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Configuration Optimization and Physics Basis for ARIES-CS

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CONFIGURATION OPTIMIZATION AND PHYSICS BASIS FOR ARIES-CS

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We report on the development of the physics basis for ARIES-CS. ARIES-CS is to study fusion power plants--their economics and engineering characteristics--with a compact stellarator as the core. Our efforts have been centered on the quasi-axially symmetric stellarator configurations with small number of field periods and low aspect ratios. The baseline design chosen has 3 field periods, aspect ratio 4.5 and major radius 7.75 m to yield 1 GW electric power. The configuration is optimized to be MHD stable to ~ 4-5% beta calculated using the linear ideal MHD theories. The transport properties are also optimized to limit the ripple losses. The configuration has effective ripples < 0.6% everywhere and the energy loss of alpha particles <5% when operating in high collisionality regimes. In addition, modular coils have been designed which are optimized to provide adequate space for blanket and shielding requiring only moderate major radii for power plants generating GW electrical powers.

I. INTRODUCTION

The discovery of drift-orbit optimized stellarators raised the hope that fusion reactors may be designed with good particle confinement typically found in tokamaks and MHD stable plasmas free of frequent disruptions typically found in stellarators. These positive traits of stellarator power plants have to be balanced by the complexities they introduce. The feasibility of remotely maintaining the machine and the cost associated with it must also not be overwhelming. Indeed, the singly most important figure of merit that a power plant is measured is the cost of electricity (COE), and to reduce COE the plant size is one of the most important parameters in the design optimization. It is the aim of the ARIES-CS project to study compact reactor systems consisting of an optimized compact stellarator core. This paper addresses the physics basis for the core plasma configuration.

Stellarator fusion reactors have been studied in the past but almost all of them are much larger than plants based on toroidally symmetric power cores. Among different field symmetries in stellarators, we have been concentrating on quasi-axially symmetric (QA) configurations. If the magnetic field strength can be made truly axially symmetric, then the configuration will look exactly the same as tokamaks from the point of view of particle drift orbit and therefore particles will be well confined, just like in tokamaks. On the other hand, the stronger toroidal coupling as the aspect ratio gets smaller has smaller effects on the symmetry properties of the axially symmetric configuration. Thus, using QA it is easier to design configurations with smaller aspect ratios and smaller number of field periods and hence they can be made more compact.

The basic physics properties of a stellarator are determined once the last closed magnetic surface (LCMS) is prescribed. Being in the three-dimensional space, there are literally infinite numbers of ways that a boundary can be described, but in reality there are only finite sets of parameters that are useful to define a plasma boundary which gives meaningful physics properties. The drift orbit of particles depends only on the magnitude of the field strength, not the vector components. If certain symmetry condition is satisfied, the drift orbit will be entirely confined. The shape of the LCMS may be entirely non-symmetric but the underlining field strength still can be made to follow certain symmetry. On the other hand, the residues in the magnetic spectrum that arises from the shaping of the LCMS to give certain symmetry can not practically be eliminated totally in the entire volume of the plasma. Such residues can, however, be minimized. Therefore, designing a modern stellarator becomes a non-linear, mathematical optimization problem, one that is to maximize the selected symmetry property while subject to additional constraints by varying the shape of the LCMS.

Being like tokamaks, QA configurations will have boostrap currents which have the same sign as the external transform provided by the stellarator coils. The amount of current depends on β, temperature and density distribution and the overall rotational transform. The bootstrap current helps increase the overall transform, but it is also a potential driving force for MHD instability. The current also tends to produce large magnetic shear, leading to the possibility of introducing more rational surfaces. The goal of QA configuration design is to make the configuration as QA as possible yet at the same time to use the freedom of shaping the plasma to modify the local shear to minimize the effects of the unstable modes in driving MHD instabilities and to reduce the resonance perturbations that cause the formation of large islands.

It is important to point out that unlike other ARIES studies for tokamks where ample theoretical and experimental understanding are used to extrapolate to the design of reactors, stellarator devices with optimized drift orbits are just being built and the understanding of the plasma in three-dimensional geometry has much to be explored and understood.

We started the development of configurations for the ARIES-CS by limiting $m \leq 6$ and $n \leq 4$ in the initial description of the LCMS, where m and n are the poloidal and toroidal mode numbers in a double Fourier representation of the plasma. We require that the external transform accounts for at least 50% of the total transform and that the transform is an increasing function of radius. We require further that the configuration is MHD stable to the ballooning and external kink modes and that the residues ought to be low enough so that the effective ripples will be < 1% and the collisionless loss orbits of the α particles are minimized. To achieve these goals we have relied on an efficient non-linear optimization package, STELLOPT, to search the configuration space. A description of the optimization process and code package is given in [1].

