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Stellarator Fusion Reactors – an Overview

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Abstract

The stellarator system offers a distinct alternative to the mainline approaches to magnetic fusion power and has several potentially major advantages. Since the first proposal of the stellarator concept many reactor studies have been published and these studies reflect the large variety of stellarator configurations. The main representatives are the continuous-coil configurations and the modular-coil configurations. As a continuation of the LHD experiment two reactor configurations, FFHR1 and FFHR2, have been investigated, which use continuous helical windings for providing the magnetic field. The modular coil concept has been realized in the MHH-reactor study (USA 1997) and in the Helias reactor. The Helias reactor combines the principle of plasma optimisation with a modular coil system. The paper also discusses the issues associated with the blanket and the maintenance process. Stellarator configurations with continuous coils such as LHD possess a natural helical divertor, which can be used favourably for impurity control. In advanced stellarators with modular coils the same goal can be achieved by the island divertor. Plasma parameters in the various stellarator reactors are computed on the basis of presently known scaling laws showing that confinement is sufficiently good to provide ignition and self-sustained burn.

Keywords:

stellarator, fusion reactor, modular coil, helical system, power balance

1. Introduction

The main properties of a stellarator reactor are the potential of steady-state operation and the absence of current disruption; a summary of the main features is given in Table 1. The steady-state magnetic fields simplify superconducting magnet design, remove the need for pulsed superconducting coils, and eliminate energy storage required to drive pulsed coils. Plasma confinement during startup and shutdown is aided by the presence of magnetic surfaces at all times during this phase. Steady-state plasma operation after ignition is an outstanding advantage of the stellarator concept.

A stellarator can have a relatively high aspect ratio and does not require expensive complicating auxiliary magnets for field shaping, position control coils and current drive. Its coil configuration permits access to the device from all sides and facilitates a modular approach to blanket and shield design. Since stellarators and torsatrons can operate free of induced toroidal current and do not suffer from major plasma disruptions, the major concern of an excessive energy dump on the first wall and plasma facing components can be eliminated.

2. Stellarator Reactors

Early stellarator reactor designs [1,2,3,4] concluded that the coupled problems of high coil cost and low system power density (i.e., low beta) were particularly severe for the classical stellarator. A particular disadvantage of the classical stellarator configuration

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Table 1 Main features of a stellarator reactor

- Steady-state magnetic fields. No induced eddy currents. No enhanced fatigue of the structure due to pulsed thermal load.
- Steady-state operation at high Q, $Q \rightarrow \infty$.
- No energy storage and low recirculating power requirements.
- Moderate plasma aspect ratio (8-12) which offers good access to the reactor core.
- Start-up on existing magnetic surfaces with good confinement at all instances.
- No positioning or field shaping coils necessary.
- No major disruptions that could lead to an energy dump on the first wall or on the divertor target plates.
- Several potential methods for impurity control and ash removal exist. Magnetic islands at the plasma edge can be used for divertor action
- No toroidal current drive is required.

arises from the interaction between the toroidal field coils and the helical windings. The torsatron [5,6] and modular-coil configurations [7,8], however, show strong promise for alleviating the coil problem per se.

The MIT T-1 torsatron design [6] was the first study of a torsatron reactor. It reflects an attempt to reduce the total power output to < 4 GWt under the assumption of conservative beta limits. Helical reactor design studies are based on the LHD-concept, which has been developed at the National Institute of Fusion Studies (NIFS) in Toki, Japan. The LHD-experiment is a torsatron with l = 2 helical windings and 10 field periods [9] and the advantage of the LHD-concept is the natural divertor with two X-points, which helically encircle the plasma. Apart from this helical structure it has many features in common with the tokamak divertor. Two versions of the LHD-type reactor exist: the Force-Free Helical Reactor (FFHR [10,11]) and the Modular Helical Reactor (MHR [12]). The main feature of the FFHR is the arrangement of the helical windings in such a way that the forces on the helical windings are minimized. This requirement leads to an l = 3 system with 18 field periods (see Table 2).

