stellarator news

Published by Fusion Energy Division, Oak Ridge National Laboratory
Building 9201-2; P.O. Box 2009; Oak Ridge, TN 37831-8071, USA
Editor: James A. Rome
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Issue #22 July, 1992

Special issue on Stellarator Reactors

The many advances in the art and science of stellarator design have led to correspondingly better near-term stellarator experiments and to proposals for stellarator reactors that solve many of the problems inherent in tokamak reactors. This issue of *Stellarator News* presents summaries of two important invited stellarator reactor talks that were presented at the *Tenth Topical Meeting on the Technology of Fusion Energy* of the American Nuclear Society that was held June 7–12 in Boston.

Assessment of torsatrons as reactors

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Stellarators have significant operational advantages over tokamaks as ignited steady-state reactors because stellarators have no dangerous disruptions, no need for continuous current drive and power recirculated to the plasma, less severe constraints on the plasma parameters and profiles, and access from the inboard (small-*R*) side for easier maintenance. The torsatron type of stellarator configuration can have helical divertors outside the windings to reduce the power density on the divertor plates and a near-perpendicular loss region to eliminate helium ash accumulation, at the expense of a reduction in alpha-particle heating. A new ORNL study shows that torsatron reactors could also have double the mass utilization efficiency and a significantly lower cost of electricity (COE) than conventional tokamak reactors.

Reactor optimization approach and assumptions

The recent extensive ARIES studies [1,2] have explored the potential for improving the attractiveness of tokamak reactors. Our study applies the ARIES-I costing and component assumptions to optimization of stellarator reactors. This approach allows a more accurate relative comparison of stellarators and tokamaks as reactors

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Modular stellarator reactors and plans for Wendelstein 7-X

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With the start of construction of the experimental fusion reactor ITER getting closer, it is now becoming urgent to check whether after ITER the presently followed tokamak approach will 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 such as unit size, power and particle exhaust, pulse length, power density, power loadings, plasma stability, technical requirements, maintenance approach, costing, etc.

We argue that the Advanced Stellarator has indeed the potential to offer a set of more desirable reactor properties than the tokamak system. In most respects, the stellarator is very similar to the tokamak and shares with the tokamak most of its beneficial properties. Therefore, the comparison is made by concentrating on the differences between the two systems. In this way the comparison becomes relative and is thus much more conclusive than a point-by-point comparison between two independently developed and designed reactors.

The stellarator shares with the tokamak the basic concept of nested magnetic surfaces to achieve confine-

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and is possible because of their general similarity. The models for the detailed reactor engineering and reactor performance constraints, however, are specialized for the example of a particular torsatron reactor because of specific differences between different types of stellarators and between stellarators and tokamaks.

ARIES-I Benchmark. We recalculated the ARIES-I costs using the ARIES-I parameters to benchmark our calculations and to provide a more detailed comparison with our torsatron reactor calculations. The values obtained for our 'ARIES-I' are very close (a few tenths of a percent) to a May 1992 recalculation of ARIES-I. The slight differences arise from the different approximations used in calculating the masses of the blankets and shields.

Reactor Component Assumptions. The thicknesses, compositions, average mass densities, and unit costs for the first-wall/blanket/reflector assembly and the neutron shielding are the same as those for ARIES-I. However, the thickness of the blanket directly under the helical windings on the inboard side of the torus depends on the particular case studied. The superconducting winding pack is assumed to have a rectangular cross section with toroidal elongation k. The maximum allowable current density j_{max} in the helical winding is calculated using the same relation as in the ARIES-I studies. The maximum magnetic field on the superconductor, B_{max}, is calculated from an expression that gives an excellent fit over a wide range of k to results obtained using the finite-element code MAGFOR with accurate helical winding trajectories. The divertor area required is calculated by dividing the total power to the divertor by 3 $MW{\cdot}m^{-2}$, rather than assuming that the divertor plate area is 15% of the first-wall area as in the ARIES-I study. The other engineering and materials assumptions are the same as those for ARIES-I.