In addition to the plasma configuration, coils need to be designed and optimized so that the chosen plasma can be realized. Coils for the target plasma may be designed by requiring that the normal components of the magnetic field on the LCMS due to the coils cancel that due to the plasma current. Because of the discrete nature of coils the normal field on the LCMS may not vanish exactly, but the errors may be minimized. We represent the coil geometry parametrically as double Fourier series in terms of toroidal and poloidal angles on a winding surface. The winding surface itself in turn is represented as double Fourier series in the toroidal and poloidal angles. An optimal solution is sought such that residues of the normal field on the LCMS are minimized while other design constraints are satisfied. For DT reactors the tritium breeding and coil protection from radiation damage typically require a blanket and shield to have certain minimum thickness. We included the coil aspect ratio R/Δ_{min}(C-P) as a constraint in the design optimization, where R is the plasma major radius and Δ_{min} (C-P) is the minimum separation between coil centers and LCMS. In addition, we impose in the coil optimization the constraints of coil separation ratio R/Δ_{min}(C-C), where Δ_{min}(C-C) is the minimum separation among coils, and the minimum radius of curvature as well. We allow the coils to have different currents, but they have to maintain stellarator symmetry. Typically we search solutions for which the coil aspect ratio is <6, coil separation ratio < 12, and major radius to minimum radius of curvature <12.

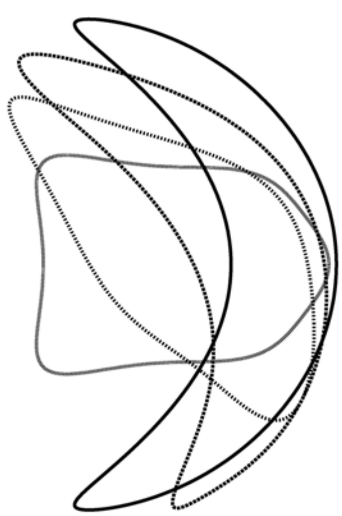

Fig. 1. The last closed magnetic surface of the baseline plasma shown in four equal toroidal angles over half a field period.

II. PHYSICS PROPERTIES OF THE BASELINE PLASMA CONFIGURATION

II.A. Features of the Plasma Configuration

The baseline plasma configuration is a three field period configuration with an aspect ratio 4.5 and major radius 7.75 m. The major radius is determined primarily by the fusion power and tritium breeding requirements. The systems code optimization for COE has led to B=5.7 T, $\langle n \rangle = 3.58 \cdot 10^{20}$ m^{-3}, $\langle T \rangle$=5.73 keV and β=5% at the operating point, where B is the magnetic field on axis, $\langle n \rangle$ is the average ion density, $\langle T \rangle$ is the density weighted average temperature and β is the ratio of the average plasma pressure to the vacuum magnetic pressure.

The configuration is a member of the family whose magnetic spectrum has a toroidally quasi-symmetric magnetic structure with small but non-negligible mirror and helical terms [2]. As noted earlier, QA allows us to developed configurations with smaller aspect ratios. Allowing some selected non-axially symmetric components in a QA stellarator helps move secondary ripple wells away from regions where particles are toroidally deeply trapped so that the loss due to ripple trapping is reduced.

Figure 1 shows the LCMS in four equally spaced toroidal angles. One notices the presence of triangular and square components in its shape, resulting in the distinctive bullet shaped section at the half period. The general shape is much like that of the NCSX [3]. Figure 2 shows the contours of magnetic field strength as functions of normalized toroidal and poloidal angles on a flux surface half-way in the radial coordinate of the normalized toroidal flux. In general the contours show the underlining quasi-axisymmetry as the lines are running in the toroidal direction in most poloidal angles. But one will also observe deviations from this quasi-symmetry, particularly on the inboard, high field side of the torus.

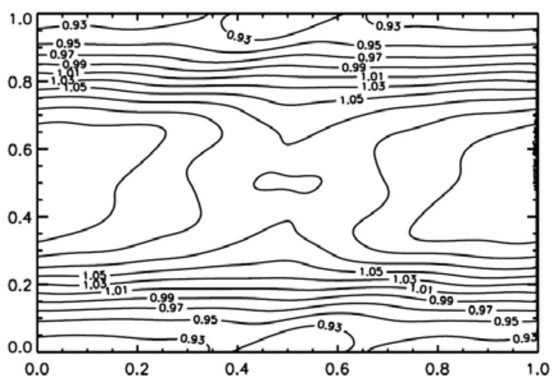

Fig. 2. Contours of the magnetic field strength on the r/a=0.7 surface displayed in the coordinate of normalized toroidal (horizontal) and poloidal (vertical) angles showing both the quasi-axisymmetry and the deviation from it of the baseline configuration.