The first studies of modular stellarator reactors started from the well-known magnetic field configurations, which before have been realized by helical windings and toroidal field coils. The UWTOR-M reactor [13], designed at the University of Wisconsin, is one of the first devices utilising the concept of modular coils. The modular stellarator reactor (MSR) developed at the Los Alamos Laboratories is a classical l = 2, m = 6 configuration generated by 24 modular

Table 2 Helical system reactors

Parameters		FFHR1	FFHR2
Major radius	[m]	20	10
Av. radius of coils	[m]	3.33	2.3
Coil current	[MAturns]	66.6	50
Plasma radius	[m]	2	1.2
Plasma volume	[m^3]	1579	284
Magnetic field $B(0)$	[T]	12	10
Max. field on coils	[T]	14	13
Average beta	[%]	0.7	1.8
Density $n(0)$	[m^{-3}]	2×10^{20}	2.8×10^{20}
Temperature $T(0)$	[keV]	22	27
Fusion power	[MW]	3000	1000
Polarity l		3	2
Field periods		18	10
Magnet. energy	[GJ]	1290	147

coils. The thermal output is on the order of 4000 MW. The study on the modular stellarator ASRA6C was carried out in 1987 as a joint effort of IPP Garching, KfK Karlsruhe and the University of Wisconsin [14]. This study was based on the Wendelstein 7-AS configuration and the aim was directed towards the clarification of critical issues of an advanced modular stellarator reactor and was not meant as a point design.

The MHH is a 4-period modular stellarator reactor, which has been designed in a joint effort by a group of fusion laboratories in the USA [15]. The approach is similar to that used in the ARIES-tokamak reactor studies [16]: an integrated physics, engineering, reactor component and cost optimisation. This stellarator power plant study is the first attempt at an integrated design of a stellarator reactor addressing all major components of the plant: the physics base, the coil system, the blanket, maintenance, the power balance and, last but not least, the cost analysis. The magnetic configuration is basically a Heliac including elements of optimisation as have been developed in the Helias concept. This explains the name: Modular Helias-like Heliac (MHH, see Table 3).

The method to calculate the coil system after the magnetic field has been specified offers the chance to optimize the magnetic field first according to criteria of optimum plasma performance [17] and then to compute the coil system after this procedure has come to a satisfying result. Along this line the advanced stellarator [18] has been developed. The concept of modular coils and the principle of optimisation have been combined in the Wendelstein 7-X device [19], which will demonstrate the reactor capability of the advanced stellarator line. HSR5/22 and HSR4/18 are fusion

Table 3 Modular coil reactors

Parameters		MHH	HSR4/18
Major radius	[m]	14	18
Av. radius of coils	[m]	4.75	5.5
Coil current	[MAturns]	13.8	10.8
Number of modular coils		32	40
Av. plasma radius	[m]	1.63	2.1
Plasma volume	[m^3]	735	1567
Magnetic field $B(0)$	[T]	4.94	5
Max. field on coils	[T]	14.5	10.3
Average beta	[%]	5	3.7
Density $<n_i>$	[m^{-3}]	1.3×10^{20}	2×10^{20}
Temperature $<T>$	[keV]	10	15
Fusion power	[MW]	1730	3000
Field periods		4	4
Magnet. energy	[GJ]	80	100

This list displays only the recent stellarator reactors. (MHH data from ref. [15]).

reactors based on this concept.

The main criteria in designing stellarator reactors are the space needed for a breeding blanket and shield and the conditions of self-sustained burn. The size of blanket and shield depends on atomic physics and is more or less the same in all fusion devices; it is on the order of 1–1.3 m. Any stellarator reactor must be large enough to accomodate a blanket. The confinement time must satisfy the Lawson conditon $n\tau_E > 2\times10^{20}$, which implies that the confinement time must be about 1-2 seconds. Any chance to operate the reactor at high density relaxes the requirements on the confinement time.

Another design parameter is the magnetic field strength, which should be large enough to provide sufficient confinement and high magnetic pressure (small beta values) while on the other hand it should be small to avoid large mechanical forces in the coils and expensive superconductors. Feasibility of the coil system and maintainability of the blanket are other important design criteria. Since there is no consensus yet on a priority list among these criteria, stellarator reactor concepts differs appreciably with respect to size, magnetic field and layout of the blanket.