Reference Configuration Assumptions. The reference stellarator configuration chosen was a Compact Torsatron [3] with six toroidal field periods (CT6). It is not an 'optimum" stellarator configuration, only one of the family of Compact Torsatron configurations obtained by maximizing the average radius of the last closed magnetic surface subject to constraints that maximize the beta limit.

The relatively open coil geometry allows access between the helical windings for blanket removal and maintenance without disassembly of the reactor core. The thermal particles and alpha particles exit from the plasma in a thin helical strip between the helical windings. Although the space between the plasma edge and the helical winding on the outboard (large-R) half of the torus is more than adequate, that on the inboard half of the torus is much smaller. Because the minimum value possible for the major radius R_0 is proportional to the distance d between the plasma edge and the center of the coil winding pack, minimizing d (in particular, the radial depth of the winding pack and the blanket thickness under the inboard half of the helical winding) is important in reducing the cost of torsatron reactors.

Reference Transport Assumptions. A global energy confinement time τ_E is used to calculate the conduction power, rather than assuming that the conduction power is a fixed fraction of the power in the plasma as in the ARIES-I studies. Because different energy confinement scalings fit present stellarator data but scale differently to reactors, we choose the dimensionally correct Lackner-Gottardi scaling, which fits both tokamak and stellarator data, with a confinement improvement factor H' similar to the H-mode confinement improvement factor for tokamaks. Evidence from experiments in ATF and W VII-AS and theoretical arguments support such a confinement improvement. However, the usual factor of 1.3 improvement with ion mass is not assumed in our study. The density and temperature profiles are the same as those assumed in the ARIES-I studies.

Because the relatively large helical ripple in torsatrons, combined with symmetry-breaking toroidal effects, can lead to a near-perpendicular loss region for energetic particles, we assume that all helically and toroidally trapped alpha particles are lost and calculate the additional energy lost by pitch-angle scattering into the loss region during the slowing-down process. The combined loss can reduce the effective alpha-particle heating by up to ~ 40%. The majority of alpha particles that are not born in the loss region transfer their energy to the background plasma until they slow down from 3.5 MeV to an energy $E \approx 30\, T_e \sim 0.3$ MeV, below which they rapidly scatter into the loss region. This loss prevents accumulation of helium ash in the plasma and the attendant dilution of the fuel ions.

Selection of Reference Assumptions. The two main stellarator-specific assumptions are the confinement improvement factor H' and the thickness b_i of the blanket under the inboard half of the helical winding. $H' = 1.75$ (= 1.35 with the mass correction) was chosen as representing a reasonable target for stellarator confinement improvement; similar or better confinement improvement factors have been obtained in tokamaks. 'ARIES-I' requires $H' = 1.6$. $b_i = 0$ was chosen because the inboard half of the helical winding typically covers only 8–12% of the area available for the tritium breeding blanket; the small decrease in the global tritium breeding ratio can be compensated by increasing the local tritium breeding ratio by a corresponding amount. The toroidal elongation k of the helical winding was chosen to be 5 as a compromise between higher values of k (a smaller coil depth that leads to a smaller R_0 and a lower COE) and lower values of k (more room for blankets between the helical windings on the inboard side). The level of safety assurance (LSA) factors appropriate to the low-activation materials assumed are the same as those assumed for ARIES-I (LSA = 2).

Comparison of the Reference CT6 Case with 'ARIES-I'

The main device and plasma parameters for the CT6 reference case are compared with those for 'ARIES-I' in Table I. The CT6 case has a slightly smaller major radius than 'ARIES-I' and 31% smaller plasma volume. The field on axis for CT6 is 55% of that in 'ARIES-I', and the beta is 2.8 times higher. The density-averaged temperature $\langle T \rangle$ is only one-third that in 'ARIES-I' and the volume-average density $\langle n \rangle$ is correspondingly higher. The neutron wall loading is only 11% higher than in 'ARIES-I'. The maximum field on the helical winding is three-quarters that on the toroidal field (TF) coil in 'ARIES-I', and the total stored magnetic energy is 38% of that in 'ARIES-I'. The total mass of the fusion power core is 4,130 tonnes, vs 10,067 tonnes for 'ARIES-I'. The mass utilization efficiency for CT6 is 2.4 times that of 'ARIES-I', which results in a 25% lower unit direct cost and a 20% lower COE for the CT6 reactor. As in the ARIES reactor studies, the costs assume the 'learning curve' credits of about 50% associated with a 'tenth-of-a-kind' production reactor.