II.B. Plasma Equilibrium and MHD Stability

In fig. 3 are the rotational transform profiles from an equilibrium with a pressure profile $\sim(1-x^2)^{1.2}$, where x is the normalized radius. Both that due only to the plasma shaping using external coils and that including the internal contribution from the plasma current are shown. The external transform ranges from ~0.4 to ~0.5 for the full donut and the total transform rises to ~0.7 near the edge. The plasma current which is entirely from bootstrap is ~3.5 MA. The flux surfaces calculated by the PIES code [4] which does not presuppose the existence of nested tori are illustrated in fig. 4. Field lines are followed to trace out separatrix near rational surfaces. It is seen that although the total rotational transform crosses 3/5 and 3/6 resonances, the flux surfaces maintain good integrity throughout the entire plasma volume with relatively small flux losses.

Our MHD stability analyses have shown that the configuration is robustly stable to the vertical modes due to the large amount of external transform relative to the total so that control coils normally required in tokamaks will not be needed here. The configuration is also stable to the ideal external kinks up to at least 4% β without a close-fitting wall. The calculation of the kink stability here uses the same algorithm and assumptions as those used in the design of NCSX [5]. The calculation for the ballooning modes indicates that the configuration is stable at ~5%. Here we are benefited by the presence of the mirror term in the spectrum which helps improve the stability against ballooning. We note here that recent results from W7AS [6] and LHD [7] showed that ideal linear MHD stability limits were surpassed in experiments. While there is much to be understood in the stability β limit in stellarators, we have designed the configuration in a more conservative fashion. If the plasma in our configuration turns out to be more stable than the calculations here indicated, some relaxation of the shaping constraints may be made which could potentially lead to simpler LCMS and hence the coils. These will further lead to improvement in plant complexity and reduction in COE.

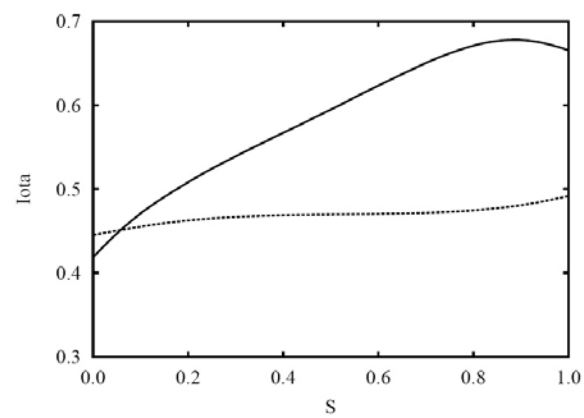

Fig. 3. The rotational transform of the baseline plasma, both external (dotted) and total (solid), as function of the normalized toroidal flux.

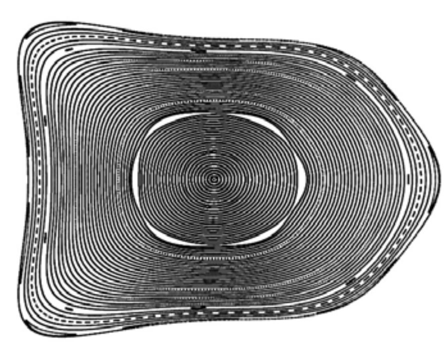

Fig. 4. Poincaré plot from PIES calculation at 5% β showing the quality of the flux surface.

The plasma equilibrium and the internal rotational transform depend upon the details of the plasma current distribution and the details of the density and temperature profiles. Sensitivity studies by varying these profiles show that the configuration is robust, both in the flux surface quality and the MHD stability [8].

II.C. Transport and Confinement

As discussed earlier, the magnetic field structure, although mostly quasi-axially symmetric, has certain distinctive features in the spectrum. In a QAS, the connection length between good and bad curvature regions is long so that particle transport is sensitive to the ripple well distribution along field lines. The small mirror component (1-2%) together with the side-band helical term reduce the secondary wells along the path of the toroidally trapped particles, thereby reducing transport losses, as confirmed by the analysis of the ∇B drifts [9]. The secondary wells are either shallow or closer to the inboard or outboard where particles with turning points inside the ripple wells average over smaller gradient drifts. We note that the presence of secondary ripples is unavoidable because we demand external fields to generate desirable rotational transform and to shape plasma for MHD stability. However, it is possible to reduce the harm caused by secondary ripple wells by preferentially selecting or biasing the components in the magnetic spectrum. In our baseline configuration the effective helical ripple, ε-eff, is calculated to be < 0.6% in the plasma so that we expect the neo-classical loss, which is proportional to $(\varepsilon\text{-eff})^{1.5}$ will be small when compared to the anomalous loss. Using the ISS95 scaling [10],

$$\boldsymbol{\tau(s) = 0.256 a_m^{2.21} R_m^{0.65} P_{MW}^{-0.59} n_{20}^{0.51} B_T^{0.83} \iota^{0.4}} \qquad (1)$$

one finds that the energy confinement time needed for power balance is typically ~1 s with an enhancement H factor of ~1.5. In (1) τ is the energy confinement time, a the average minor radius, P the absorbed power, n the line averaged electron density, B the magnetic field on axis and ι the rotational transform at the two-thirds way out.

