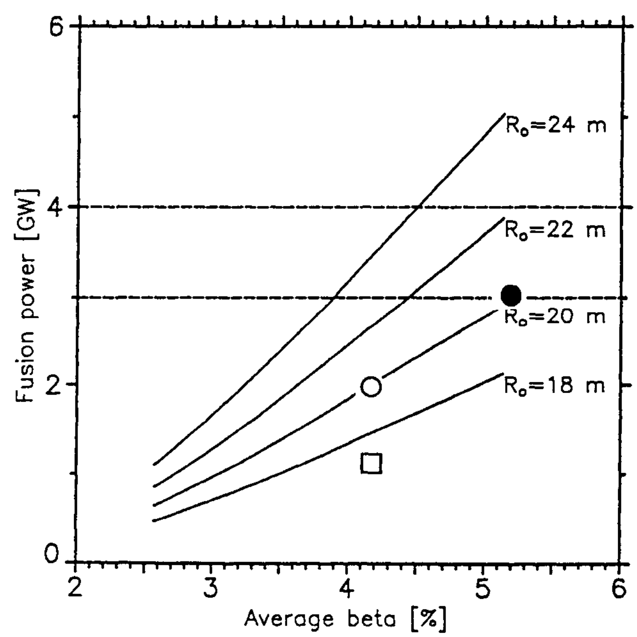

Fig. 2. Fusion power vs. <ß> for Helias reactors with R_0=18, 20, 22, 24 m. The dashed lines indicate P_{fusion}=3 GW and 4 GW. B=5 T, f_α=10 %, f_O=1%, f_C=0.1%, Z_{eff}=1.8.

- ● Helias power reactor, broad profiles, P = 3 GW
- O Helias power reactor, P = 1.8 GW
- Helias reactor, ITER plasma profiles, □ = 1.26 GW

favourable consequences for Advanced Stellarators. In essence, these are the following:

- One single coil system is sufficient.
  Since there is neither need for establishing a loop voltage nor to balance the hoop force of a time varying plasma current, the external confining magnetic field can be kept steady in time right from the beginning and thus be produced by one single coil system. This has the effect that the interaction forces between the coils are minimal, that there is no integral twist force on the coil system, that a modular coil system as used in W7-AS and envisaged for W7-X becomes possible, that the stray magnetic field is minimal and the field energy concentrated where it is needed, and that it allows a different maintenance approach.
- The plasma position is stably determined by the fixed external field.
  All field components needed for confining the plasma by their interaction with the diamagnetic currents are produced by currents flowing in external coils. For the range of β-values envisaged, this has the effect that for each value of β the plasma position is stable, without any need for position control by feed-back stabilization, and that there is a very small Shafranov shift.
- There is no net toroidal plasma current required.
  This has the consequence that there is neither need for an ohmic heating nor for a current drive system, which has the additional consequence that there is no equipment needed for driving the current, that the already heavily loaded divertor plates do not need to handle the current drive power in addition, and that it becomes possible to exploit the positive density scaling for confinement and for achieving optimal, low-temperature conditions in front of the divertor plates.
- There is no major current disruption.
  This is another consequence of the fact that there is no net toroidal plasma current and that the plasma position control is inherent. This is an important property because a possibility of major current disruptions would necessitate a particularly strong mechanical structure to prevent the danger of damage.
- The free energy of the plasma is minimized.
  This is because the plasma currents are only slightly in excess of their possible minimum, the diamagnetic currents. Thus, the driving power for instabilities is less than otherwise.
- Steady-state operation is inherent and rests upon refueling and exhaust only.
  This very important stellarator property necessitates, however, the solution of the impurity problem, which the stellarator shares with the tokamak, and both lines have concentrated considerable forces on solving this issue. In this respect it is worth noting that the optimized stellarator has available for exploitation an inherent divertor system (see later) with the total power to be handled limited to the fusion a power.
- Stellarators allow a different maintenance approach.
  This also results from the fact that only one single coil system is needed for producing the confining magnetic field and that there is no integral twist force on the magnet. In fact, each properly defined field period, except for bending forces, is mainly experiencing axial contracting forces and the whole magnet the conventional centripetal force. For a major repair, this offers the chance to radially remove whole sections of a field period without large difficulties, to carry them on rails to a maintenance hall and to maintain the blanket and shield region from either end of these sections where accessibility is large (see Figs. 3 and 4). This also allows using a more compact blanket system. Minor components like tiles or divertor plates will be maintained the same way as foreseen for Tokamaks, namely through ports without dismanteling the machine.