The magnet system is the most expensive component of the reactor core and for this reason a careful optimisation is required. The continuous coil system as realized in FFHR1 and FFHR2 is a large item, which must be fabricated on the site of the reactor. Since superconducting coils on Nb3Sn basis require the wind-and-react technique large helical windings must either use NbTi-superconductors or another superconductor, which avoids heat treatment after the winding procedure. Modular coils offer the chance to be fabricated separately and be tested before installing these into the reactor. Heat treatment is possible also after finishing the winding process. Spacing beween modular coils should be small to avoid ripple trapping of localized particles, however this conflicts with the requirements for maintenance. Maintenance through portholes needs sufficiently large gaps between adjacent coils. In the Helias reactor HSR5/22 or HSR4/18, 10 coils per period turned out to be a compromise between physics requirements and the need for sufficient access. In MHH the alternative method has been choosen: a whole sector of the coil system together with the blanket will be moved horizontally and replacement of blanket and divertor components occurs through the horizontal gaps. The critical issue of this procedure is the seperation and reconnection of the sector, which must be done by remote control in a radioactive environment. With respect to maintenance the continuous helical coil approach is the ideal one, since sufficiently large gaps between helical windings exist and access to the blanket and the divertor is possible nearly everywhere.

3. Blanket in Stellarator Reactors

As in any toroidal fusion device the purpose of the blanket is to provide sufficient breeding of tritium and to shield the superconducting coils against neutrons. The size of the blanket and its radial width is an absolute figure and any fusion device has to provide enough space to accomodate a blanket with about 1.3m radial build. In contrast to tokamaks the stellarator, however, requires a 3-dimensional design of the blanket, which must conform to the 3-dimensional shape of the plasma. Blanket segments must be small enough to be replaceable through portholes, therefore a stellarator needs a large variety of different modules, which in case of the Helias reactor HSR22 are 250 comprising 25 different shapes. The various blanket concepts, which have been studied for tokamak reactors [20], are also suited for stellarators. The MHH-design favours a self-cooled lithium blanket while in the FFHR1 and FFHR2 molten salt FLIBE ($LiF-BeF_2$) has been selected [21]. Either a solid breeder blanket (HCPB [22]) or a liquid LiPb blanket with additional cooling [23] are the options in the Helias reactor HSR5/22 [24]. Blanket design developed for NET and DEMO reactors have been adapted to the stellarator reactor by W. Daenner [25].

At present no decision can be made as to which blanket concept is the optimum for stellarators. The three-dimensional geometry requires a careful engineering design taking into account the maintenance procedure and the safety requirements. With respect to safety a helium-cooled ceramic breeder is very favourable, however the large amount of beryllium and the accumulation of tritium in the blanket makes reprocessing of the breeder cassettes necessary. Also the size of the blanket segments depends on the available space, the accessibility and the type of the breeder. With a liquid breeder relatively large segments can be installed in a Helias reactor; before maintenance the breeder material will be drained into a dump tank and the empty container can be removed through portholes.

In all stellarator-reactor concepts the average neutron wall load turned out to be rather small, peak values are below 2 MWm^{-2}. This results from the ratio between plasma surface and plasma volume, which increases with major radius. This simple geometrical relation makes a large aspect ratio stellarator more favourable for neutron wall load than a small aspect ratio tokamak, provided the plasma volume and the fusion power are the same. This circumstance has a beneficial effect on the lifetime of first wall and blanket; in MHH a lifetime of 11 years has been estimated. This estimate also holds for the Helias reactors while in FFHR the lifetime is 10 years. However, the number of blanket elements to be replaced during down time is larger than in the equivalent tokamak, and therefore the long lifetime of the structural material in stellarators is only an advantage if by parallel operation the time for maintenance in stellarators is the same as in a tokamak reactor.

4. Divertor

In early stellarator reactor studies little attention has been paid to the divertor problem. In the meantime, however, theoretical and experimental results have strongly increased the data base on divertor action, and the design activity of Wendelstein 7-X and the experiments in LHD and Wendelstein 7-AS have contributed a large amount of experience to the physics and technology of divertors in stellarator geometry. The basic requirement for the divertor in stellarators is the same as in tokamaks: the divertor should protect the first wall from excessive thermal load and thus diminish the influx of impurity ions released from the wall. In tokamaks the separatrix and the associated X-point are axisymmetric while in stellarators the X-point-like structure follows the coil system or the helical shape of the last magnetic surface.

Depending on the stellarator type, divertors are realized in different ways. In moderate to high-shear heliotrons like Heliotron E, CHS and LHD, one can make use of intrinsic diverting field lines to create a helical divertor. By employing additional field perturbation coils these devices benefit from the additional flexibility to create externally imposed islands, which allow the installation of a so-called local island divertor (LID). In low-shear advanced stellarators like W7-AS and W7-X, one makes use of the intrinsic islands [26]. Recent experiments in Wendelstein 7-AS have demonstrated the efficiency of the island divertor in realizing a high density plasma detached from the divertor target plates [27].