A more important consideration in QA configurations is the loss of α particles as its not only affects the power balance but also drives the diverter and protective plate design. Our configuration is designed to minimize the collisionless loss by optimizing the secondary ripple well distribution. It is also optimized to reduce the collisional loss by increasing the collisionality of the α with background particles. The operating temperature and density are chosen such that the baseline plasma has high density and high magnetic field intensity consistent with the density limit experimentally observed and the maximum allowable field in the superconducting coils. The configuration has an α energy loss fraction ~5% based on a Monte Carlo calculation in which the statistical error is ~15%.

III. DESIGN OF BASELINE COILS

The baseline coils for ARIES-CS consist of a set of modular coils providing the main magnetic field and a set of planer, circular poloidal field coils providing the position control for changes in plasma pressure. A picture showing the modular coils are given in fig. 5. The modular coils follow stellarator symmetry, with 3 distinctive coils per half-period for a total of 18 coils over the three field periods. The coils are designed to be sufficiently far from LCMS to reduce the ripple due to the coil discreteness and to allow rooms for blanket and shielding, yet at the same time they are still close enough so that the multipolar fields are available for shaping the plasma without causing excessive bending and twisting of the coils. Indeed, one of the most important figures of merit for optimizing stellarator power plant is the coil aspect ratio, R/Δ_{min}(C-P), defined in section I. Too large a coil aspect ratio leads to a large power plant because of the shield/blanket thickness required for tritium breeding and coil protection. Too small a ratio leads to an overly complex coil geometry since the high order field harmonics decay rapidly. The baseline coils are designed to have a coil aspect ratio 5.9 and coil to coil separation ratio 10. The maximum field in the coil winding pack is estimated to be ~15 T when the field strength at the magnetic axis is 5.7 T. This high peak field demands the use of Nb_3Sn superconductor to be the current carrying material. The currents in the coils are allowed to be different but they are all ≤10 MA, giving a winding current density of ~90 MA/m^2.

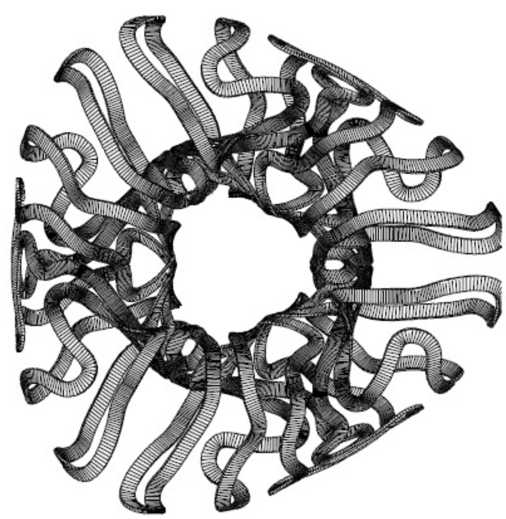

Fig. 5. Plane view of modular coils for the baseline design.

IV. SUMMARY AND CONCLUSIONS

We have developed stellarator configurations that meet the fundamental requirements to be the core of compact fusion power plants. These configurations have small aspect ratios, high equilibrium and MHD stability beta and good confinement for both thermal and energetic particles. Coil design optimization has also been carried out to make sure these plasma configurations can be realized. The particular plasma configuration selected as the baseline for the systems and power plant studies has an aspect ratio of 4.5 and the coil configuration has 18 modular coils and four sets of PF coils with coil aspect ratio ~6, leading to a 1 GW power plant with R=7.75 m.

The baseline plasma configuration is a member of a broader family that appears to possess attractive properties for compact stellarator power plants and whose potential and design tradeoffs have yet to be fully explored. Perhaps an equally important outcome of our studies is the discovery of the richness of QA configurations that offers us many different possibilities in designing future devices. On the other hand, despite the progress we've made, there is still need for reducing the α loss and making the flux surfaces more robust. In addition, there are rooms for further coil design optimization and improvement. We've found in many occasions that the design optimization would have to be compromised due to various imposed constraints. With the new QA device NCSX now under construction, experiments in coming years should help establish a physics data base that will further clarify the design requirements and will enable us to better design future QA stellarators for compact fusion power plant applications.

ACKNOWLEDGMENTS

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