For a large number of issues stellarators and tokamaks possess similar properties. This is considered an advantage because it allows the transfer to stellarators of the correspondent tokamak knowledge and the application of some of the tokamak-oriented technology developments. It is for this reason that stellarators should not be considered an independent alternative line but that their development is better described by the term "concept improvement". This is so because it just so happens that the Advanced Stellarator concept offers most of the properties wanted for the needed improvements of the tokamak-based fusion reactor concepts without worsening the essential ones of the others. It is indeed very little one has to pay when going to exploit these potential advantages.

In more detail:

- Stellarators and tokamaks use the same basic confinement concept, namely toroidally closed, nested magnetic surfaces.
- The fusion power density is very similar in both systems. Both use very similar values of β and B.

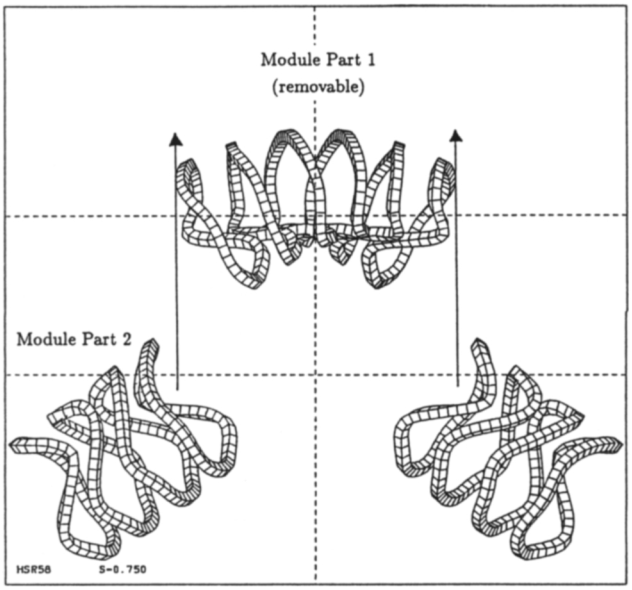

Fig. 3. Maintenance Scheme: Horizontal displacement of one module of 6 coils.

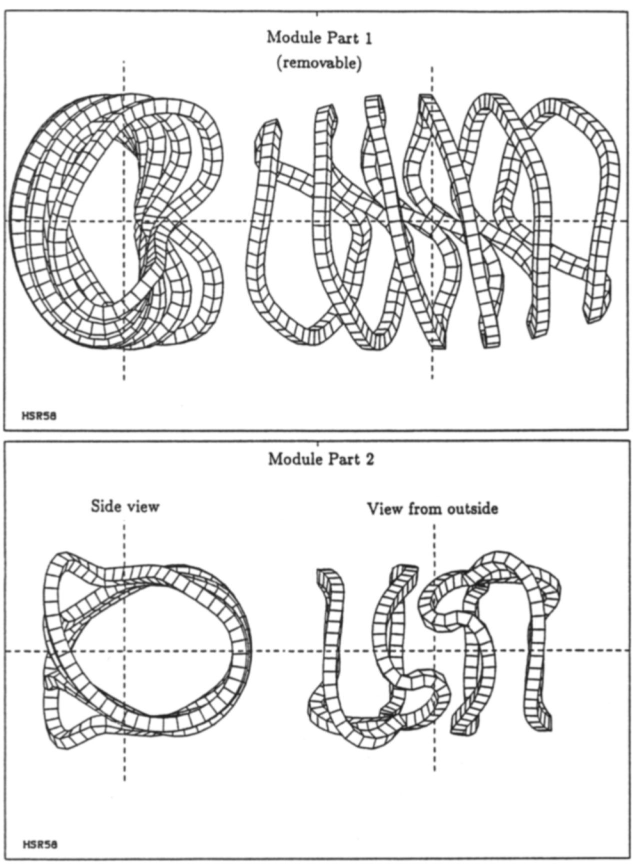

Fig. 4. Maintenance Scheme: Side view and view from the outside of the removable module (upper part), same for remaining module (lower part).

- Both systems have practically the same unit size. This follows from the fact that confinement for both systems is probably practically proportional to plasma volume so that the difference in aspect ratio has very little effect except for the wall loading density, which is somewhat smaller in stellarators, and for the blanket and shield whic'i have to be a bit more compact in stellarators. This however is alleviated by the different maintenance approach mentioned above. In fact, plasma volume, magnetic field energy and mass of blanket, shield and structure are about the same (see section V).
- As already mentioned above, power and particle exhaust are encountering very similar problems in both concepts, not counting that for stellarators the power density to be handled is smaller for two reasons, higher aspect ratio and no power needed for current drive. The basic divertor technology could be conceived to be very similar for both systems although the stellarator offers more possibilities of exploiting configurational effects (islands) because of the absence of large plasma currents and the associated higher configurational stability.

