Stellarator Research Opportunities

A report of the National Stellarator Coordinating Committee [1]

June 20th, 2017

This document is the product of a stellarator community workshop, organized by the National Stellarator Coordinating Committee and referred to as Stellcon, that was held in Cambridge, Massachusetts in February 2016, hosted by MIT. The workshop was widely advertised, and was attended by 40 scientists from 12 different institutions including national labs, universities and private industry, as well as a representative from the Department of Energy. The final section of this document describes areas of community wide consensus that were developed as a result of the discussions held at that workshop. Areas where further study would be helpful to generate a consensus path forward for the US stellarator program are also discussed.

The program outlined in this document is directly responsive to many of the strategic priorities of FES as articulated in "Fusion Energy Sciences: A Ten-Year Perspective (2015-2025)" [2]. The natural disruption immunity of the stellarator directly addresses "Elimination of transient events that can be deleterious to toroidal fusion plasma confinement devices" an area of critical importance for the U.S. fusion energy sciences enterprise over the next decade. Another critical area of research "Strengthening our partnerships with international research facilities," is being significantly advanced on the W7-X stellarator in Germany and serves as a test-bed for development of successful international collaboration on ITER. This report also outlines how materials science as it relates to plasma and fusion sciences, another critical research area, can be carried out effectively in a stellarator. Additionally, significant advances along two of the Research Directions outlined in the report; "Burning Plasma Science: Foundations - Next-generation research capabilities", and "Burning Plasma Science: Long pulse - Sustainment of Long-Pulse Plasma Equilibria" are proposed.

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[1] See appendix A for the membership of the NSCC at the time of completion of this report

[2] http://science.energy.gov/~/media/fes/pdf/program-documents/FES_A_Ten-Year_Perspective_2015-2025.pdf

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1. Executive Summary

The stellarator offers ready solutions to critical challenges for toroidal confinement fusion: it provides a steady-state, disruption-free reactor concept with minimal power requirements for plasma sustainment. The stellarator concept has undergone a rebirth in recent years as a result of major advances in theoretical understanding, the advent of computational capabilities, and experimental research that have made predictive understanding of many aspects of three dimensional magnetic confinement systems a reality. As a result of these advances stellarators are at the forefront of plasma physics research. The configurational flexibility afforded by the removal of the toroidal symmetry constraint opens up new physics regimes. It allows us to test our understanding of symmetry effects on plasma confinement and to produce the most physics-optimized fusion configuration yet conceived.

Historically, stellarators lagged behind tokamaks due to relatively poor neoclassical confinement. Groundbreaking optimized designs from the 1980's, first demonstrated on the W7-AS in Garching, Germany and then on the quasi-helically symmetric HSX device in Madison, Wisconsin demonstrated that neoclassical optimization improves the confinement of stellarators up to a level similar to tokamaks. The remarkable success of the initial 2016 campaign on the W7-X stellarator at IPP-Greifswald, the world's first large neoclassically-optimized stellarator, is the most recent advance on the path to a viable solution to the problem of maintaining fusion in steady-state. Progress toward steady-state (~30 min) confinement of high performance plasmas in W7-X in coming years will validate our understanding of optimized helical confinement and will establish the stellarator as a serious fusion reactor candidate.

The US has played an important role in the development of optimized stellarators. From both a theoretical [1,2] and an experimental [3,4] point of view, the US has been at the forefront of stellarator design by developing of a type of optimized configuration called quasi-symmetry. Quasi-symmetry (QS) is a hidden underlying symmetry property that leads to drift trajectories similar to those in symmetric configurations when viewed in an appropriate coordinate system. Quasi-symmetry is crucial complement to the approach taken in W7-X which is based on quasi-omnigeneity (these concepts are described within this report). Quasi-symmetry is topologically isomorphic to the tokamak, such that the large understanding accrued from the tokamak should transfer to QS stellarators. QS stellarators also allow large plasma flow velocity, which is important for achieving high confinement regimes, and introduced size flexibility. Currently, the US lacks any large-scale effort in this area. Given the exciting initial results of W7-X, and the opportunities presented by QS, a renewed US stellarator program is therefore timely.

1.1. US partnership on W7-X

OFES support for the US partnership on W7-X has made the US team (PPPL, ORNL, LANL, U. Wisconsin, Auburn U., MIT, Xantho Technologies) a key partner in W7-X, with major investments in configuration control, diagnostics, and divertor components, as well as strong participation in the research program. The primary goal of the upcoming W7-X campaign (OP1.2) will be qualification of the island divertor concept, using inertially-cooled Plasma Facing Components (PFCs). OP1.2 will be followed by an extended shutdown to install the water-cooled island divertor hardware required for OP2 operation (2019) with confinement of high performance plasmas for ~30 minutes.

W7-X is expected to show reduced neoclassical transport at high plasma pressure. Ongoing US diagnostic/modelling efforts give the US team a significant role in this research area. Key W7-X design choices led to a tight coupling of the main confinement configuration with the divertor structure. Residual pressure-driven plasma currents can change the edge rotational transform sufficiently to cause diverted heat fluxes to miss the geometrically-resonant, armored island divertor structure. Configuration control during the heating sequence is thus essential for long-pulse operation of W7-X. Investigation of these issues is the central task of the US divertor scraper project and the associated edge plasma/PMI program.

Strengthened funding for the US partnership on W7-X, building on the significant US presence there, is the quickest way to allow more US researchers (senior researchers, post-docs and graduate students) to pursue activities of strategic benefit to the US in steady-state pellet fuelling, configuration control, turbulent transport, and high-heat flux PMI.

1.2. Development of improved stellarator reactor concepts

The US stellarator theory program, which presently comprises work at PPPL, ORNL, U. Wisconsin, Auburn U., Columbia, New York University, and U. Maryland, has for some decades made major contributions to toroidal confinement physics (equilibrium, stability and transport in 3D systems), automated stellarator design optimization, and analysis of stellarator experiments. This community has also produced computational tools, which are used in stellarator research around the world.With its established expertise, the US stellarator community now has an opportunity to leverage its capabilities along with the lessons-learned from the W7-X device to develop advanced stellarator designs, which balance physics performance with robust engineering. W7-X was designed nearly 30 years ago, and there have since been numerous conceptual advances in stellarator physics and engineering that can be the basis of a world leading stellarator research program. Improved understanding from both tokamaks and stellarators enables more comprehensive optimization and identification of new approaches.

US researchers, working as a team with international collaborators, should undertake an integrated stellarator optimization initiative to test innovations that can dramatically improve plasma confinement and provide a more robust basis for the development of fusion energy. Optimized designs would combine features of neoclassically optimized high-β concepts with opportunities for new conceptual advances such as:

- Reduction of turbulent transport.
- Use of QS to improve high-energy particle confinement and reduce impurity accumulation.
- Simplified coils/support structures, with reduced non-planar distortion and increased access.
- A robust divertor system that is insensitive to the details of the equilibrium.

