Critical Issues and Comparison of Optimized Stellarators

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Abstract. Optimized stellarators are compared and some of their critical issues are discussed. Deficiencies of traditional stellarators are briefly recalled. Characteristics of three lines of stellarators which have emerged from the effort to remedy those traditional deficiencies are briefly described. There are different critical issues associated with these lines. Five topics have been selected and tentative conclusions are given.

1. Introduction

Due to Spitzer stellarators now exist for more than half a century. For more than a quarter of a century they did not seem viable for fusion because they were plagued by bad magnetic surfaces, estimates of β too low for fusion, bad orbits. Figure 1 shows just one example: a typical reflected α -particle in the W7-AS magnetic field at power plant size drifts out of the confinement region in about 1/10 of a millisecond [1].

So, with all these problems a decisive improvement is not likely through improvement of details but only with a different concept. Traditionally, stellarators were designed by selecting coil systems and studying their magnetic fields. In search for a possibility of optimization, theory indicated a different path: the relevant physics, MHD and driftkinetics, show that if one assumes, or somehow guarantees, the existence of magnetic surfaces the decisive structure governing MHD and driftkinetic properties is the structure of the strength of B on magnetic surfaces. This structure is only nonlinearly and not transparently linked to the vector field $\vec{B}$ so that computational theory has been essential.

Three lines have emerged from these theoretical guidelines [2]. If one can make the structure of mod B helically symmetric then many deficiencies are simultaneously eliminated. If one can introduce external rotational transform into a tokamak without losing too many of its advantages one may mitigate the problems of disruption and current drive. If one can satisfy fusion requirements without bootstrap current then plasma dynamics and control might be simplest. These choices lead to different characteristics of the structure of B and associated characteristic orbit behaviors which are briefly recalled.

Figure 2 shows quasihelical symmetry in a configuration with 4 periods [3] prototypical of HSX [4]. The coloring of the strength of B on the boundary shows the helical structure completely overwhelming the toroidal effect. However, as Garren and Boozer [5] have shown, this helical symmetry of B can not be exact. It can

however be realized to such an extent that orbits indeed drift in the helical groove of low B values, see Fig. 3. Ref. [4] gives experimental evidence for improved energetic-particle confinement.

Figure 4 shows quasi-axisymmetry [6] in a configuration with 2 periods prototypical of planning at NIFS [7]. A configuration with three periods is under construction at PPPL [8]. Again, this quasi-symmetry does not exist exactly, but can be realized to the extent that particles drift in the toroidal groove of low B values, see Fig. 5, so that neoclassical $1/\nu$ transport becomes negligible [9].

Figure 6 shows quasi-isodynamicity [10] in W7-X at finite β [11], obviously a truly three-dimensional structure which as the minimum of B on a surface is not on the outside of the torus leads to poloidal closure of reflected-particle orbits, see Fig. 7. In W7-X the maximum of B is on the inner side of the crescent-shaped plasma cross-section so that shallowly reflected particles can transit between periods on the outer torus side. This class of particles can lead to collisionless stochastic loss, which is not necessarily strong [3] but can be eliminated by extending the maximum to a poloidal line.

Figure 8 shows quasi-isodynamicity with poloidally closed contours of B in a configuration with 6 periods [12]. Results for a 12-period version are presented in Ref. [13]. A 2-period version is being planned at Oak Ridge [8].

The inward shifted configuration obtained in LHD is an example of an experimental optimization of a configuration for given coils [14].

Of the various critical issues associated with optimized stellarators the following topics have been selected: the topology of the confinement region as is in particular appropriate for stellarators, MHD stability (briefly, because of an MHD review in this conference [15]), neoclassical α -particle confinement, turbulence and exhaust.

2. Magnetic topology

Figure 9 shows 400 magnetic surfaces of the vacuum field as obtained from the coils of W7-X computed with perfect periodicity [16]; so, in principle a three-dimensional confinement region can have quite nice surface structure with only thin islands. The lowest-order island of significant width here is the 10/11 island.