Finally, one has to assess, in which fields the Advanced Stellarator concept might be inferior to the tokamak one. Usually, under this category it is argued that the higher aspect ratio of the stellarator might lead to larger unit sizes, and that the twisted coils are more difficult to manufacture than planar ones. That the first point does not hold to be true has already been argued above and will be picked up again in Section V. For the second point the arguments go as follows:

There are additional constraints arising from the fact that the needed twist of the field has to be produced by currents flowing in outside coils. For this purpose fields of high multipolarity are required and such fields have a finite decay length. Thus, the relative distance between coils and plasma is limited. However, since only absolute distances matter, smaller devices are more difficult to design than larger ones, and as shown in Section III, for reactor sizes the needed space between plasma and coils can be made available. Nevertheless, the volume between coils and plasma should be used as economically as possible which is done by considering a compact blanket, and a lateral maintenance approach. Thus, for reactor dimensions the constraints introduced by the Advanced Stellarator concept can be met by appropriate engineering measures.

The coils for the confinement magnet can be built at sufficiently low cost, because (i) size and distribution of the forces allow costeffective winding procedures as already demonstrated by W7-AS, (ii) for large coil sizes, the coil shape is not that important anymore, (iii) there is only one single coil system needed, and (iv) the size of the single coil is significantly smaller than that needed for tokamaks.

Thus, from these two issues, one would not be inclined to deduce any negative point for stellarators. More exact conclusions would require basing the previous evaluation of systems integration and maintenance details on a blanket and shield fully designed under the above constraints. First glances look positive. In summary, and considering the present status of reactor design in general, it is very well possible that the current tokamak approaches will turn out the more difficult ones.

V. CHECK OF SOME SELECTED ITEMS

The above comparison between tokamaks and Advanced Stellarators was concentrated on the differences between the two systems. There are items, however, which have to be checked in addition for their potential to modify the above results. Among the more recently treated ones are: (A) Is there sufficient confinement for the fusion generated α-particles? (B) Is an efficient divertor compatible with the W7-X concept? (C) What is the relation between the two systems concerning their size and cost?

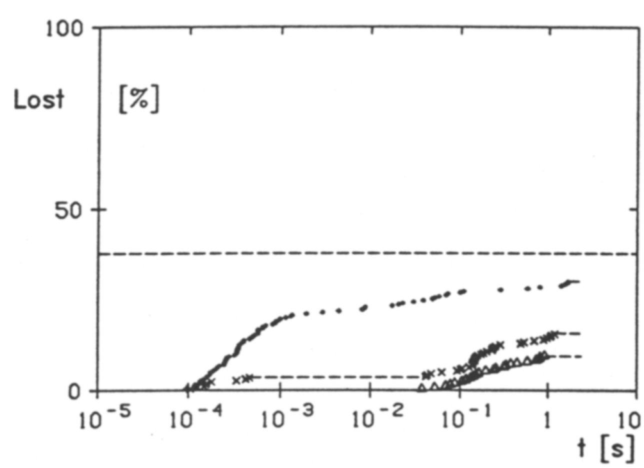

Fig. 5. α-losses in W7-X. A sample of 100 α-particles started at aspect ratio 40 and comprising the reflected particles (whose fraction of a complete sample is ≈ 35 %, see dashed line) is followed in time with the collisionless guiding centre equations.
(o) $<\beta> = 0$, (x) $<\beta> = 0.024$,
(D) $<\beta> = 0.049$ (reactor relevant case). The slowing down time is approximately 0.1 s.

A. Fast Particle Loss Fraction

In principle, three dimensional configurations like stellarators are indeed subject to fast losses of highly energetic particles. It is however a particular characteristic of the configuration optimized for W7-X that the β effect already for values smaller than the intended operating $<\beta> \approx 5\%$ is sufficient to improve the α-particle confinement in such a way that for a duration of the slowing down time the fraction of collisionless losses is reduced to not more than approximately 3%. This behaviour is shown in Fig. 5. Since a simplified calculation assumes that the particles are not losing their energy before the end of the slowing down time, the calculation is pessimistic so that losses of fast α-particles can be neglected indeed. This result is obtained for ten coils per field period.