A successful optimization and design effort based on the above steps will lead directly to plans for enhanced research facilities.

1.3. A Forward-looking US Experimental Program

The US domestic stellarator experimental program presently comprises four university experiments. HSX (U. Wisconsin) explores the effect of quasi-symmetry on core plasma transport, flows and turbulence and in edge transport/PMI studies. The CTH stellarator (Auburn U.) studies equilibrium and stability of current-carrying stellarator plasmas. HiDRA at the U. Illinois focuses on plasma-materials interactions. The CNT device at Columbia U. is studying electron Bernstein wave heating.

To be competitive on the international scale, and greatly accelerate the pace towards stellarator fusion, a reinvigorated experimental U.S. stellarator program, based on the concept of quasi-symmetry, is needed to evaluate new optimization innovations.

In the near term, a mid-scale US facility that advances quasi-symmetry to more fusion-relevant plasma regimes and tests the non-resonant divertor concept will provide fundamental new information and guidance to the US fusion program. Such an experiment will be unique worldwide. It will also be coupled to a large experimental initiative, with the two facilities forming a complementary set.

A quasi-symmetric U.S. stellarator facility, comparable in impact with its major contemporaries like W7-X and JT60-SA, is necessary to generate information that can influence design decisions on the path toward fusion energy in the ITER era. The current lack of a clear path to steady-state, disruption-free fusion systems makes it a matter of some urgency to exploit these promising solutions as rapidly as possible. A national concept design effort for these two new facilities, the first step in translating innovative ideas into practical experimental devices, needs to start now.

References - Executive Summary

[1] A. H. Boozer, *Phys. Fluids* **26**, 496 (1983)
[2] H. E. Mynick, T. K. Chu, and A. H. Boozer, *Phys. Rev. Lett.* **48**, 322 (1982).
[3] F.S.B. Anderson, et al. , *Fusion Technology* **27**, 273 Suppl. S (1995)
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2. Important topics for advanced stellarators

This section lays out the important physics questions and conceptual developments in the area of optimized stellarators that motivate and help define an exciting outline for a stellarator physics program. The topics presented in this section describe advances in understanding that have arisen since W7-X and HSX were designed, which can form the scientific basis for such a renewed US program.

In particular, Section 2 covers the following topics:

1. Neoclassically optimized magnetic configurations
   - This section describes the various strategies for neoclassical optimization of stellarators, outlines their features, and lists some of their individual advantages
2. Turbulence and transport optimization
   - Ways to optimize turbulent transport in with 3D shaping are described – an exciting new prospect for stellarators
3. 3D Divertors
   - The various types of stellarator divertor geometries are described and the issues associated with each type of system are discussed
4. Plasma-Materials Interactions in 3D systems
   - Issues associated with Plasma Materials interactions that are specific to three dimensional confinement are discussed – advantages of inherent steady state systems for PMI research are enumerated
5. Fast particle confinement
   - Good confinement of the high-energy alpha particles released from deuterium-tritium fusion is required of all magnetic confinement fusion reactors. The special considerations for achieving good fast particle confinement in stellarators are presented.
6. Equilibrium and Stability at High-β
   - An area where stellarators excel is in the stable achievement of high plasma pressure (β). There are several areas where further research could enhance this characteristic further.
7. Impurity confinement and accumulation
   - A potential issue of maintaining a stellarator in steady-state is the accumulation of impurities that is predicted by neoclassical theory. Operating regimes without impurity accumulation have been observed. The current state of knowledge is summarized.
8. Reactor Issues
   - 3D system face specific challenges when the incorporation of reactor technologies is considered. This section lists those challenges and summarizes the state of stellarator reactor design.
9. Design Improvements and Coil Simplification
   - Recent efforts focused on developing simpler magnet designs and improved theoretical understanding of the constraints on coil location are summarized.

2.1. Neoclassically optimized magnetic configurations

There are three types of stellarators that may lead to reactor designs, with the types distinguished by the method of obtaining confinement for trapped particles. Passing particles are well confined in all stellarator types. These three types are illustrated in Figure 1. Quasi-symmetry [1] is one concept for obtaining trapped particle confinement. Precise quasi-symmetry is defined by the magnetic field strength having the form $B(l+L) = B(l)$ along each field line, where l is the distance and L is a constant along the line. Quasi-symmetric stellarators are of two types: quasi-helically symmetric (QH) [2,3] and quasi-axisymmetric (QA) [4,5].

Quasi-omnigeneity (QO) is a more general concept for obtaining trapped particle confinement [6,7,8] and has quasi-symmetry as a special case. QO confinement is based on conservation of the action $J(\psi_t, \alpha, \mu, H) \equiv \oint m v_{||} dl$, where the magnetic field is written as $\vec{B} = \vec{\nabla}\psi_t \times \vec{\nabla}\alpha$ where ψ_t is the toroidal magnetic flux enclosed by a magnetic surface. The quantity $\alpha \equiv \theta - \iota\varphi$ is constant along a magnetic field line; θ and φ are poloidal and toroidal angles, and $\iota(\psi_t) = 1/q$ is the rotational transform. The velocity $v_{||}$ of the particle along the magnetic field line is given by the energy, $H = \frac{1}{2} m v_{||}^2 + \mu B + q\Phi$, with both the energy H and the magnetic moment μ held constant. A stellarator would be omnigeneous if $\partial J/\partial\alpha$ were zero, for then J conservation would imply that trapped particles could have no ψ_t drift in their banana orbits, their motion between turning points [9]. Neither perfect symmetry nor perfect omnigeneity are achievable in a stellarator [10], but both have been approximated as quasi-symmetry and quasi-omnigeneity in stellarator designs.

2.1.1. Quasi-Helical Symmetry

A quasi-helically (QH) symmetric stellarator has |B|=constant along a helical trajectory. QH stellarators have a large effective rotational transform, $\iota_{eff} \equiv \iota - N_p$, where N_p is the number of periods. In practical stellarator designs the rotational transform $\iota \approx 0.2N_p$. The banana width of trapped particles and the two types of pressure-driven parallel currents scale as $1/\iota_{eff}$. The pressure-driven currents are the Pfirsch-Schlüter current, which has zero average over a magnetic surface, and the bootstrap current, which is constant on a magnetic surface. Consequently both are small, $\iota/\iota_{eff} \sim -1/4$, and of the opposite sign from quasi-axisymmetry, therefore the bootstrap current reduces ι. Neoclassical tearing modes are stabilized when $d\iota/dr < 0$.