5. Operational Limits

In stellarators the density limit and the beta limit define the operational regime of the fusion plasma. Since a large toroidal current does not exist, also no limit on the rotational transform exists, which is necessary to prevent tearing modes and disruptions. Even if in quasi-axisymmetric stellarators a finite bootstrap current exists, any disruption of this current will be mitigated by the external stellarator field. In contrast to tokamak reactors, where the Greenwald limit keeps the average density in the range of $\langle n \rangle \approx 1\times10^{20}$ m^{-3}, stellarators can reach a line averaged density of 3.5×10^{20} m^{-3} [27]. The density limit is a radiative limit and can be explained as the result of the power balance. In a fusion plasma, however, the alpha-heating power grows with the square of the density, which implies that a radiative density limit, in principle, does not exist in a fusion plasma. But in the start-up phase, when the external heating power is large and the alpha-heating power is small, the radiative losses are important and an appropriate route to ignition must be found.

Numerical investigations of the MHD-stability in Helias configurations have shown stability up to an averaged beta of 5% [28]. Stability analysis using CAS3D has shown that the equilibrium of HSR4/18 at $\langle\beta\rangle$=4.3% is unstable against global modes in the boundary regions, while at $\langle\beta\rangle$=3.5% these modes are stable. In tokamaks neoclassical tearing modes threaten to lower the MHD-beta limit. Even if small bootstrap currents occur in stellarators the positive shear ι' stabilizes neoclassical tearing modes.

Neoclassical ripple losses provide a special loss mechanism in stellarators and due to the strong increase with temperature ($\chi_{\rm neo} \sim (\varepsilon_{\rm eff})^{1.5} T^{3.5}/(B^2R^2)$) the ripple-induced loss can be prohibitive to ignition. For this reason the effective helical ripple has been minimised to less than 0.01 in the Helias configuration. In the LHD device this effective ripple can also be reduced appreciably. Furthermore, operating the fusion plasma at low temperature helps to make ripple losses small. Thus it is expected that an effective helical ripple of 0.01 is tolerable.

The most desirable temperature regime in a fusion plasma is the region between 10 keV and 20 keV since (with plasma beta fixed) the fusion power output has a maximum in this region. The Lawson parameter $<n>\tau_E$ has its minimum in this region and the requirements on the energy confinement are the weakest: $<n>\tau_E \geq 2\times10^{20}$. The possibility to choose the average density in the Helias reactor in the range of $<n>=2\text{–}3\times10^{20}\,\mathrm{m}^{-3}$ puts less stringent requirements on the necessary confinement time than in a tokamak reactor with lower densities. In summary, the parameter regime of a stellarator fusion plasma is

- Average density $<n> \approx 2\text{–}3\times10^{20}\,\mathrm{m}^{-3}$
- Average beta $\leq 4\%$
- Temperature regime $T_{max} \leq 20$ keV
- Energy confinement time $\tau_E = 1 - 2$ s

Due to the chance to operate at high density the requirement on the confinement time in the stelllarator reactor is reduced; the required confinement time is beween 1 and 2 seconds. The fusion triple product $<n><T>\tau_E$ is roughly a factor of two smaller than in a tokamak power reactor, where the temperature is about a factor of two higher.

6. Fusion Plasma in Stellarators

Power balance in stellarator reactors has been studied by various methods: local transport calculations using ASTRA [29] and the TOTAL_P-code, and extrapolation of empirical scaling laws of stellarator confinement to reactor conditions [30]. The latter method allows one to make quick parameter studies. In order to check whether the empirical confinement times allow for ignition the following procedure is applied: starting from the design parameters including the plasma profiles the required confinement and the confinement times from empirical laws are computed. As a guideline to model the plasma profiles the results of the ASTRA-code were used. Parameter studies have been made to explore the dependence on the shape of the profiles.