There are two main reasons for the lower costs for CT6: the absence of current drive and the smaller magnet mass. Because no power is needed for current drive in CT6, the recirculating power fraction is only 9% (vs 20% for 'ARIES-I') and the cost of the supplemental heating system (only needed for plasma startup) is much lower. The helical winding mass is much less than the coil mass in 'ARIES-I' because the CT-6 coil perimeter is smaller (the CT6 winding is closer to the plasma) and the winding cross section is much smaller (less total ampere-turns and higher average current density because of the lower magnetic field).

The total reactor equipment cost is $767 million (vs $1289 million for 'ARIES-I'), and the total reactor plant equipment cost is $1089 million (vs $1628 million for 'ARIES-I'). The geometry-dependent fusion power core components (blanket and first wall, divertor, shields, magnets, vacuum vessel, and primary structure) are only 40% of the $1744 million total direct cost for CT6 and 49% of the $2318 million total direct cost for 'ARIES-I'. Components that depend on the thermal and

Table I. Main plasma and device parameters for CT6 torsatron reactors and 'ARIES-I'

Quantity	Ref. CT6	B_{max}= 12 T CT6	'ARIES-I'
Net electric output (MW)	1000	1000	1000
Major radius R_0 (m)	6.34	6.56	6.75
Ave. plasma radius a_p (m)	1.64	1.72	1.95
Plasma volume (m^3)	335	388	489
Toroidal field on axis (T)	5.83	5.61	10.56
Maximum field on coils B_{max} (T)	15.2	12.0	19.9
Electron density $\langle n \rangle$ (10^{20} m^{-3})	3.49	3.20	1.43
Plasma temperature $\langle T \rangle$ (keV)	6.77	6.84	20.0
Central ion temperature T_0 (keV)	12.8	13.0	39.0
Volume-average toroidal beta (%)	5.38	5.37	1.9
Neutron wall loading (MW/m^2)	3.03	2.75	2.72
Mass utilization efficiency (kW/tonne)	242	197	99.3
Unit direct cost ($/kW)[a]	1744	1846	2318
Cost of electricity (mills/kWh)[a]	65.0	67.8	81.4

[a] In constant 1990 dollars.

electrical power make up the remainder. The total capital cost ($3365 million for CT6 and $4473 million for 'ARIES-I') is almost twice the total direct cost for both because of financial charges that are proportional to the total direct cost and depend on the LSA credits assumed.

Dependence on Blanket and Confinement Assumptions. The COE vs the confinement improvement factor H' is shown in Fig. 1 for two different values of the blanket thickness b_i under the inboard HF winding. Adding a 0.335-m-thick blanket, half the thickness used in ARIES-I but thicker than the beryllium-rich blankets used in earlier reactor studies, increases the COE to a level that is closer to the 'ARIES-I' value. The COE is less than that for 'ARIES-I' for $H' > 1.2$ ($b_i = 0$) and $H' > 1.4$ ($b_i = 0.335$ m). Improved energy confinement is reflected most strongly in a factor of 7 increase in $\langle \beta \rangle$, from 1.1% to

7.6%, and a factor of 5.3 decrease in the plasma volume, from 1662 m^3 to 312 m^3, as H' increases from 1 to 2.

The factor of 1.5 confinement improvement needed for an attractive reactor is similar to the L-to-H confinement improvement factor in tokamaks. For the reference $b_i = 0$ assumption, the mass of the fusion power core decreases from 17,157 tonnes at $H' = 1$ to 3,772 tonnes at $H' = 2$. The rapid reduction in the mass with increasing H' is due to smaller masses for the blankets and shields ($\propto R_0^2$) and for the coil systems. The helical and VF coil masses decrease with increasing H' due to both the decreasing coil lengths ($\propto R_0$, which drops from 10.8 m to 6.2 m) and the decreasing coil cross sections (area $\propto B_0 R_0/j$, where B_0 decreases from 8.7 T to 5 T and j increases from 28 MA/m^2 to 87 MA/m^2).