QH stellarators are apparently uniquely able to confine collisionless particles in a way that is arbitrarily close to the way they are confined in exact symmetry [10]; the difference scales as $\sim 1/N_p^3$. Three implications are that QH stellarators can have (1) arbitrarily good confinement of energetic

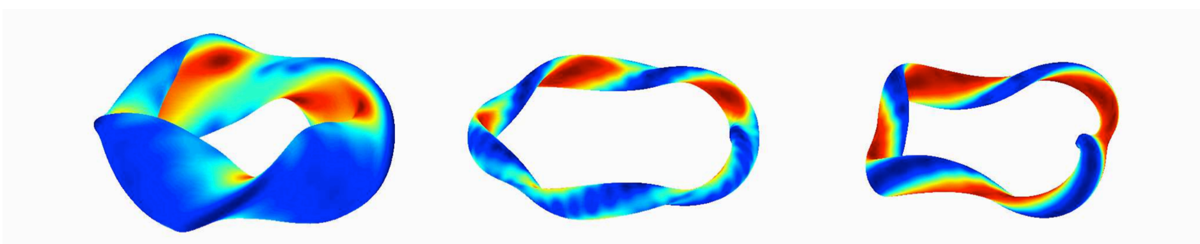


Figure 1 *Equilbria from a) NCSX, a quasi-axisymmetric plasma, b) W7-X, a quasi-omnigenous plasma, and c) HSX, a quasi-helically symmetric plasma. In all cases the color represents |B| on a flux surface.*

particles, (2) the neoclassical transport can be made small and be consistent with impurity screening, and (3) a component of plasma flow is weakly damped, which may be important for reducing microturbulent transport. Nevertheless, the achievement of quasi-helical symmetry requires a relatively large distortion of the shape from helical symmetry, $\sim 1/N_p$.

QH stellarators appear to require a larger aspect ratio than QA stellarators. A typical aspect ratio of a stellarator is $2N_p$. When the aspect ratio per period is tighter than two, optimization of physics across the plasma cross section becomes difficult.

2.1.2. Quasi-Axisymmetry

Quasi-axisymmetric devices have drift trajectories that look like those in a tokamak. Axisymmetry is a special case of quasi-axisymmetry. It is possible to add a small non-axisymmetric field to a tokamak in such a way that quasi-axisymmetry is preserved. In QA configurations, the bootstrap current increases the rotational transform, as in tokamaks. The 3D shaping can be optimized to enhance

MHD stability beyond that of tokamaks. A full range of quasi-axisymmetric configurations are possible, from a perturbed tokamak to a device where most of the rotational transform is produced by the 3D geometry, rather than the toroidal plasma current. This is a unique property of quasi-axisymmetric configurations.

No sharp distinction in design space exists between tokamaks and QA stellarators, and the principles of quasi-symmetry could be used to design more effective non-axisymmetric control coils for tokamaks. The term stellarator implies that a large fraction of the rotational transform is external, $\iota_{ext} \approx N_p \Delta_s^2$, where Δ_s is the amplitude of the non-axisymmetric plasma shaping. The non-axisymmetric shaping changes the minor radius roughly as $r = (1 \pm \Delta_s) r_0$, with r_0 a constant. In QA stellarators the shaping is strong $\Delta_s \sim 0.5$.

A strong centering force, $\propto \iota_{ext}$, arises if the plasma moves away from the magnetic axis of the external rotational transform. In all stellarators, this centering force is sufficiently strong that a disruption involving loss of position control is impossible. Neoclassical tearing modes are stabilized when $d\iota/dr > 0$.

2.1.3. Quasi-Omnigeneity

A quasi-symmetric stellarator, $B(l+L) = B(l)$, also satisfies quasi-omnigeneity. Of the three types of of neoclassically optimized stellarators, only the QO stellarator, W7-X, is being explored at a large scale, and W7-X has fundamentally different properties than quasi-symmetric stellarators.

W7-X type QO stellarators have comparable $n=0$ and $n=N_p$ toroidal components, so the banana orbits have no systematic radial shift proportional to $v_{||}$. This feature can be used to make the bootstrap current arbitrarily small. The Pfirsch-Schlüter current can also be greatly reduced from its magnitude in QA stellarators. When both the bootstrap and Pfirsch-Schlüter current are negligible, the shapes of the flux surfaces and the rotational transform become independent of $\beta \equiv 2\mu_0 p/B^2$.

W7-X is designed so that either a 6/5, 5/5, or 4/5 island chain at the edge defines the divertor [11,12]. The edge transform must be accurately held to ι = 1.2, 1, or 0.8, which makes the W7-X divertor a resonant design.

2.1.4. Common features

All three stellarator types require careful control of local shear and curvature to achieve high-β stability and acceptable microturbulent transport. All three types can be designed to have the important features of stellarators: (1) External control rather than plasma self-organization, which allows accurate computer design to speed the development of fusion energy. (2) Robust positional stability, which prevents tokamak-like disruptions. (3) No apparent limit on plasma density other than power balance. (4) Intrinsic MHD stability, including to neo-classical tearing modes. (5) A net plasma current that can be restricted to whatever level is required to avoid major runaway-electron issues. (6) An intrinsically steady-state magnetic configuration.

2.2. Turbulence & transport optimization

The level of turbulence in toroidal devices is strongly dependent on their shape, and through this, their magnetic field structure. In view of the strong dependence of the cost of fusion power on the level of turbulent transport, understanding and if possible reducing these levels is an issue of considerable importance. For this purpose, two powerful numerical tools for stellarators have emerged.

The first of these are gyrokinetic (gk) codes capable of simulating microinstabilities in the 3D toroidal equilibria of stellarators. Probably the earliest of these was FULL[13], a purely linear code, upgraded in 1999 [14] from a 2D (tokamak) code to be valid for 3D geometries. Since then a number of gk codes valid for 3D have become operative, including Vlasov codes such as GENE[15,16], GKV[17, 18], and GS2[19,20], and the global PIC codes EUTERPE[21] and most recently GTC[22]. All of these can do linear calculations. GENE, GKV & GTC can also do nonlinear calculations, and the first 2 have been used extensively for this purpose. Since GKV is not widely disseminated by its NIFS developers and a 3D GTC is very recent, GENE has been the principal workhorse for doing nonlinear simulations in 3D geometries. One fruitful line of application with these has been to investigate the sensitivity of plasma turbulence to the shape of the device, identifying key geometric parameters (such as the radial curvature or local shear), which affect the transport levels [23, 24, 25].