B. The Divertor Concept

The divertor concept exploits a property arising as a consequence from the optimization procedure described in Sec. II. The optimization entails sharp edges formed in the outermost magnetic surface which are helix-like and lead - on the outboard side of each field period - from the lower to the upper ends of indented cross-sections one period apart. For the field lines in the region beyond but close to the last closed flux surface, it is always when they cross these edges that they are displaced in the radial direction. When five trough-like collector surfaces are arranged following these edges in a distance of about 1/5 plasma radius and each one made a bit longer than one field period, see Fig. 6, then all field lines are arriving on the collector surface without occurrence of leading edges[6]. Furthermore, the field lines move around the machine several times before arriving at the collector surface, thus producing a sufficiently long connection length. Within this approach there are the options of having no large islands in the region beyond the last closed flux surface and of having a chain of large 5/5 islands which - in its phase - is chosen so as to be compatible with the Helias geometry. With the assumption of an anomalous diffusion coefficient at the plasma boundary of the order of 1 m^2/s, the divertor plates - for both options - have a sufficiently large active surface and a rather smooth load distribution on an area of order 10^2 m^2. This leads to a power density of several MW/m^2 which is considered tolerable.

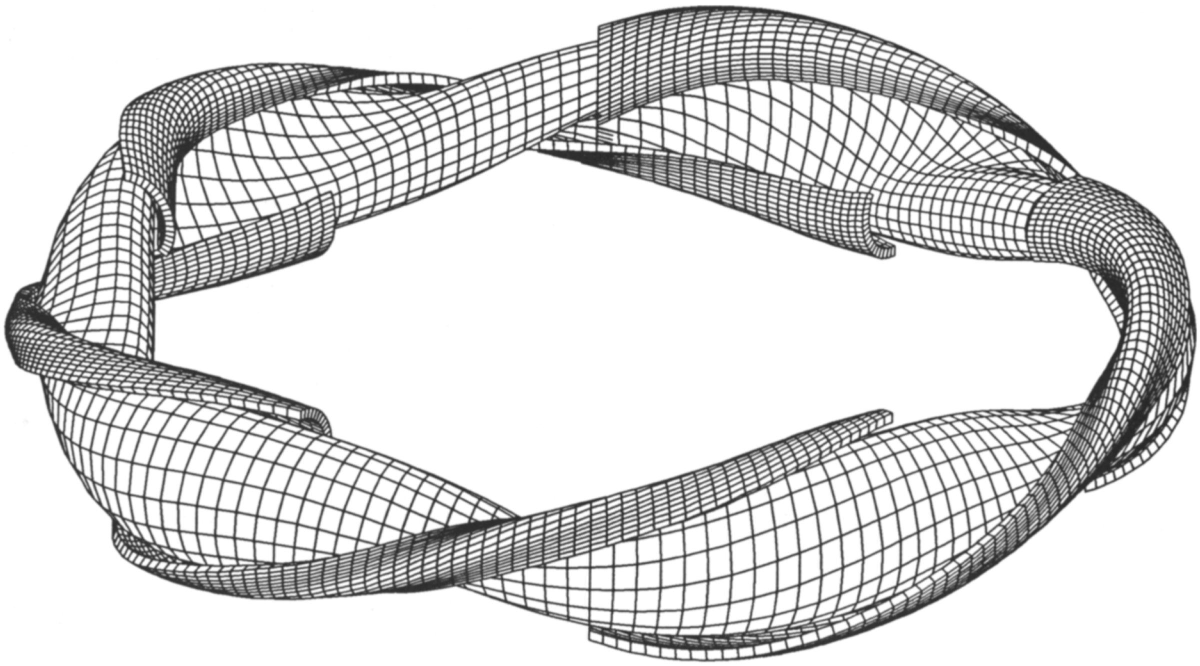

Fig. 6 Plasma and divertor troughs for Wendelstein 7-X.

The concept allows sweeping by very moderate ac magnetic fields generated by coils as shown in Fig. 7 to reduce - if necessary - peak power loads. The coils are optimized in such a way that the space for the divertor hardware and the pumping facilities is available where it is needed. It is one of the major tasks for W7-X to develop this divertor system.

C. Comparison in Size and Cost With a Tokamak Systsem

The described reactor properties of Helias reactors only hold for moderate aspect ratios which are typically near to 10. If the aspect ratio were made smaller, always some of the optimisation goals could not be reached. From Fig. 8 it is striking to see that a Helias stellarator designed for the same fusion power output as ITER would look like a belt around an ITER reactor[7]. This is a consequence of the assumption that the confinement for both systems is practically proportional to the plasma volume, and, in fact, the plasma volumes of the two systems of Fig. 8 are very close to each other.