Sensitivity to Coil Assumptions. A nominal set of coil parameters was chosen for the base reference case. The most critical parameters (j/j_{max}, k, and B_{max}) were varied over representative ranges to test the sensitivity of the reactor parameters to those assumptions: j/j_{max} from 0.5 to 1, B_{max} from 8 T to 16 T, and k from 1 to 11 (where there was no space between windings on the inboard side, as in a tokamak). Decreasing j/j_{max} to 0.5 only increased the COE by 4.9%. Decreasing k to 1 (a square coil cross section) led to a larger (19%) change in the COE. The reference CT6 case had $B_{max} = 15.2$ T. Lower values of B_{max} are possible at somewhat increased values of the COE, as shown in Fig. 2. The case with $B_{max} = 12$ T, 60% of that for 'ARIES-I', and a COE of 67.8 mill/kW(e)h, 17% less than 'ARIES-I', is also attractive and is summarized in Table I. The reactor costs are relatively insensitive to additional costs related to fabrication of more complex coils; increasing the cost of the helical and VF coils by a factor of 3 only increases the COE by 3%. The reactor costs are more sensitive to additional costs associated with the blankets and shields; increasing the costs of blankets, shields, and coils by a factor of 1.5 increases the COE by almost 20%. However, none of these variations (except $B_{max} < 9$ T) leads to a COE as high as that for 'ARIES-I' (25% higher than the reference CT6 case).

Sensitivity to Other Parameter Assumptions. A nominal set of physics parameters was chosen for the base reference case. These parameters were varied over representative ranges to test the sensitivity of the reactor parameters to those assumptions: the fraction of trapped alpha particles lost from 0 to 100%; the fraction of oxygen impurities from 0.5% to 1.5%; the wall reflectivity from 50% to 90%; and the shapes of the density and temperature profiles. Several of these variations (e.g., more peaked density and temperature profiles, higher wall reflectivity, less oxygen impurity content, and no alpha-particle losses) lowered the COE, but none reduced the COE by more than 2.5%. Broader density and temperature profiles, higher oxygen impurity content (1.5%), and 25% reduction in the maximum allowable current density in the coils increased the COE, but not by more than about 3%. Without any low-activation materials credits (LSA = 4), the COE increases to 74.9 mill/kW(e)h. Although power plants with $P_E < 1$ GW(e) are possible, better reactor economics are obtained for larger power plants: increasing P_E to 1.5 GW(e) and 2 GW(e) reduces the COE by 17% and 25%, respectively.

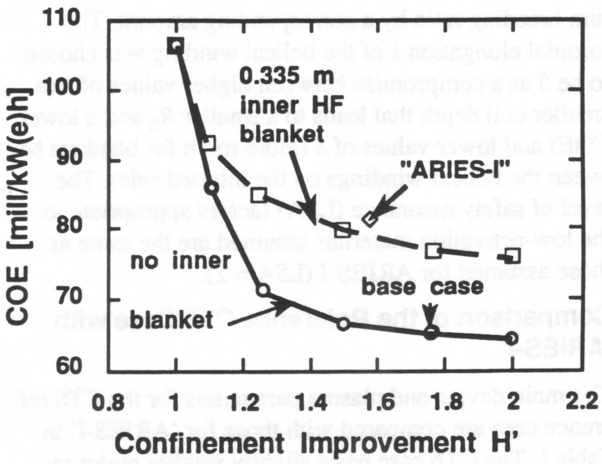

Fig. 1. Dependence of the cost of electricity on the confinement improvement factor H' for two different inboard blanket thicknesses.