The second critical numerical tool is the stellarator optimization code, such as STELLOPT[26], originally developed during the NCSX design process. The gk studies led to the realization [27] that, in conjunction with STELLOPT, the numerical tools are now in hand to evolve stellarator (or tokamak) designs via shaping to ones with substantially reduced levels of turbulent transport, often without degrading the neoclassical transport, resulting in "turbulence-optimized" designs [27, 28, 29, 30], analogous to the "neoclassically-optimized" concepts which emerged in the 1980s and 1990s. A 2-step method was developed, step one being a STELLOPT optimization using a "proxy function" to rapidly estimate the transport level of a given configuration, and step-2 being a nonlinear GENE run to

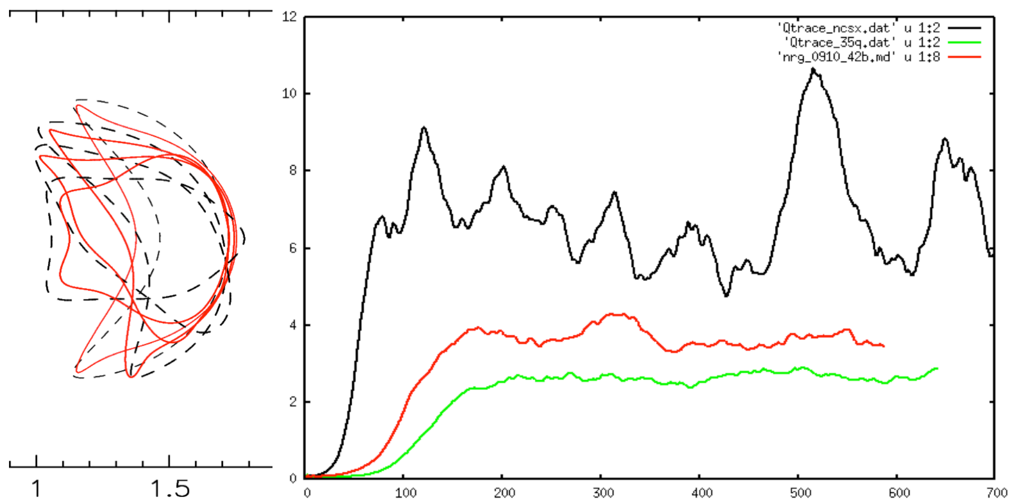


Figure 2 *(a) Poloidal cross-sections and (b)surface-averaged heat flux Q versus time from GENE simulations for NCSX and 2 turbulence-optimized configurations QA_40n (red) and QA_35q (green), showing the reduction in Q from NCSX by factors of 2 to 2.5.*

corroborate that the evolved configuration in fact has reduced transport. Most work to date has focused on optimizing for ion temperature gradient (ITG) turbulence [27,28,29], in each of the 3 main neoclassically-optimized stellarator designs (QA, QO/QI, & QH) (See, *e.g*., Figure 2) as well as in tokamaks, and most recently [30], for trapped-electron mode (TEM) turbulence on the HSX QH. Significantly, it has been found that the shape deformations found by this method to diminish the ITG turbulence also tend to reduce TEM as well as ETG turbulence, because of the related physics. The heat flux Q_{gk} from nonlinear gk runs has been reduced by factors of 2-3 based on these optimizations, an amount comparable to the improvement in going from L to H mode in tokamaks.

In coming years, further progress in turbulence-optimization may be achieved by pursuing this exploration in various directions:

2.2.1. Improving the method.

The method works only to the extent that the proxy function correctly correlates with Q_{gk}. The proxies thus far used have been mainly theory-based semi-analytic expressions, using the same geometric information as is given to the gk code for its run.

1) The analytic theory on which these are based could be improved, e.g, including critical gradients, & possible nonlinear effects, such as zonal flows & other saturation mechanisms (eg, [31]).

2) Considerable improvement may be achieved by instead making use of gyrokinetically-computed results in the STELLOPT optimization cycle. Linear gk runs are far faster than nonlinear ones, bringing optimizations based on growth rates within the capacity of current large computer clusters. Calculations of growth rates, along with quasilinear expressions for Q, ($Q_{ql} \sim \int dk\ \gamma/k_\perp^2$) within the optimizer may provide more accurate estimates of Q_{gk}, and without needing different expressions for differing turbulence channels or parameter regimes. Covering the extensive optimization parameter space of stellarators requires efficient, accurate quasilinear transport modeling; one research direction will be to ascertain whether the inclusion of linearly unstable eigenmodes in models [31] is sufficient to reproduce transport scaling reliably for different magnetic configurations and plasma profile characteristics.

3) As computing power increases toward exascale-size machines, the present 2-step method may be reduced to a single-step, with the nonlinear Q_{gk} being used as the figure of merit in the optimizer cycle.

4) Optimizing to maximize the critical gradient: If one assumes a device will operate near the critical gradient of the dominant instability, targeting this may give a better predictor of the level of transport than the approach up to now, which seeks to minimize Q at fixed gradient.

2.2.2. Comparative turbulence characteristics

Determination of the characteristics of the 4 classes of neoclassically-optimized devices (including tokamaks) should be carried out. Some studies comparing growth rates and heat fluxes across confinement classes have already been done. However, a more systematic, apples-to-apples assessment of whether some classes are better or worse at achieving low turbulent transport is needed, as one component of determining what an optimal reactor design should be. Regarding tokamaks versus stellarators generally, both linear and nonlinear simulations [20,24,25,29] indicate that the toroidal variation of a stellarator tends to break up the turbulent structure seen on the outboard side of tokamaks, resulting in reduced growth rates and more localized turbulence. Earlier results show that QAs and QOs have lower ITG and TEM growth rates than QHs, which are comparable to those in tokamaks (see Figure 3). However, recent [32,33] simulations indicate that the heat flux of HSX is comparable to other optimized stellarators, suggesting that nonlinear effects may play a larger role in QHs, and showing the need to include nonlinear effects in proxy functions. Some work [33,33] has already been done, and completing this program would constitute an important step. Comparisons have so far been done using only one design for each confinement class, and a wider exploration of the potential of each class for turbulent optimization should be pursued.