The scaling laws used in the following procedure are described by a power law

Table 4 Exponents of empirical scaling laws

		LGS [31]	ISS95 [32]	W795	New LHD1	New LHD2
C_0		0.21	0.256	0.36	0.263	0.115
P	a5	-0.6	-0.59	-0.54	-0.58	-0.64
R	a1	1	0.65	0.74	0.64	1.02
a	a2	2	2.21	2.21	2.59	2.09
B	a3	0.8	0.83	0.73	1.01	0.85
ι	a7	0.4	0.4	0.43	0.0	0.0
$<n>$	a4	0.6	0.51	0.5	0.51	0.54

$$\tau_E = C_0 R^{a1} a^{a2} B^{a3} <n>^{a4} P^{a5} \iota^{a6}.$$

The coefficients are listed in Table 4. Units: s, m, m, Tesla, $10^{20}\,\mathrm{m}^{-3}$, MW.

A possible isotope effect has not been included in the empirical scaling laws. The geometry of the plasma column is described by two parameters only, the major radius and the effective minor radius a, elliptical elongation or triangularity have not yet been introduced as in tokamaks. The toroidal variation of the plasma cross section is included in the definition of the effective minor radius. NLHD1-scaling and NLHD2-scaling [33] do not depend on the rotational transform, however the experimental data of Wendelstein 7-AS indicate a dependence on the rotational transform and therefore they support the Lackner-Gottardi scaling law in this respect.

Good confinement of highly energetic alpha particles is a necessary condition for self-sustaining burn of the fusion process in a stellarator reactor. In this context the following problems are of importance: sufficient confinement of trapped alpha particles, a small number of particles trapped in the modular ripple, anomalous losses of alpha particles by plasma oscillations. In the present reactor configurations HSR5/22 and HSR4/18, the number of lost alphas is smaller or equal to 2.5%; thus only 2.5% of the heating power is lost by poorly confined alpha particles. Further fine-tuning of the magnetic field is possible to improve the confinement of alpha particles further.

7. Conclusions

In comparing the various reactor candidates the same temperature and density profiles have been assumed. The results are listed in Table 5. Alpha heating power minus bremsstrahlung is the available heating power; also 3% loss of alpha particles has been assumed. In all cases the required confinement time lies

Table 5 Fusion plasma in stellarator reactors

	HSR4/18	MHH	FFHR2	FFHR2/1	
Major radius	18	14	10	15	[m]
Av. minor radius	2.1	1.6	1.2	1.8	[m]
Plasma Volume	1567	734	284	960	[m^3]
Iota	1.0	0.7	0.7	0.7	[]
Magnetic field	5.0	5.0	10.0	6.0	[T]
Line av. density $<n>$	2.056×10^{20}	2.2×10^{20}	2.12×10^{20}	2.12×10^{20}	[m^{-3}]
Electron Density $n(0)$	2.940×10^{20}	3.1×10^{20}	3.04×10^{20}	3.04×10^{20}	[m^{-3}]
Temperature $T(0)$	15	15	15	15	[keV]
Av. Electron Temperature	4.96	5.0	5.0	5.0	[keV]
Beta(0)	13.7	14.6	3.5	9.8	[%]
Average Beta	3.67	3.95	0.96	2.66	[%]
Fusion power P_{fus}	3.155	1.72	0.62	2.09	[GW]
Energy Conf.Time (req.)	1.71	1.56	1.72	1.64	[s]
Energy Conf.Time (NLHD2)	1.57	1.08	1.6	1.5	[s]
Energy Conf.Time (LGS)	2.22	1.34	1.7	1.79	[s]
Energy Conf.Time (ISS95)	1.20	0.74	1.0	0.96	[s]
Energy Conf.Time (NLHD1)	2.37	1.53	2.02	2.2	[s]

between 1.5 and 1.75 s; it is about a factor of two larger than the ISS95 confinement time. However, predictions on the basis of LGS and NLHD-scaling are more favourable: in HSR4/18 LGS and NLHD1 predict confinement times which are larger than the required ones, in FFHR2 the result is marginal and the MHH device needs a small improvement factor. The configuration FFHR2/1 is an enlarged version of FFHR2, where the increase of size has been compensated by a reduction of the magnetic field. Also in this case empirical scaling laws predict ignition without the need of an improvement factor. In all cases the average beta value stays below 4%, which in view of the highest beta values in LHD and W 7-AS seems to be an achievable goal. Also the assumed density of $<n>\approx2\times10^{20}$ m^{-3} is below the highest density in W 7-AS. Thus, only a moderate amount of optimism is needed to extrapolate present stellarator results to a self-sustained fusion plasma.