Effect of Coil Modularization. Although a continuous helical winding was used in most of the examples in

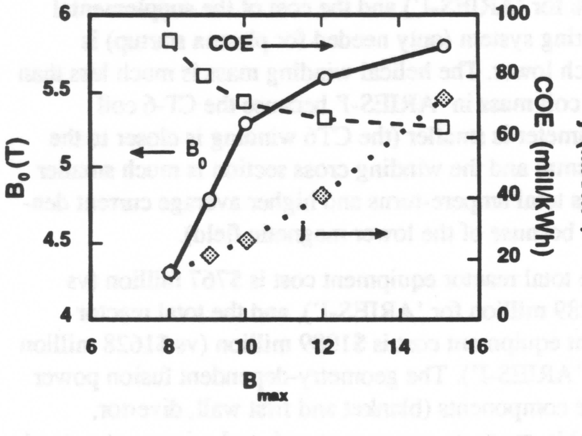

Fig. 2. Dependence of reactor parameters on the maximum magnetic field on the superconducting coils.

this study, these coil configurations can be modularized to have one (or two) coils per toroidal field period (a 'symmotron') or five to ten nonplanar coils per field period (the 'Garching' modularization scheme) because it may be necessary to replace a coil, even if the coils are not as stressed as in a tokamak reactor. The 'symmotron' modularization is simulated by adding VF coil segments that connect the beginning and end of each field period, and carry the full helical coil current, and extra VF coils to give the correct net VF coil currents. The resulting COE is 65.4 mill/kW(e)h, only 0.6% higher than for the reference CT6 case. Modularization using several nonplanar TF coils per field period is simulated by reducing the effective helical extent of the coils by a factor of two and eliminating the VF coils. Because there is no longer space on the inboard half of the torus for the blanket segments, the full blanket thickness (0.67 m) must be used under the nonplanar TF coils on the inboard side. As a result, the major radius increases to 10.9 m and the COE to 91.2 mill/kW(e)h.

Conclusions

Torsatron reactors could be competitive with tokamak reactors for a range of assumptions. The 20% lower COE for the reference CT6 case vs 'ARIES-I' allows relaxing different assumptions and constraints while still keeping the COE less than that for 'ARIES-I'. Although present stellarators are developing much of the physics basis needed for stellarator optimization, improvements in the reactor embodiment are still needed. The stellarator reactor optimization code developed for this study will be used to examine other stellarator configurations using more refined 1-D stellarator transport models that include self-consistent calculations of the radial profiles of the ion and electron temperatures and densities and the ambipolar radial electric field.

1. F. Najmabadi, R.W. Conn, and the ARIES Team, 'The ARIES-I Tokamak Reactor Study', Fusion Technol., 19, 783 (1991).
2. R.L. Miller, R.A. Krakowski, and the ARIES Team, 'Options and Optimizations for Tokamak Reactors: ARIES', Fusion Technol., 19, 802 (1991).
3. J.F. Lyon, B.A. Carreras, V.E. Lynch, J.S. Tolliver, and I.N. Sviatoslavsky, 'Compact Torsatron Reactors', Fusion Technol., 15, 1401 (1989).

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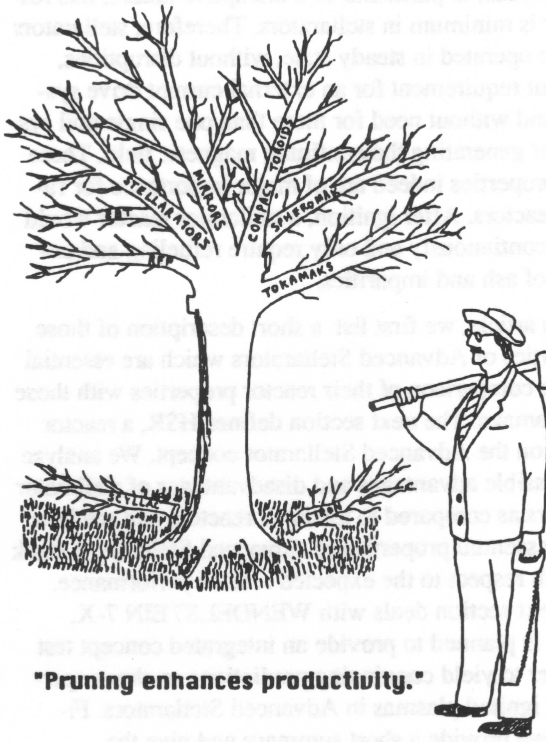
Response to the editorial in the May, 1992 *Stellarator News*