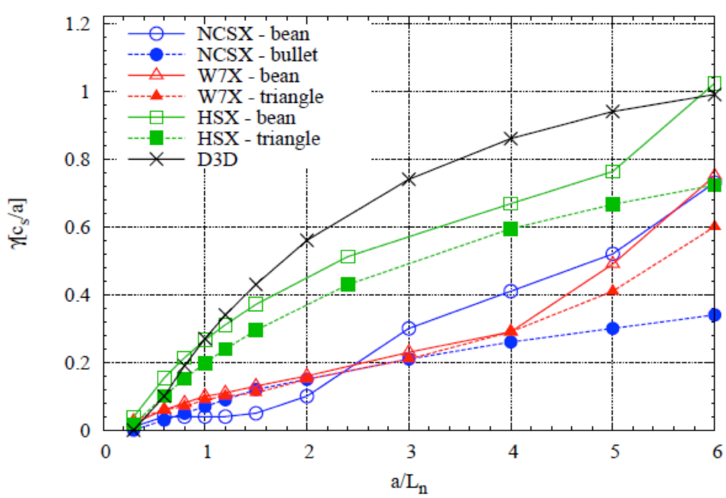


Figure 3 *(a) Growth rates[30] for TEM turbulence from GENE simulations for NCSX, W7X, HSX and D3D.*

2.2.3. Research needs and opportunities

The changes in turbulence with plasma shape, which are the basis for turbulent optimization, need to be tested experimentally, and compared with theoretical and numerical expectations. The present small set of operating stellarators makes finding suitable testbeds more difficult. The HSX stellarator is the most likely initial prospect for providing such tests, for TEM turbulence. W7X and LHD are also obvious choices, having ITG turbulence and being better diagnosed, but being much larger machines, getting such tests into the run schedules will be more difficult and take longer. Tokamaks with a good deal of shaping capability, such as TCV, could also provide useful tests. A QA experiment would fill an important gap here in our ability to assess the potential for this concept, and how it compares with the other classes.

2.3. Divertor design and optimization

2.3.1. Background

A fusion reactor requires a sub-system – called the divertor - which enables to exhaust helium as the ash of the fusion process, control plasma density and impurity levels and handle the heat exhaust from the plasma edge without overloading the material surfaces. The eventual heat and particle fluxes to the divertor material have to be controlled such that an acceptably long material lifetime is accomplished while enabling compatibility with good core confinement (density and impurity control). Tokamak experiments have almost exclusively settled on poloidal diverted plasma configurations and their variations [32, 33, 34] (see Figure 5). In this concept, the plasma material interface in the divertor is separated from the Last Closed Flux Surface (LCFS) by the divertor volume. This allows establishing a largely separated divertor plasma which can be used as a buffer plasma between plasma core and the material interface. This can yield several key advantages for stable and steady state operation of divertors:

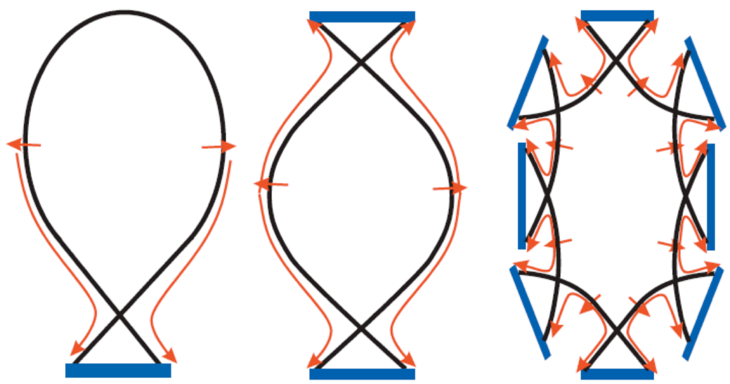


Figure 5 *Sketches of tokamak single-null and double-null configurations (left and middle), and a stellarator island divertor (right figure) [34]. The red arrows show the convective flow pathway in an ideal, simplified plasma boundary solution.*

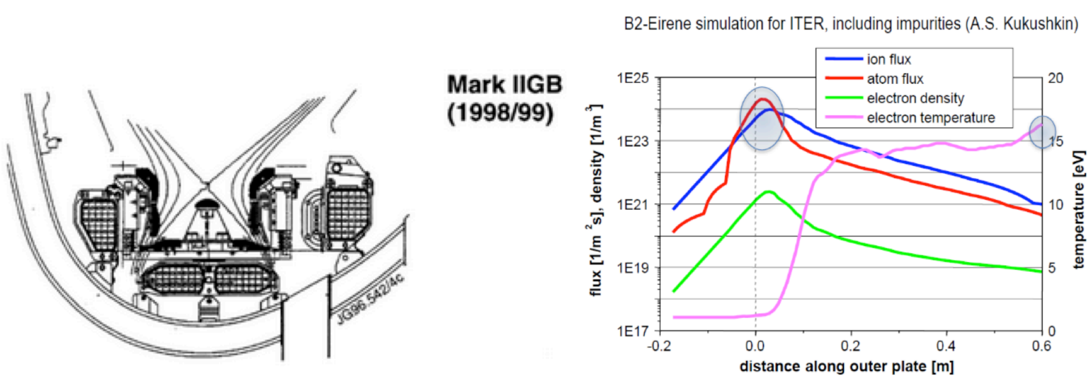


Figure 4 *Schematic of a tokamak divertor system (at the Joint European Torus JET) and simulated plasma parameters (SOLPS modeling) along the divertor plates*

1. Allowing for a finite parallel temperature gradient in the plasma pressure and temperature and volumetric power losses in the scrape-off-layer (SOL).
2. Spatial separation of the area of impurity production from plasma material interaction (PMI) and core, which reduces core contamination and provides a region where power can be dissipated by interaction with a substantial neutral gas pressure.
3. The ability to create a 'closed' divertor volume trap recycled neutral particles, increasing the neutral pressure in the pumping volume and increasing power and momentum loss in the SOL.

One main advantage of the diverted plasma shape in tokamaks is that it facilitates access to the high confinement H-mode. This bifurcate transition into H-mode is thought being caused by high local magnetic shear and strong shearing of the radial electric field, both induced by the poloidal divertor separatrix. One major disadvantage of the poloidal divertor configuration in tokamaks, however, is that the divertor heat and particle flux is focused into very narrow stripes. Given that reactors have to operate at high upstream density and temperature to match a viable fusion gain requirement, this introduces a severe challenge for the integrity of the divertor target materials. Hence, the requirements on the plasma parameters at the plasma material interface are determined by engineering limits, and are in the range of $Q_{max} \sim 5$ MW/m^2, with $T_e < 5$ eV and $n_e > 10^{21}$ m^{-3} at the strike point [35]. The latter requirement is to mitigate sputtering and material erosion in order to maintain a low impurity source level and to provide an acceptable material lifetime. Such parameters are achievable only by operating in edge plasma transport regimes with significant power and momentum losses in the flux tubes connected to the strike point in order to reduce the power and particle flux to the material surfaces. This has to be accomplished simultaneously with maintaining an upstream plasma pressure compatible with the fusion gain requirement. The so-called "detached" plasma regime is suitable to match this requirement as here the particle and heat flux are detached from the material surface and much of the energy being conducted towards the divertor plate is exhausted by in a volume recombination domain in front of the divertor targets. However, this regime involves highly non-linear atomic and molecular processes, which challenge the demand for a stable, non-transient and reliable divertor plasma state. Such stable, detached plasmas are not easy to obtain and maintain as seen in the development of radiative instabilities [36] in poloidal tokamak divertors. Thus, partial detachment is envisioned for ITER operation [37] avoiding the necessary control of the atomic and molecular processes in the divertor. Another issue with full detachment is that the particle flux and hence the neutralization region is detached from the pumping, i.e. mechanical exhaust system of the device. Hence, it is anticipated that other regions of the divertor must be in the high recycling regime (HRR), where the particle flux increases rapidly with upstream density, to establish the neutral pressure required for Helium pumping and density control [38].