If one has an isolated low-order island in a vacuum field – Figure 10 shows a 1/1 island from misplaced modular coils superimposed over a chain of 5/5 islands – a set of control coils can be used to eliminate it as is

seen in Figure 11. With a good set of control coils they can be at several plasma radii distance and the total control current of order 10^{-4} of the total current needed for the confining field if it is a 1/1 island [17].

Similar statements can be made about the quality of surfaces at finite β . Figure 12 shows a cross section of NCSX as obtained in a PIES equilibrium calculation. The lower half shows finite- β 3/5 island formation as resulting from a preliminary coil design; the upper half shows the elimination of these islands with an improved coil design [18]. This particular equilibrium is obtained at $\langle \beta \rangle = 0.04$.

Figure 13 shows a PIES equilibrium obtained for W7-X at the same $\langle \beta \rangle (\approx 0.04)$, but here without any finite- β correction to the coils generating the vacuum field. Another important feature here is the ergodization of the 5/5 island region. At zero β the island separatrix is nearly as clear as in a twodimensional case while here the islands are wider and the separatrix region is ergodized so that the plasma volume is about 10% smaller than in the standard high-mirror case [19]. Also, this island situation will be discussed in the context of plasma exhaust, see below. The phenomenon of ergodization at finite β has also been observed with the other 3d equilibrium code that does not assume nested surfaces, the HINT code, in computations of LHD equilibria at 1.4 and 3.7 % of β , see Fig. 14, where the region of long field lines has been found to be part of the main plasma region for the present high- β high-collisionality plasmas [20].

The following points try to summarize this section on magnetic topology.

One critical issue may be the experimental controllability of islands in the plasma core. Because of the linkage of the rotational transform profile to the bs current profile one may expect that this is a more advanced task in NCSX than in W7-X so that probably NCSX will be the crucial experiment in this respect. On the other hand there already is positive experience with handling of the 1/1 island in LHD [21] and small islands may behave more benign in more refined physics descriptions.

Another issue is the finite- β ergodization of the plasma boundary. Here, from the point of view of theory, NCSX is in an advanced situation as compared with W7-X since its coil design is strictly based on finite- β equilibria while W7-X's coil design is so old that it only took into account a given plasma shape that it approximately realized as a vacuum field. On the other hand ergodization - if it does not reduce the plasma volume too much - may be beneficial for divertor operation as will be seen below. Certainly, more HINT and PIES studies are needed and one has to keep in mind that in none of these devices high- β experiments will be truly lmfp experiments.

3. MHD stability

With the experiments LHD and W7-AS the assessment of the significance of linear MHD in stellarators has only recently started, a situation different from that of tokamaks where linear ballooning and kink modes give clear operation boundaries. The experimental situation is discussed in a separate review [15], so that a few remarks may suffice. Linear MHD mode stability thresholds apparently do not set rigid operational boundaries. It is not yet known whether this will be the same for a low-shear high- β stellarator like W7-X, if one does not extrapolate W7-AS's results to W7-X. Until NSCX's operation it will not be definitely known whether tokamak kink-mode behavior is similar to a qa device with external rotational transform. Below, theoretical results for the three lines, in particular the role of the bs current, will be briefly described.

QH configurations can have a high MHD stability limit; this is in particular brought out by their helical limit for which β can be as high as 30% as was already shown in the 80's but at that time without considering the bs current [22]. With bs current, there is a significant reduction of rotational transform – with considerable effort created by modular coils from the outside – in the lmfp regime [23] which, intuitively, does not appear to be an attractive feature in itself, however, this effect also reduces the excursions of orbits from flux surfaces. Furthermore, the most limiting instability in standard high- β tokamaks associated with the bs current, the neoclassical tearing mode, should be absent [24]. Convergence problems of VMEC equilibria with self-consistent bs current encountered in case studies [23] and the possible instability with respect to kink modes need further clarification.