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ment. A net toroidal plasma current is needed in tokamaks, but is not required in stellarators. Since a net toroidal current provides free magnetic energy to drive instabilities, in particular of a disruptive nature, this reservoir is minimum in stellarators. Therefore, stellarators can be operated in steady state, without disruptions, without requirement for an external current drive system, and without need for more than one single coil system for generating the confining magnetic field. These four properties indeed are of major importance for fusion reactors. After ignition, a stellarator reactor would work continuously and only require refueling and exhaust of ash and impurities.

In this article, we first list a short description of those properties of Advanced Stellarators which are essential for the comparison of their reactor properties with those of tokamaks. The next section defines HSR, a reactor based on the Advanced Stellarator concept. We analyze the possible advantages and disadvantages of stellarator reactors as compared to tokamak reactors and see how some essential properties of Advanced Stellarators stack up with respect to the expected reactor performance. The next section deals with WENDELSTEIN 7-X, which is planned to provide an integrated concept test in order to yield convincing predictions on the properties of ignited plasmas in Advanced Stellarators. Finally, we provide a short summary and give the conclusions of the ANS-Boston invited paper.

In the following summary of the ANS paper, the Helias optimization characteristics are reviewed. Some recent numerical results on the reactor performance of HSR are given. Three specific items are highlighted: losses of fast alpha particles, the divertor and maintenance concepts in HSR, and aspects of size and cost. Characteristic data of HSR are compared to those of a European tokamak reactor study and to ITER. This is followed by a summary and conclusions.

The ANS-paper is published in Fusion Technology, Vol. 21, No. 3, Part 2B, 1767 (1992); a copy can be made available on request by F. Rau, IPP Garching. The reader of *Stellarator News* is also referred to the review on Helias Reactor Studies, contained in Issue No. 19, January 1992.

The HELIAS concept chosen for W7-X is the result of a simultaneous optimization with respect to all criteria which are considered indispensable for good reactor conditions:

High quality of vacuum-field magnetic surfaces to yield good transport properties, i.e., small thickness of islands and avoidance of major resonances.

- ➡ Good finite-beta equilibrium properties to achieve a robust configuration with rising beta at fixed coil currents.
- ➡ Good MHD stability properties to achieve $\langle\beta\rangle \approx 5\%$.
- ➡ Small neoclassical transport under reactor conditions.
- ➡ Small bootstrap current to minimize the ability of the plasma to affect the confinement configuration.
- ➡ Good collisionless alpha-particle containment at operational values of beta.
- ➡ Good modular coil feasibility for coils providing a sufficiently large distance between coils and plasma and thus providing space for a divertor.

The simultaneous achievement of these criteria essentially determines the structure of the magnetic field strength distribution of the configuration. Its geometrical shape is then a consequence of this structure.

A stellarator reactor must 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 5 T in order to keep the NbTi technology available for the superconducting magnets.

There will be no provision for current drive because the configuration essentially eliminates the bootstrap current and a small residual current is tolerable.

For a given plasma pressure distribution, 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.

The magnetic configuration is scaled up linearly in dimensions from $R = 5.5$ m for W7-X to $R = 20$ m. A minor radius of 1.6 m has been selected. Plasma parameters for this reactor are shown in Table I for two fractional concentrations of helium, 1% to simulate the start of burn, and 10% for a burning plasma. The data show that $\langle\beta\rangle$, n, and T are within the intended limits of the reactor. The fusion power output of 2.5 GW for the burning plasma at $\langle\beta\rangle = 4.8\%$ is in the right range. With a value of about 1 MW/m^2, the average neutron wall loading is by no means excessive. The table also shows that the Lackner-Gottardi scaling (LGS), which describes today's Wendelstein experiments rather well, even without any improvement, yields a confinement