Tokamak 'closed poloidal divertor' designs are believed to allow for the above requirements to be satisfied simultaneously in a reasonable fashion. The divertor can be detached at the strike point where the particle fluxes are highest, yet further along the plate T_e is larger and the plasma is in the high recycling regime, see Figure 4. Neutral particles from this region are guided into the divertor legs increasing power and momentum losses through plasma-neutral interactions and increasing pressure in the pumping volume. However, demonstration of such a combined detached (close to divertor leg) and HRR (outer SOL) regime in a stable fashion is challenging and hence a key element of present days divertor and plasma edge physics research.

2.3.2. Stellarator divertor systems

The configurational flexibility for optimized stellarators provide a large degree of freedom for design and optimization of custom-made divertor configurations. However, one challenge in stellarator edge science is to design a divertor magnetic structure, which is insensitive to effects of the confined plasma equilibrium. The resulting divertor structure then has to match the same functional criteria as outlined above for the tokamak poloidal divertor.

A variety of promising divertor concepts have been and are being tested on stellarator devices. They can be broadly separated into three categories

1. **Helical divertors** (e.g., LHD). The helical divertor is continuous, located between the helical coils and follows the helical geometry of the device. A large separation between plasma and the material interface is enabled, but very thin magnetic strike lines are found. They facilitate good baffling for improved neutral compression but yield highly localized heat fluxes. The helical divertor structure is bound to the field periodicity of the device and hence is ridged against changes of the internal magnetic equilibrium of the helical plasma core.
2. **Island divertors** (e.g., W7-AS, W7-X, CTH). The edge transform is configured to provide a resonant value at the plasma edge. A magnetic island is formed on this low order rational surface by matching harmonics in the magnetic field structure of the device. This magnetic island then is intersected with plasma facing components (PFCs) forming the Island Divertor. This magnetic structure of this divertor concept is sensitive to the actual value of the rotational transform defined by the 3-D equilibrium. This might have adverse consequences for stable divertor operation, in particular with regard to the heat flux localization and neutral pumping stability.
3. **Intrinsic divertor** (e.g., NCSX, HSX). Generally stellarator systems have well defined exit pathways for field lines from the LCFS. The resulting interaction with the vessel wall will have the form of intrinsic 'stripes' of magnetic field line intersections when mapped from the LCFS to a vacuum vessel designed as magnetic surface offset from the LCFS and conformal to it. These patterns are determined by the shape of the LFCS [39,40] and for fixed field periodicity of the device are not sensitive to the edge transform and hence conceptually less prone to adverse effects from 3D equilibrium effects. This generic stellarator feature represents an excellent and worldwide unique opportunity for innovation on divertor concepts for stellarators, which are insensitive to effects of the plasma equilibrium and finite pressure effects.

2.3.3. Research needs and opportunities

At a high level, the stellarator divertor configurations have significant similarities with tokamak poloidal divertor (see Fig. 1, right side for the island divertor). However the boundary between closed and open field lines in a 3D system is not sharp. In general, there is a stochastic layer at the plasma edge, even in the case of an island divertor. In view of the requirements for the stellarator divertor, essentially the same requirements described above also apply. 3D divertor systems however have several potential advantages:

- The diverting field structure is inherent to the confining magnetic field and does not require additional coils – like the poloidal field coils to form the poloidal X-point.
- Island divertor systems feature significantly longer connection lengths than poloidal divertors in tokamaks [34], potentially allowing for increased parallel temperature gradients and a larger perpendicular to parallel transport ratio. This beneficial feature is capable to reduce the heat flux peaking on the divertor targets, and retaining impurities in the edge islands, reducing core contamination.
- Stellarators have a great flexibility in configuration space, allowing the divertor to be optimized along with the magnetic field. The magnetic field topology for both, the Island Divertor as well as the Intrinsic Divertor is always dependent on the high order resonant fields of the confining magnetic coil system. In regions of the plasma edge where the LCFS features a sharp edge in the poloidal projection, field lines travel tangentially along this sharp edge. This tangential movement makes these field line is prone to perturbation by higher order resonant magnetic fields and hence, this sharp edge region of the LCFS is prone to formation of stochastic field domains. As a consequence, the interaction with the divertor targets is defined in the island divertor by the

intersection of the island with these targets plate and for the intrinsic divertor by the pathway of the stochastic field line from the sharp edge on the LCFS. They both enable definition of appropriate divertor hardware structures, which need to be tested in view of their feasibility as reactor relevant divertor concept. Comparison between these two promising divertor concepts would be enabled by comparison between a U.S. stellarator experiments with flexible divertor configurations to realize both divertor topologies as part of the strong collaborative work at W7-X including the Helical Divertor at LHD as well explored reference.

- The lack of a "hard" density limit allows for high-density operation, possibly easing the requirements on reducing the temperature between the LCFS and the divertor and for establishing a buffer plasma which enables low target heat flux and at the same time good density and impurity control.

There are several aspects of stellarator plasmas, which could be leveraged, but more research is required to gauge the credibility of this conceptually beneficial aspects:

- Stellarators do not have large edge instabilities, such as tokamak ELMs in many operational scenarios, and large ELMs will not be allowable in a reactor [41] The plan for mitigation of ELMs using 3D fields on ITER is a significant additional cost. However, it has to be investigated if high confinement regimes in stellarators are free of ELMs [42].
- The power channel width scaling in tokamaks leads to very narrow flux patterns predicted for ITER and DEMO [43]. The scaling in stellarators remains an open question, but the long connection length in the island divertor configuration is promising for increasing the heat flux width. Research to solidify understanding of the underlying physics is necessary in order to use the peak heat flux and the heat flux channel width as parameter for divertor optimization.
- 3D systems have increased momentum loss due to effects such as counter-streaming flows [44]. This has the advantage of facilitating detachment, however W7-AS and LHD do not exhibit the high-recycling regime [34], which is important to establish an integrated divertor solution with reliable neutral pumping. The additional momentum loss usually enables detachment at high temperatures (~10eV), which results in a completely different downstream condition for detached plasmas in stellarators compared to tokamaks (<1eV). Understanding the consequences of the plasma edge strucrure on the 3-D momentum balance is hence a critical area of research to understand and define stellarator divertors in reactor relevant, low temperature, high density divertor regimes.