In QA configurations the bs current increases the rotational transform so that it should improve the confinement properties except if it causes kink instability. In order to avoid this, one needs careful shaping of the pressure and current profile and can then, as shown in the NCSX design, achieve kink-mode stable equilibria of the type shown in the section on magnetic surfaces with $\beta = 4\%$ [9]. Ref. [25] presents more work for qa plasmas.

Low-node-number free-boundary modes, i.e. kink modes, are found stable in the 4%- β PIES equilibrium for W7-X described above. On the other hand, the peeling criterion is slightly violated, so that peeling-ballooning modes were candidates for unstable modes. Figure 15 shows such a mode structure: it is dominantly an $m = 27$ free-boundary mode, clearly with ballooning character; however it is stable [26], so that unstable modes must have an even finer structure.

So, with these indications, that NCSX and W7-X may be stable at β of 4% or could even operate at

somewhat higher β values, and that stable β values of about 0.1 seem possible in qi configurations [13], MHD stability does not appear to be critical in these types of optimized stellarators.

4. Neoclassical collisionless α -particle confinement

For all three lines, qh, qa, qi, configurations with good collisionless α -particle exist, i.e. for some reasonable approximate normalization of magnetic field and confinement volume, e.g. 5 T at the magnetic axis and a plasma volume of thousand cubic meters, only a very small fraction of 3.5 MeV α -particles started in the plasma core gets lost. So, the issue is, whether other physics goals, like ‘sufficiently high β value’, or ‘the bs current has the right magnitude and shape so that no current drive is needed’, can be achieved simultaneously. It appears that there is no basic problem of this type for qi configurations – an example is given below – but there may remain such a type of optimization problem for qa and qh, namely to have fully consistent qa and qh cases with kink stability and bs current only. Probably such cases exist, see also [25]. As far as α -particle confinement – seen more generally than just considered with drift orbits – is concerned an important field to be developed for stellarators is energetic-particle driven modes. α -particle losses of this nature will reliably only be described nonlinearly, so with big computational effort. Probably this will mainly be important for qi because qa will probably mainly lean on tokamak theory and experiment. Analytical estimates can be obtained [27] and computational linear theory is reasonably developed; an example is shown below.

Figure 16 shows an example of collisionless energetic-particle confinement [28]. 14.7MeV protons were started at a quarter of the toroidal flux, ie. half the plasma radius, and half the toroidal flux, ie. 70% of the plasma radius, and obviously nicely confined and the configuration is stable at an $\langle \beta \rangle$ of 5%.

Figure 17 shows the effect of an NBI slowing-down distribution on MHD modes in a W7-AS discharge [29]. Growth rates of unstable modes for maximum beam velocities beyond 1/3 of the Alfvén velocity are shown. These results have been successfully benchmarked with approximate analytical results [30] so that development of nonlinear codes can now be started.

5. Turbulence

Here, a very tentative discussion of turbulence in optimized stellarators is given. This will be even more tentative for core turbulence than for edge turbulence, because one expects that gyrokinetic turbulence has to be investigated in the plasma core while two-fluid turbulence may give reasonably reliable answers at the plasma edge. For the core, tentative comments on turbulence vs. neoclassical transport, on tokamaks vs. stellarators,

on driving instabilities and zonal flow behavior are given.