Table I. Plasma Parameters of the HELIAS Reactor HSR			
Plasma parameters	**units**	**start-up**	**burn**
$n(0)$	$10^{20}m^{-3}$	3.0	4.0
$\langle n \rangle$	$10^{20}m^{-3}$	1.33	1.77
$\langle n \rangle_L$	$10^{20}m^{3}$	1.93	2.57
$T(0)$	keV	17.0	14.0
$\langle T \rangle$	keV	7.49	6.17
$\langle \beta \rangle$	%	4.57	4.78
E_{tot}	MJ	692	725
Particle fractions			
f_{alpha}	%	1.0	10
f_{oxygen}	%	0.1	0.1
f_{carbon}	%	1.5	1.5
n_{DT}	%	88	70
Z_{eff}		1.5	1.7
Power output			
P_{alpha}	MW	682	516
P_{Fusion}	GW	3.34	2.52
$P_{Neutron}$	GW	2.66	2.0
Neutron flux	MW/m^2	1.2	0.9
Energy confinement times			
τ_{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

time already slightly larger than the required energy confinement time, τ_{req}, whereas the LHD and the gyro-reduced Bohm (GRB) scalings indeed need up to a factor of two improvement, which however, we expect to gain from the W7-X optimization via the reduced banana widths. These three scalings have been selected because they exhibit the positive density scaling ($n^{3/5}$) found in stellarators.

Among the more recently treated issues of HSR are the following three important items:

(A) Is there sufficient confinement for the fusion generated alpha particles?

(B) Is an efficient divertor compatible with the W7-X concept?

(C) What is the relation between tokamaks and HSR concerning their maintenance, size and cost?

A. Fast particle loss fraction

In principle, three-dimensional configurations like stellarators are indeed subject to fast losses of high-energy particles. It is, however, a particular characteristic of the configuration optimized for W7-X that the favorable beta effects which occur for values smaller than the intended operating $\langle \beta \rangle$ of about 5% are sufficient to improve the alpha-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%. In this calculation the modular coil ripple is taken into account. So one can conclude that in HSR the losses of fast alpha-particles can indeed be neglected.

B. The divertor concept

The divertor concept also exploits a property arising as a consequence of the optimization procedure for W7-X. It 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. When appropriately shaped collector surfaces are arranged following these edges at some distance within each field period, then all field lines arrive on the collector surfaces without occurrence of leading edges. 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 two options: having no large islands in the region beyond the last closed flux surface and having a chain of large, e.g. $\iota = 5/5$ islands. 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 100 m^2. This leads to a power density of several MW/m^2, which is considered tolerable. In addition, the concept allows sweeping the divertor exhaust using very moderate ac magnetic fields generated by small localized coils to reduce (if necessary) peak power loads. These coils are optimized in such a way that space for the divertor hardware and the pumping facilities is available where it is needed. One

foreseen for tokamaks, namely through ports without dismantling the machine.

In Table II some relevant engineering parameters of HSR are compared to those of the tokamak reactor PCSR-E and to ITER. Data for ITER are taken from the 1990 report on the *ITER Conceptual Design* . PCSR-E is a European study 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 no longer be needed once the results from the next steps are available. The assumptions about maturity made for HSR are similar to those for PCSR-E.

The essential result in Table II is that the total fusion powers are similar for the two reactors, HSR and PCSR-E, but smaller for ITER. 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 concentrated near the plasma volume, and thus is only used for confinement purposes. This has similar effects on the reactor mass, which involves the forces that must be balanced.

In reviewing other information not presented in the table, one finds that the volume for blanket and shield in HSR is only slightly larger than that in the tokamak reactor although the aspect ratio is rather different. To assess this further would require much more technology-dependent information.

Thus, in essence, these comparisons show that the costs for stellarator reactors of the Helias type and for tokamak reactors are expected to be similar, and if there is a difference, the stellarator reactor should be less costly. This result does not take into account the fact that the somewhat reduced wall loading in stellarator reactors would lead to a longer lifetime for the blanket components and divertor plates.