The following areas are potential issues for stellarator divertors that require research:

- Island divertors may require active control, as toroidal current evolution can change the edge transform. This sensitivity to the edge transform (and thus bootstrap current and current drive) is a disadvantage of this configuration. The optimal range of magnetic shear must also be explored further: low shear increases risk of core islands while high shear results in a narrow edge island chain and increased stochasticity. On the other hand, the detailed shape and position of the island in the neutral particle region offers attractive additional means for control of the edge plasma. The trade-offs between detrimental and beneficial aspects of resonant islands in the edge of a stellarator plasma is an active and attractive field of research.
- Stellarators in ion-root confinement have a neoclassical impurity pinch that is not present in tokamaks [45]. This means that the impurity production at the targets must be reduced or impurity screening must be introduced (e.g., as observed in island divertors or through positive radial electric fields in this island domain). As stellarators target reduced neoclassical transport as an

optimization parameter, optimized devices such as W7-X may be dominated by turbulent transport, possibly mitigating this issue.

- Access to the HRR. As mentioned above, existing stellarators have not observed the HRR, nor is it expected from edge simulations. However, modeling using the EMC3-EIRENE code predicts the HRR in W7-X due to the larger edge islands and the increased distance between counter streaming flows [34].
- The lack of a sharp transition from open to closed field lines may result in a change from ion orbit loss (as in a tokamak) to large electron fluxes across the stochastic layer. The former effect is thought to provide a mechanism for the large edge electric field in a tokamak H-mode [46]. However, long-range flow formation in stellarators [47] might generate the same level of radial electric field shear as considered to be required for H-mode access in tokamaks.
- In general, the 3D nature of the divertor system increases the complexity. Systems for maintenance and replacement may be more elaborate.
- The ability to design a system with a constant angle of incidence across the divertor plates may be very challenging. This can lead to variations in the flux along the flux stripes. In W7-AS toroidally and poloidally localized areas of the divertor received high flux even at densities that caused most of the divertor to detach [48]. This is in contrast to tokamaks, where detachment occurs in the flux tubes where the heat flux is the highest. This may also be an issue for achieving the T_e and n_e requirements for a reactor as described above.

To date no stellarator has been optimized with full consideration of divertor and edge transport as target parameters. The only active code able to address 3D edge transport in stellarators, EMC3-EIRENE, became available after the design of the modern stellarators optimized for neoclassical transport (e.g., HSX, W7-X, NCSX). Hence, a fully integrated optimization, which includes optimization of the divertor system is an innovative and not yet systematically tackled research goal of stellarator edge physics. The existing expertise in the U.S. program on 3-D equilibrium modeling, plasma core optimization and stellarator divertor physics makes the U.S. program a viable potential leader in this field of great generic relevance to the success of stellarator reactors.

This introduction and description of research needs can be exploited and addressed by targeted research in two main innovative areas.

Testing of divertor concepts

The existing domestic facilities can be used to explore basic divertor concepts, edge transport physics, and innovative materials within their parameter range. Many of the key aspects of a reactor-relevant divertor design will be tested on W7-X. In particular, integrated core-edge solutions, divertor heat flux width, ELMs and H-mode, access to the HHF regime and detached operation. However, no device exists in the worldwide stellarator program, which provides capability to test integration of each of the promising divertor concepts (e.g. the island divertor and intrinsic divertor) with plasma core configurations optimized for reduced neoclassical and anomalous transport. Consequently, one key recommendation of this report is to design and build such a device on the medium scale to assert leadership in development of integrated stellarator optimization.

Model validation and advancement

Such an activity necessitates availability of a suitable plasma edge and plasma-material-interaction modeling tool. The state of the art model available is the EMC3-EIRENE fluid plasma and kinetic neutral transport code. Validation of EMC3-EIRENE simulations on W7-X will be an important test of the code. Many aspects of stellarator divertor physics have only been explored for limited parameters and configurations and require additional research and this research on W7-X will for the time being represent excellent opportunities to progress development of predictive capability and at the same time explore the Island Divertor regime at W7-X. Key research tasks include access to the HRR, divertor flux widths, and access and stability of detachment. The specific numerical methods used in EMC3 make inclusion of physics such as kinetic corrections, cross-field drifts, and access to detachment challenging. Ultimately, development of a new code incorporating additional physics would significantly boost stellarator edge modeling and optimization capability. In order to include a divertor design into an a stellarator optimization code such as STELLOPT, cost functions related to divertor and edge physics must be developed. Development of such predictive capability from full 3-D modeling, derivation of appropriate cost functions for integrated optimization schemes and validation of these approaches on existing and new stellarator devices is an attractive opportunity for the U.S. stellarator program and the recommendations made later in section 4.5 will describe appropriate activities to address this.

2.4. Plasma material interaction (PMI) issues in 3-D fusion systems

2.4.1. Stellarators as PMI laboratories

Stellarators, like tokamaks and other toroidal confinement devices, need to develop solutions for the plasma facing components, which withstand the harsh environment around thermonuclear plasma. The need for a concerted effort in the area of plasma materials interaction (PMI) is well recognized and documented as one of the main goals of FES as part of the "Burning Plasma Science: Long Pulse" category in the "Fusion Energy Sciences: A Ten-Year Perspective" document as well as in the recent dedicated report from the "Fusion Energy Sciences Workshop on Plasma Wall Interaction" [49]. While the generic characteristics of the PMI such as erosion yields, surface layer stability, hydrogen retention and diffusion into material for instance are independent of the magnetic configuration, the stellarator as such introduces a new geometrical environment which affects the integrated PMI solution. This integrated solution will include the 3-D geometry of the plasma edge in the PMI characteristics. However, most importantly, stellarators enable access to a PMI regime with unprecedented density levels and hence unprecedented particle flux densities. For example, Wendelstein 7-X will be developed within the next five years to confine a high-performance plasma at nominal plasma β of 5% with plasma core temperature in the 4-5 keV range and plasma densities on the order of 10^{20} m^{-3} on a quasi-stationary time scale of 30 minutes. This regime is not only attractive in view of plasma confinement, but in particular with respect to the plasma edge transport and PMI challenges which will be significantly larger under these conditions when compared to pulsed tokamak devices. Many of the Priority Research Directions (PRDs), which were identified by the recent report from the PMI workshop mentioned before, become critical elements of the Wendelstein 7-X mission. The island divertor as innovative divertor concept (PRD B) will be tested. It has to maintain a dissipative divertor plasma state to enable the long time scale discharges foreseen (PRD-B). Understanding the interaction of this innovative boundary solution with the optimized plasma core (PRD-C) is a key goal of the emerging research program at Wendelstein 7-X and the U.S. has supplied important equipment to facilitate this mission (trim coils, crystal X-ray spectrometer, funding for 3-D equilibrium advancement, turbulence diagnostics, heavy ion beam probe conceptualization, divertor scraper element). The trim coils also will allow studying fine-tuning of the plasma edge transport to affect the overall device performance (PRD-E). This shows that Wendelstein 7-X offers a premier environment to address those PRDs dealing with the general field of core to edge and PMI coupling.