Reactor designs of sufficient accuracy for the purpose of proper costing are still lacking. This is immediately understood in view of the still existing uncertainties about the exact parameters of future reactors. This is true for tokamaks, and even more so for stellarators. Another factor which would be difficult to handle in a comparison is the type of engineering assumed in the individual reactor designs. An earlier INTOR study had shown that from these differences also large differences in costing could result even for one and the same system. Keeping these facts in mind, and since it is not the intention to find minor differences between the two systems but rather whether there are major differences of qualitative nature, it is more appropriate to make relative comparisons of cost determining quantities as the total fusion power output, the total plasma volume, the average β, the total magnetic energy needed in the magnetic circuit, mass of the reactor and so forth. These quantities are good measures for the cost of the system, for the efficiency with which the engineering is used etc.

Therefore, from the Report on the ITER Conceptual Design[7] and from the Report of the EEF Study group[8], some relevant parameters of ITER and a tokamak reactor design are extracted and together with those of HSR, the stellarator reactor based on the W7-X configuration as described in Section III, are collected in Table III for comparison. For HSR, fractional concentrations of 10% helium. 1.5% carbon and 0.1% oxygen are considered.

TABLE III
Characteristic Data of Various Reactor Designs

	HSR	ITER	PCSR-E
Average major radius [m]	19.5	6.0	9.3
Average plasma radius.[m]	1.6	2.9	3.1
$n_{DT}(0)$ [$10^{20}m^{-3}$]	2.8	1.3	≈3
T(0) [keV]	15	17	≈20
<ß>%	5.1	4.2	3.8
I(plasma) [MA]	0	22	17
B(0) [T]	5.0	4.9	6.4
V(plasma) m^3	1000	1100	2000
Fusion power[GW]	2.9	1.1	3.6
Average coil radius [m]	3.9	5.6	≈5.9
Average coil volume [m^3]	9.0	11.3	≈18.7
Current density [MA/m^2]	31.5	35	35
Total volume (winding pack)			
TF coils [m^3]	450	180	410
PF+OH coils [m^3]	0	320	≈600
Number of coils	50	30	38
Mass of coil system [t]	11×10^3	9.8×10^3	17.4×10^3
Mass (blanket, shield) [t]	$\approx 11 \times 10^3$	$\approx 8 \times 10^3$	$\approx 7 \times 10^3$
Vacuum vessel [t]	$\approx 2 \times 10^3$	$\approx 8 \times 10^3$	$\approx 2 \times 10^4$
Total mass [t]	2.4×10^4	2.5×10^4	4.4×10^4
Field on axis [T]	5.0	4.85	6.36
Max. field on coils [T]	10.7	11.4	11.3
Magnetic energy (TF) [GJ]	70	40	115
Magnetic energy (PF+OH) [GJ]	0	23	24
Magnetic energy (Total) [GJ]	70	63	139

HSR relies on the expected confinement time improvement of approximately a factor of two at Z_{eff}=1.3. The ITER parameters are taken from the conceptual design report (1990). PCSR-E is a reactor which extrapolates NET/ITER physics and technology to reactor dimensions and thus contains a number of safety factors to cover the still existing uncertainties in the predictions. These safety factors will not be needed anymore once the results from the next steps are available. The assumptions about maturity made for HSR are similar to those for PCSR-E.

Then the essential result is that the total fusion powers are very similar for the two reactors, HSR and PCSR-E, but it is interesting to see that both the total magnetic energy needed to produce this power and the reactor mass are considerably smaller for HSR. The reason for this result is that in HSR the magnetic energy, which is a good measure for comparing reactor costs, is better concentrated on the plasma volume, and, thus, is only used for confinement purposes. This has similar effects on the reactor mass which involves the forces which have to be balanced.

Comparing other information not given in the table, one finds that the volume for blanket and shield in HSR is only slightly larger than that for the tokamak reactor although the aspect ratio is rather different. To assess this further would require much more information on the used technology. The point was already made above that the better access in HSR allows the design of more compact blankets.