Summary and Conclusions

Modular stellarator reactors of the Advanced Stellarator type appear to have the potential to offer desirable reactor properties:

- ➠ A single NbTi coil system is sufficient.
- ➠ Steady-state operation is inherent and depends upon refueling and exhaust only; no current drive system is needed.
- ➠ There is no possibility for a major current disruption because no externally driven net toroidal plasma current is required, and internal plasma currents are close to the minimum allowed by physics.
- ➠ While the fusion powers are very similar, the reactor volume, mass and magnetic field energy of HSR are smaller than those of tokamak reactors; costs thus might be comparable or even lower in HSR.

A modular stellarator reactor of the Advanced Stellarator type is essentially a linear enlargement of Wendelstein 7-X by a factor of approximately three in minor radius, together with an increase of the magnetic field by a factor of two. While decisive advantages can easily be identified, no definite disadvantages of this approach have been found.

Extrapolation from W7-X to an ignited stellarator is large but plausible. Thus, the potential of the Advanced Stellarator approach for concept improvement in magnetic fusion is large. 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.

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Table II. Characteristic Data of Various Reactor Designs				
Quantity	units	HSR	PCSR-E	ITER
Fusion power	GW	2.9	3.6	1.1
Average major radius	m	19.5	9.3	6.0
Average plasma radius	m	1.6	3.1	2.9
$n_{DT}(0)$	10^{20} m^{-3}	2.8	≈3	1.3
$T(0)$	keV	15	≈20	17
$\langle\beta\rangle$	%	5.1	3.8	4.2
I_{plasma}	MA	0	17	22
V_{plasma}	m^3	1000	1800	1100
TF-coils				
Average coil radius	m	3.9	≈5.9	5.6
Average coil volume	m^3	9.0	≈18.7	11.3
Current density	MA/m^2	31.5	35	35
Total volume (winding pack)				
TF coils	m^3	450	410	180
PF+OH coils	m^3	0	≈ 600	320
Number of coils		50	38	30
Mass of coil system	t	11×10^3	17.4×10^3	9.8×10^3
Mass (blanket, shield) approx.	t	11×10^3	7×10^3	8×10^3
Vacuum vessel, approx.	t	2×10^3	2×10^4	8×10^3
Total mass	t	2.4×10^4	4.4×10^4	2.5×10^4
Field on axis	T	5.0	6.36	4.85
Max. field on coils	T	10.7	11.3	11.4
Magnetic energy, TF	GJ	70	115	40
Magnetic energy, PF+OH	GJ	0	24	23
Magnetic energy, total	GJ	70	139	63

of the major tasks for W7-X is to develop this divertor system.

C. Comparison in Size and Cost with a Tokamak System

The described reactor properties of Helias reactors only hold for moderate aspect ratios, which are typically near 10. If the aspect ratio were to be made smaller, some of the optimization goals could not be reached. Reactor designs of sufficient accuracy for proper costing are still lacking. This is immediately understood in view of the remaining uncertainties about the exact parameters of future reactors. This is true for tokamaks, and even more so for stellarators.

Another factor that would be difficult to handle in a comparison is the type of engineering assumed in the individual reactor designs. An earlier INTOR study showed that this factor could lead to large differences in costing even for a single system. With these facts in mind, and since it is not the intention to find minor differences between the two systems but rather major differences of a qualitative nature, it is more appropriate to make relative comparisons of cost-determining quantities such as the total fusion power output, the total plasma volume, the average beta, the total magnetic energy needed in the magnetic circuit, the 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.

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 corresponding tokamak knowledge and the application of some of the tokamak-oriented technology developments. For this reason, stellarators should not be considered an independent alternative line but that their development is better described by the term ‘concept improvement’.

A striking engineering aspect of Advanced Stellarator reactors is that they allow a different maintenance approach. This results from the fact that in HSR only a single coil system is needed for producing the confining magnetic field and also that there is no integral twist force on the magnet. In fact, each properly defined field period, except for bending forces, mainly experiences 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 the removed section, where accessibility is large. This also allows the use of a more compact blanket system. Minor components such as tiles or divertor plates will be maintained in the same way as