The LHD confinement results for the standard and the inward-shifted configuration appear to show [14, 31] that reduction of neoclassical transport is indeed beneficial. In NCSX the neoclassical $1/\nu$ ripple transport will be negligible against standard tokamak turbulent transport, so that the assessment of its core transport property is feasible as in a tokamak. In W7-X neoclassical transport will be non-negligible but so small that - in itself - it would not pose a problem for extrapolation to a fusion plasma. So, as in the inward-shifted LHD, it needs to be determined accurately for a reliable extrapolation to become possible. As far as standard tokamaks are concerned, experiments and theory are well advanced so that ion-temperature-gradient driven, enhanced by trapped-electron drive, turbulence [32] is likely with the role of etg turbulence still to be clarified. This picture gets substantially modified in reversed shear, ie. stellarator-type shear, tokamaks. For them, in the computational analysis, the linear mode character changes from ballooning- to slab-like [33] and to less trapped-particle drive [34] and the turbulence tends to be suppressed by zonal flows in the reversed-shear region [35]. These theoretical results correlate with experimentally observed transport barrier formation, see, e.g. [35] and references therein. This immediately leads to the temptation to predict low turbulent transport in NCSX, so that its actual transport will be very instructive. In qh symmetry, slab and trapped-electron drive are important [36]. For W7-X, a closely related experiment does not exist, but from linear itg analysis it is known that curvature drive and slab drive are comparable with the slab drive dominating at finite β [33, 37] and that the trapped-electron drive should be small because the reflected electrons are localized in small-curvature regions [38]. With the role of the trapped-electron drive still to be clarified, a tentative optimistic conjecture from these indications would be that the zonal flow behavior [39] and not the linear growth rate will determine the level of turbulence. In this context, reduced neoclassical-flow damping measured in HSX [40], and probably prevailing in qa and qi as well, may also be of importance.

Results on plasma-edge turbulence give a further hint on the importance of zonal flows. Drift-alfvén turbulence reasonably describes tokamak L-mode transport [41]. This type of turbulence does not include the drive from kink or peeling instabilities. Therefore, it is possible to consider fictitious tokamak configurations with profiles of rotational transform not possible in tokamak experiments. What has been found is that transport gets strongly suppressed by going to stellarator-like shear and stellarator-like rotational transform at the edge [42]. The evidence is seen in Figures 18 and 19; Figure 20 compares these results with results in the W7-X magnetic field.

Figure 18 shows energy and particle transport vs shear in a tokamak with tokamak-type aspect ratio 3.3 and proves that eliminating the tokamak shear leads to a reduction by about a factor of 4 in transport. If one compares the standard tokamak to an aspect ratio 10 tokamak with stellarator-type shear there still is a factor of 2.

Figure 19 shows energy and particle transport on a logarithmic scale vs edge value of rotational transform. Here, a decrease of transport is seen, going from tokamak edge transform to NCSX-type transform to W7-X-type transform, by a factor of 5 to an order of magnitude. Two specific comparisons can be made: Going from standard tokamak to NCSX type, a factor of 6 is seen which is roughly equivalent to the difference between L- and H-mode transport. Going from standard tokamak to W7-X type, a factor of about 10 is seen, so that the turbulence calculation in the actual W7-X geometry is very interesting. One finds about the same transport in an aspect-ratio 10 tokamak with W7-X type rotational transform and in W7-X with its real geometry. Figure 20 compares the turbulence structures in a standard tokamak and in W7-X. The potential shown indicates much stronger zonal flow in the W7-X case than in the standard tokamak case.

6. Power and particle exhaust

Finally, the divertor problem is briefly discussed. Some very general statements are: up to now all theoretically optimized stellarators have similar flux surface shape; so, in particular, they exhibit helical edges along which field lines beyond the lcms are preferentially diverted so that divertor surfaces vis-a-vis these edges collect these field lines which leads to separation between field-lines starting from the wall and those starting from the plasma. Two illustrations are seen in Figures 21 and 22 [43]: the helical edges characteristic of optimized stellarators and the separation of wall field lines and plasma fieldlines caused by collecting the fieldlines in the areas vis-a-vis the edges. As far as the plasma parameters are concerned, stellarators do not exhibit the Greenwald limit of tokamaks [44] so that there is a tendency for stellarators to work at high density. The HDH mode discovered in W7-AS [44] is maybe not completely understood, but see [45], so that it is not clear whether it will also appear in W7-X but the crucial experiment in this context will perhaps be NCSX because of its closer relation to tokamaks: will it or will it not show the Greenwald limit?