Thus, in essence, these comparisons show that the cost for stellarator reactors of the Helias type or for tokamak reactors are expected to be similar, and if there is a diffe-

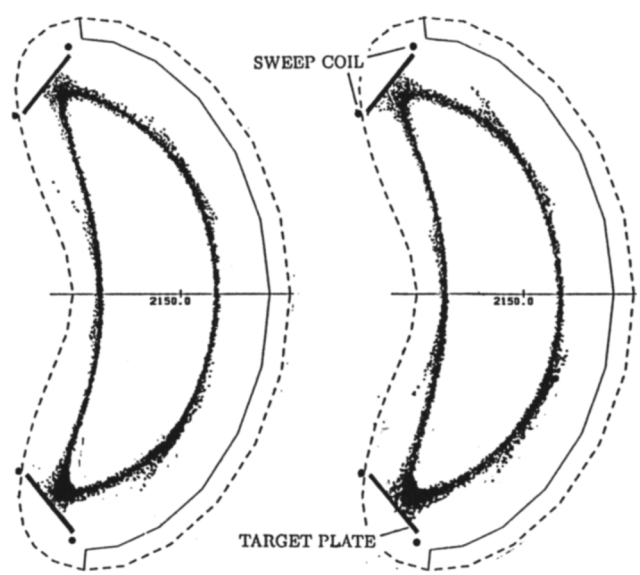

Fig. 7. X-point divertor sweeping. Monte-Carlo calculation of particle orbits in the boundary layer. Anomalous diffusion coefficient D_{an} = 2 m^2/s.
AC-current in the sweep coils: 40 kA. The two figures demonstrate the displacement of the peak load position on the target plates by changing the sign of the current in the sweep coils.

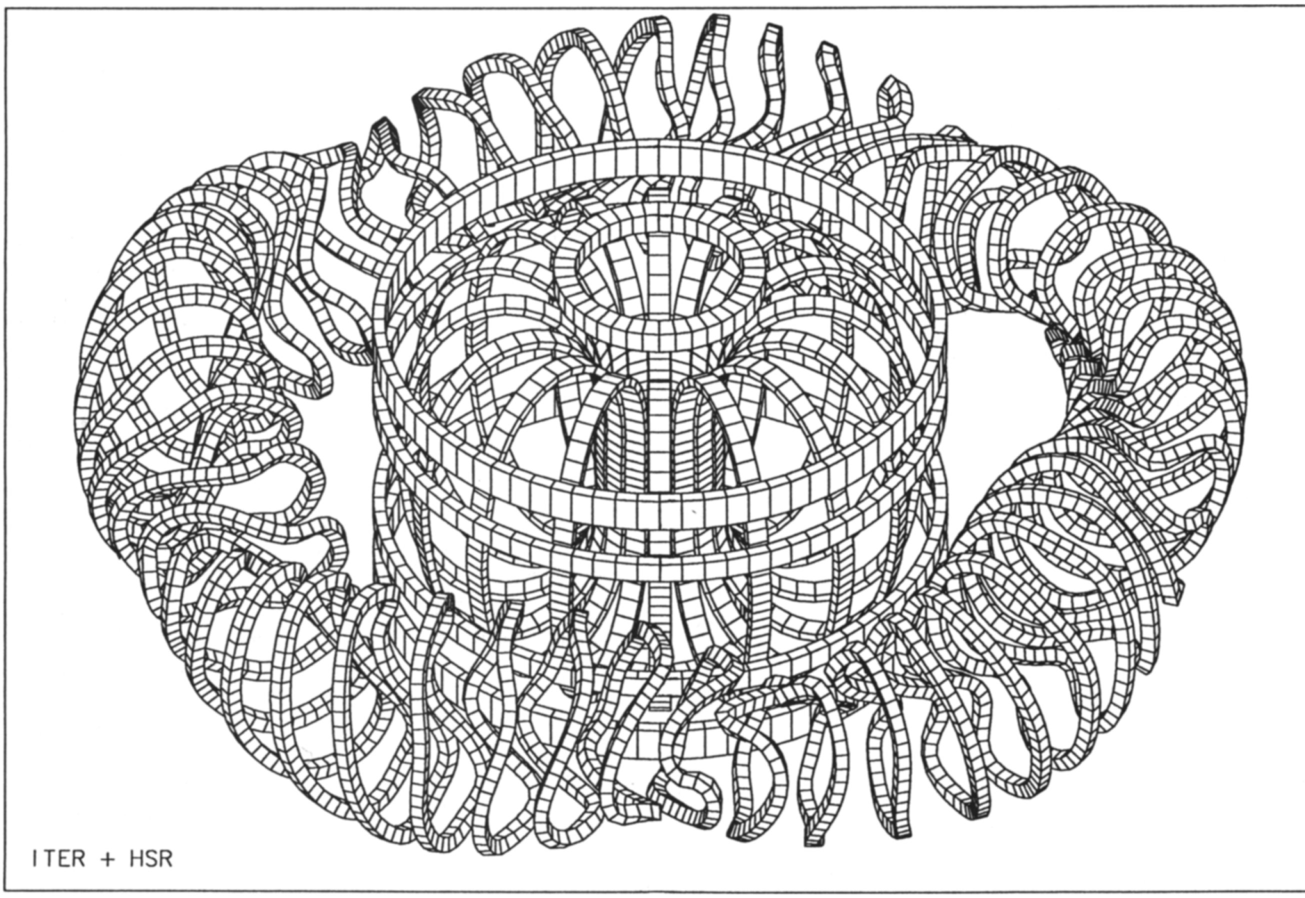

Fig. 8. View on the coil systems of ITER (at the centre) and HSR (surrounding ITER)