The economic viability of stellarator reactors is mainly determined by the magnet system and the complexity of the blanket. These are the cost-driving components and for this reason any chance should be taken to operate the stellarator at low magnetic fields, even if this requires an increase in size. In this case there are good prospects that superconducting coils on NbTi basis can be used. Concerning the blanket, several options as in tokamaks are available, a preference, however, cannot yet be made. In this respect further studies are necessary taking into account the maintenance concept.

References

[1] L. Spitzer, D. Grove, W. Johnson, L. Tonks and W. Westendorp, USAEC Report NYo-6047 (1954).

[2] K. Miyamoto, Nucl. Fusion **18**, 243 (1978).

[3] A. Gibson, Proc. BNES Nuclear Fusion Reactor Conf. at Culham Laboratory, 233 (September, 1969).

[4] A. Gibson, R. Hancox and R.J. Bickerton, Proc. 4th Inter. Conf. on Plasma Physics and Controlled Nuclear Fusion Research, IAEA-CN-28/K-4, III, 375 (June 17-23, 1971).

[5] C. Gourdon, D. Marty, E.K. Maschke and J. Touche, Nucl. Fusion **11**, 161 (1971).

[6] P.A. Politzer, L.M. Lidsky and D.B. Montgomery, Massachusetts Institute of Technology report PFC/TR-79-1 (March, 1979).

[7] W. Wobig and S. Rehker, Proc. 7th Symp. on Fusion Technology, Grenoble, 333 (1972).

[8] R.L. Miller and R.A. Krakowski, Los Alamos National Laboratory report (June, 1981).

[9] A. Iiyoshi *et al.*, Fusion Technol. **17**, 169 (1990).

[10] A. Sagara *et al.*, Fusion Eng. Des. **41**, 349 (1998).

[11] K. Yamazaki *et al.*, Proc. 25th EPS Conf. On Contr. Fusion and Plasma Physics, Maastricht, June 14-18, 1999.

[12] K. Yamazaki *et al.*, 16th IAEA-Conf. Montreal 1996, paper IAEA-CN-64/GI-5.

[13] B. Badger *et al.*, University of Wisconsin Report UWFDM-550 (1982).

[14] G. Böhme *et al.*, IPP-report IPP 2/285, KfK-report KfK 4268, Fusion Power Assoc./Univ. Wisc. FPA-

87-2.

[15] *Stellarator power plant study*, UCSD-ENG-004 (1997).

[16] *The Aries – I Tokamak Reactor Study*, UCLA-PPG-1323 Volume II.

[17] J. Nührenberg, R. Zille, Proc. of the workshop on Theory of Fusion Plasmas, Varenna 1987, Editrice Comp., Bologna 1987, EUR 11336 EN.

[18] U. Brossmann *et al.*, Plasma Phys. and Contr. Nucl. Fusion Res., (Proc. 9th Int. Conf., Baltimore 1982) Vol. III, 141 (IAEA Vienna 1983).

[19] C.D. Beidler *et al.*, Fusion Technol. **17**, 148 (1990).

[20] Malang, S, Fusion Eng. and Des. **46**, 193 (1999)

[21] H. Yamanishi *et al.*, Fusion Eng. Des. **41**, 583 (1998).

[22] M. dalle Donne, KfK-report 5429 (1994).

[23] S. Malang *et al.*, FZKA 5581 (1995).

[24] H. Wobig, E. Harmeyer, F. Herrnegger, J. Kisslinger, IPP-Report IPP III/244.

[25] W. Daenner, IPP-report IPP2/331 (1996) p.262.

[26] H. Wobig, J. Kisslinger and V. Welge, Rep. IPP 2/ 313, p.189.

[27] Grigull *et al.*, 28th EPS-Conf. on Plasma Phys. and Contr. Fusion (Madeira 2001) to be published.

[28] J. Nührenberg *et al.*, Plasma Phys. Control. Fusion **35**, B115-B128 (1993).

[29] N.E. Karulin, IPP Report IPP 2/337 (1997).

[30] Beidler, C.D. *et al.*, Proc 17th IAEA Conf. on Fus. Energy, Yokohama 1998, IAEA-F1-CN-69/FTP/01.

[31] K. Lackner and E. Gottardi, Nucl. Fusion **30**, 767 1990.

[32] U. Stroth *et al.*, Nucl. Fusion **36**, 1063 (1996)

[33] K. Yamazaki et al. IAEA-Conf. on Fusion Energy, Sorrento 2000, paper (FTP2/12).