Various topologies of the magnetic field at the plasma edge can be envisaged. There may be a lcms beyond which all field lines starting close to it are open field lines. This might be the typical situation in NCSX. There also may be a well-defined chain of islands with a clear separatrix. This was the situation of W7-AS and will be

the one of W7-X at small β values. This chain of islands as it appears in the Poincaré plot may be a single flux tube, as. e.g. the 5/9 island in W7-AS, or the 5/6 island in W7-X, or it may consist of a number of separated flux tubes depending on the number periods. For example in W7-X at $\iota = 1$ or 5/5 at the boundary there will be five flux tubes. These will be sensitive to low-order field perturbations so that such perturbations should be avoided or compensated as discussed in the section on magnetic topology.

This situation will be ergodized by the finite- β effect of the plasma, as discussed in the PIES W7-X solution, or it may be ergodized by vacuum field perturbations as e.g. in DIII-D, see e.g. [46]. Also, there is the possibility to have one large island as in the LID experiment in LHD [47]. Both of the stellarator-divertor realizations, LID in LHD and W7-AS's, give quite encouraging results [47, 45]. Very promising results are obtained from divertor operation of the DIII-D ergodized edge [46]. Figure 23 [48] shows a Poincaré plot of the magnetic field of the ergodized DIII-D plasma edge with magnetic islands, e.g. the outermost one the 1/4 island chain, embedded in an ergodized field, to the left a calculation of the 2d temperature distribution from the core outwards to the divertor plates, to the right a cut through the 3D temperature distribution with a characteristic splitting of each of the two temperature fans also seen in the experiments.

W7-X is being built with an island divertor [49]. Its various operating scenarios include three somewhat different modes of divertor operation. Remembering the 5/9 island situation in W7-AS one might expect similarities with the 5/6 island situation in W7-X. Operating at $\iota_{\text{edge}} = 1$ one will have 5 separate island flux tubes if they are not perturbed too much by a 1/1 island from fabrication inaccuracies which might require trim coils. And at 5/4 a partially ergodized field line structure can already be explored at low β . Figure 24 [50] shows a simplified CAD drawing of the ten divertor modules W7-X will have; they were designed to enable divertor operation of the three different island topologies mentioned above. The plasma will be about 12 m across, so that these are large devices hopefully creating a possibility for relatively low local heat loads. Figure 25 shows a preliminary evidence for this possibility in a calculation of heat flux densities on such targets from a three-dimensional solution of the temperature equation between the plasma edge and the targets [51]. Three different cases are considered: the vacuum field case with negligible ergodization and anomalous perpendicular heat conductivity of $1 \text{ m}^2/\text{s}$ with peak loads up to 12 MW/m^2 ; a finite- β case with significant ergodization in which the heat load strongly drops to a few MW/m^2 with a peak on short field lines of 5 MW/m^2 ; and a test case with nearly vanishing anomalous perpendicular heat conductivity which shows that there is no artificial numerical perpendicular heat conductivity.

7. Summary

Qa, qh and qi appear sufficiently separately positioned in stellarator configurational space to give complementary physics insight. Of the four issues, topology of the confining field, MHD stability, neoclassical α particle confinement and exhaust probably none of these issues is without solution. It will be very instructive to explore the high- β perspective of qh confinement in more detail.

In the future energetic-particle driven modes will become a crucial issue because ITER will have experimental results from fusion α -particles, which stellarators will not have, so that experiments with NBI and ICRH in stellarators and computational theory explaining such experiments as well as ITER results will be needed for a credible extrapolation.

The topology of the magnetic field, MHD stability and α -particle confinement subject to the bootstrap current dynamics appear to be most demanding in qa so that NCSX will probably be the crucial experiment.

Qi may have a unique characteristic, the possibility of very high β , however only at high aspect ratio.