rence, the stellarator reactor should be less costly. This result has not yet considered that the somewhat reduced wall loading in stellarator reactors would lead to a longer life time of the blanket components and divertor plates.

VI. WENDELSTEIN 7-X

The task of the W7-X experiment is to provide an integrated concept test which is needed for producing convincing predictions on the properties of ignited plasmas in Advanced Stellarators. Its major parameters are given in Table IV, its basic configuration in Fig. 1. A careful discussion comes to the conclusion that a W7-X device of R=5.5m and B=2.5T is about the minimum in size compatible with divertor conditions and thus just combines the possibilities of good optimization of the stellarator configuration with reasonable space for successful divertor performance, which is indispensible for creating elevated plasma parameters.

The next question is how conclusive the extrapolation from W7-X to a burning plasma device will be. Here the arguments go as follows: The two configurations are identical, only the strength of B is doubled. The value of β needed for the reactor should be accessible in W7-X, though at higher collisionality. This restriction, however, cannot be considered as serious. As to the other two nondimensional parameters, normalized mean free path and plasma radius to ion gyro radius ratio, for the above reactor parameters, $L^* = \Lambda\iota/\pi R \approx 100$ and $Q_\rho = a/\rho_i \approx 350$. Plasma parameters of $T \approx 3$keV, $n \approx 0.6 \cdot 10^{20} m^{-3}$, which are possible to be established within the ECRH scenario in W7-X, lead to $L^* \approx 100$ so that the reactor collisionality is well accessible. As to the value of Q_ρ, this is best approximated in a hydrogen plasma: for $T_i \approx 1$keV $\longrightarrow Q_\rho \approx 300$ (and, if $n = 0.5 \cdot 10^{20} m^{-3} \longrightarrow L \approx 13$). The dependencies of the scalings discussed in Sec. III on plasma radius and on major radius are represented by the plasma volume in very good approximation. With this assumption the fusion product $n \cdot \tau \cdot T$ scales with $\beta \cdot Q_\rho{}^3 \cdot B$ so that Q_ρ is the most important scaling parameter.

TABLE IV
Characteristic Data of the Experiment Wendelstein 7-X

Rotational transform, ι, on axis/boundary	0.84/ 0.99
Variation of ι	$\pm$ 0.2
Variation of shear	$\pm$ 0.1
Variation of mirror field	0.1
Pfirsch-Schlüter currents $< j_{\parallel}^2/j_{\perp}^2 >$	0.5
Magnetic well depth	0.01
MHD stability limit, $< \beta >_{st}$	0.043
Equivalent ripple, δ_e	0.015
Ratio of bootstrap currents, $J_{BS,stel}/J_{BS,tok}$	$\leq$ 0.1
Average major radius, R_o	5.5 m
Average plasma radius, r_a	0.53 m
Average coil radius, r_c	1.14 m
Min.distance plasma-coils, Δ_{pc}	0.29 m
Min.distance plasma-wall, Δ_{pw}	0.12 m
Induction on axis, B_o	3.0 T
Max. induction at coils, B_m	6.1 T
Total magnetic energy, W_m	600 MJ
Max. net force (one coil), F_{res}	3.6 MN

The prediction on the achievable value of $n{\cdot}\tau{\cdot}T$ [$10^{20}m^{-3}$ s keV] in W7-X depends on both the scaling law and the transport improvement assumed and ranges from 3 (with the assumption of losses at a level of 1/8 of those given by the Lackner-Gottardi scaling) to 0.5 (gyro-reduced Bohm scaling). Depending on the results of W7-X the extrapolation to reactor parameters will therefore be in the range of 10 if the effects of the optimization take place as expected, or in the range of 100 if one is pessimistic. Therefore, this range of $n{\cdot}\tau{\cdot}T$ values is a good range for W 7-X to explore.