Core turbulence in qa and tokamaks should be relatively directly related so that their comparison will be very instructive if this issue will be clearly separable from the topology of magnetic surfaces. Understanding qh turbulence should give additional insight into the role of trapped electrons. For qi it will be crucial to distinguish turbulent from neoclassical transport and computationally explain both quantitatively.

Results up to now suggest that edge turbulence could be very different from tokamak L-mode behavior but will be more complex because of ergodization.

Qa and qi devices will probably explore different plasma edge topologies. The existence of W7-AS's HDH regime in LHD, NCSX and W7-X will be of great importance. Closed divertors in larger island structures in larger devices, topologically similar to the LID in LHD, might also be very attractive.

The mid-size experiments LHD, NCSX and W7-X together with their detailed theoretical interpretation will strongly drive understanding fusion.

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Figure Captions

Fig. 1. Reflected α -particle born in the core of a confinement region of W7-AS geometry with 10^3 m^3 volume and 5 T magnetic field.

Fig. 2. A quasi-helically symmetric configuration with 4 periods. The strength of B at the plasma boundary is color-coded.

Fig. 3. Drift of a reflected particle in a quasi-helically symmetric configuration.

Fig. 4. A quasi-axisymmetric configuration with 2 periods. The strength of B at the plasma boundary is color-coded.

Fig. 5. Drift of a reflected particle in a quasi-axisymmetric configuration.

Fig. 6. A quasi-isodynamic configuration with 5 periods (the optimized configuration for W7-X at $\langle\beta\rangle 0.05$). The strength of B at the plasma boundary is color-coded.

Fig. 7. Drift of a reflected particle in the quasi-isodynamic configuration of Fig. 6.

Fig. 8. A quasi-isodynamic configuration with poloidally closed contours of B and 6 periods. The strength of B at the plasma boundary is color-coded.

Fig. 9. 400 magnetic surfaces calculated from the W7-X modular coils.

Fig. 10. A W7-X magnetic field with $\iota = 1$ in the confinement (this is not one of the standard scenarios foreseen) perturbed by misalignment of two modular coils.

Fig. 11. The situation of Fig. 10 corrected with a set of island control coils.

Fig. 12. Magnetic surfaces of a PIES NCSX equilibrium for a preliminary coil design in the lower half and a refined coil design in the upper half.

Fig. 13. Magnetic surfaces of a PIES W7-X equilibrium.

Fig. 14. Magnetic surfaces of two HINT equilibria.

Fig. 15. Structure of a stable ballooning-peeling mode in the PIES equilibrium of Fig. 13.

Fig. 16. Loss history of 1000 14.7 MeV protons started at .5 and .7 of the plasma radius and randomly distributed in the angular variables and the pitch angle. Normalization: plasma volume 10^3 m^3 , magnetic field 5 T. Each symbol marks the loss of one particle.

Fig. 17. Damping by thermal ions and growth by NBI ions of a TAE mode in W7-AS.

Fig. 18. Normalized fluxes associated with drift-alfvén turbulence for circular tokamak configurations with near-edge rotational transform 0.3, various shears and two aspect ratios.

Fig. 19. Normalized fluxes associated with drift-alfvén turbulence for circular tokamak configurations with various near-edge rotational transforms, two shear values and two aspect ratios.

Fig. 20. The electric potential associated with drift-alfvén turbulence in a flux tube (y poloidal, x radial coordinates).

Fig. 21. The helical edges similar in structure in qh, qa, qi optimized stellarators.

Fig. 22. Cross-section with crescent-shaped plasma (yellow), field lines started at the plasma boundary and subjected to field line diffusion (red), divertor troughs (blue) and field lines started at the wall (green).

Fig. 23. Temperature fans extending from the plasma boundary in a 2d DIII-D (left) and in an ergodized DIII-D (middle) configuration (right).

Fig. 24. Divertor structures foreseen in W7-X.

Fig. 25. Influence of finite- β ergodization on heat fluxes to divertor plates in W7-X.

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