The largest extrapolation factor appears to occur in the magnetic energy, namely by two orders of magnitude. But this step size can be accepted because it is mainly in engineering, and there is neither a change in the magnetic configuration nor a change in the basic technologies to be applied. Furthermore, prototypes of full size will give the necessary assurance that the engineering will work.

Since W7-X will not be operated with DT there is no direct study possible on α-particle physics. α-particle confinement can be simulated on W7-X though by injecting high energy particles. Further information will come from DT operated tokamaks which will yield information on collective effects which are not considered too dangerous at present.

As far as impurities and divertor action are concerned, these items will be studied extensively in W7-X and the means be optimized. It is argued that the flux pattern will stay the same when going from W7-X to the reactor, and that the flux densities and energies tend to approach reactor levels at the highest heating powers envisaged for W7-X. To handle these fluxes at the divertor plates, also the results of tokamaks with reactor-grade scrape-off fluxes, like ASDEX-upgrade and JET, and the R&D work for NET/ITER-like devices will be exploited to arrive at a design feasible for reactor operation. Once the fluxes and their distribution are known, this procedure should be adequate.

It thus follows that the step size from W7-X to a reactor grade plasma is acceptable. Many of the properties can be simulated rather well in W7-X. In addition, when the question of extrapolation will arise for stellarators, the situation will be different from the present one for tokamaks because by then experience from large-scale tokamaks will be available and quite a number of items will also hold for stellarators.

In summary, the extrapolation is not larger than by one order of magnitude. For the magnetic energy, however, there are two orders of magnitude, but considering the test of prototypes, this step size is argued to be acceptable.

VII. SUMMARY AND CONCLUSIONS

A stellarator reactor based on the Helias concept has been discussed. This essentially consists in a linear enlargement of Wendelstein 7-X by a factor of approximately 3 together with an increase of the magnetic field by a factor of 2. Reactor volume, mass, and magnetic field energy are comparable with those of a tokamak reactor. While decisive advantages can easily be identified, no definite disadvantages of the stellarator approach have been found. The Wendelstein 7-X experiment will provide an integrated concept test of the Advanced Stellarator approach. Extrapolation from W7-X to an ignited stellarator is large but plausible. In summary, the potential of the Advanced Stellarator approach for concept improvement in magnetic fusion is large.

REFERENCES

1. L. SPITZER, "The Stellarator Concept", *Phys. Fluids*, 1, 253 (1958).

2. J. NÜHRENBERG, R. ZILLE, "Stable Stellarators with Medium β and Aspect Ratio", *Phys. Letters*, **114**, 129 (1986).

3. W. LOTZ, J. NÜHRENBERG, C. SCHWAB, "Optimization, MHD Mode and a-Particle Confinement Behaviour of Helias Equilibria," *Proc. 13th Int. Conf. on Plasma Physics and Contr. Nuclear Fusion Research*, Washington, USA, 1990, IAEA-CN-53/C-III-5, IAEA, Vienna (1991), **II**, 603.

4. H. RENNER, U. GASPARINO, H. MAAßBERG, et al., "Confinement Properties of the 'Advanced Stellarator' WENDELSTEIN W VII-AS", *Proc. 13th Int. Conf. on Plasma Physics and Contr. Nucl. Fusion Research*, Washington, USA, 1990, IAEA-CN-53/C-I-2, IAEA, Vienna (1991), **II**, 439.

5. G. GRIEGER, E. HARMEYER, J. KIßLINGER, F. RAU, H. WOBIG, "Advanced Stellarator and Burner Studies in Fusion Reactor Design and Technology", *Proc. 4th IAEA, Tech. Committee Meeting*, Yalta, USSR, 1986, IAEA, Vienna 1986, **II**, 341

6. E. STRUMBERGER, "Magnetic Field Line Diversion in Helias Stellarator Configurations "Perspectives for Divertor Operation," *Proc. 18th Europ. Conf. on Contr. Fusion and Plasma Physics*, Berlin, Germany, ECA, 1991, 15C, Part II, 173, and *Nucl. Fusion* (1992), in press.

7. K. TOMABECHI J.R. GILLELAND, Yu. A. SOKOLOV, R. TOSCHI, "ITER Conceptual Design," *Nuclear Fusion*, **31**, 6, 1135 (1991).

8. R.S. PEASE, J. DARVAS, R.H.,FLOWERS, L. GOUNI, G. GRIEGER, K. KÖBERLEIN, A. RONCAGLIA, "Environmental, Safety-Related and Economic Potential of Fusion Power - Main Report by the EEF Study Group," Progressive Engineering Consultants Ltd., 105 Walton Road, Warrington, U.K. WA4 6NR, United Kingdon Brussels, (1989).