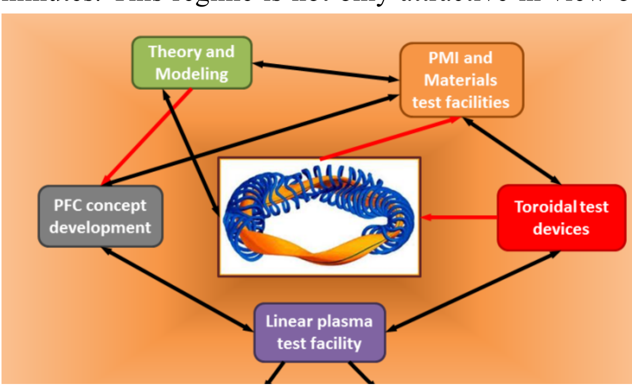


Figure 6 *Flow chart showing the development pathway from modeling to material testing facilities through to linear and toroidal test devices and onto larger experiments such as W7-X.*

At the same time Wendelstein 7-X represents also a state of the art environment for addressing the PRDs, which target the direct contact of the plasma with material surfaces and the science of materials in extreme environments. PRDs A and C include the quest for transformational boundary solutions and PRD D explicitly highlights the grand challenge of understanding the plasma material interface and its capability to withstand the conditions around a high performance fusion plasma. In this respect, Wendelstein 7-X can serve as a unique exposure facility featuring high particle flux (up to 10^{24} -10^{26} s^{-1} m^{-2}) and high power loads (up to 20 MW m^{-2}) on quasi-stationary time scales.

This assembly of relevant PMI conditions, which will be obtained at W7-X inherently distinguishes PMI research on stellarators altogether. A new facility in the U.S. could also deliver the same infrastructure and hence would deliver a large spin off for PMI research altogether. In tokamaks, a set of satellite devices is being developed to address the long pulse, high flux PMI questions which has to be addressed ultimately. A similar approach could be taken for stellarator PMI research as shown in Figure 6. However, W7-X itself delivers the relevant high flux, long pulse PMI conditions, so the need for long pulse plasma exposures is not a priority for a PMI program in stellarators. Rather, dedicated test facilities are needed which serve as concept exploration tools to make informed decision for long pulse, high flux PMI experiments at W7-X or on a new U.S. stellarator. Particular emphasis would be placed on the item “PFC concept development”, “Theory and modeling” and engineering for next generation PMI solutions.

Given the high particle fluence and the high heat fluxes which can be generated on stellarators with quasi-stationary time scales, it is promising to consider testing a reactor relevant first wall and divertor setup in a stellarator device in the U.S.. This would encompass with highest priority the use of metal material for the divertor target – an issue not addressed anywhere in the world program in combination with an optimized stellarator core. As the ion-root confinement anticipated in a neoclassical optimized stellarator is prone to impurity accumulation, such an integrated test of the compatibility with a high-Z wall interface is critical to demonstrate the viability of this stellarator concept with such wall choices. As second option in view of reactor relevant walls, assessing of operation in a non-absorbing wall regime is attractive. So far, all stellarator devices have been operated with absorbing walls and merely in saturated regime with a high overall fraction of desorbing wall inventory. Hence, the wall and divertor surfaces are often the most effective sink in the system, which aids density control. Heating the wall to about 150C would make it desorbing and remove this strong sink term from the particle balance. Assessing the level of discharge stability on the level of several particle confinement times (time scale of several second for the mid-scale device) in accompany with usage of metal wall and divertor material will represent an innovative integration test of aspects of reactor relevant plasma material interface with an optimized stellarator. This mission goal would be unique in the world-wide stellarator program and accompany the performance extension research on devices like W7-X and LHD as well as being an important preparatory step for a possible large scale, world class stellarator facility in the U.S.

Recently the WEGA stellarator was donated to the University of Illinois by the Max-Planck Institute for Plasma Physics in Greifswald, Germany with the idea in mind to also take advantage of this inherent steady state advantage. WEGA has been repurposed as the Hybrid Illinois Device for Research Applications (HIDRA). The research program offers attractive opportunities for development and testing of innovative technologies to improve the plasma material interface. For instance, development of the technology to deploy liquid lithium as a plasma facing material in steady state is tackled on HYDRA and would impact onto the two new facilities proposed in this report.

2.4.2. Research needs and opportunities

The overall description delivered before introduced several issues, which are discussed here in more detail systematically. The integrated challenge of the Plasma Material Interface encompasses both, the plasma physics in close contact with the material surface, which also includes considerations on the 3-D geometry of the stellarator edge as well as the material science directly. Highlights of leading research needs in form of targeted questions are:

1) Long term PMI: It has been observed in LHD that re-deposited carbon flake production impacts the PFCs and also can cause a collapse in the plasma. Can this be overcome or is this an issue faced by stellarators and tokamaks alike when operating in long pulse regime, which exceed the wall equilibration time scale for developing of surface coatings as well as for equilibration of wall temperatures. This topic is inherently linked to the question of far SOL material migration and deposition as this defines layer thickness and stability. In the long pulse regime, walls will get heated up and equilibrate at surface temperatures which might be above the delamination energy for the deposited layers yielding loosening of the material and sudden, large impurity injections. Such a process has to be mitigated and exploring this on reactor relevant, i.e. low H retention, material choices is an unaddressed issue in present day stellarator PMI physics.
2) How does the divertor geometry and material choice impact recycling, impurity production and the back-flow into the main plasma? What are the consequences of a given divertor geometry and material choice on the impurity exhaust and is the impurity fueling of the main plasma compatible with a possible ion-root neoclassical impurity pinch (inward impurity transport)?
3) What is the actual leading term scrape-off layer physics and in how far does it deviate from the tokamak paradigms? For instance, can a two point situation be identified to link upstream to downstream conditions and extrapolate the PMI physics for a given plasma performance goal. Which role does the 3-D edge magnetic structure play to identify such regions and do stochastic domains impact strong enough on the edge transport characteristics to require introduction of entirely new elements into such scaling models?
4) How do different heating scenarios influence the PMI? – ECH, ICH, LHW and NBI can all increase the fast ion population, which directly impacts the PFC material. At the same time localized heating scheme affect locally the collisionality which can alter the neoclassical transport features in this domain, An integrated test between the heating demands needed to obtain a given plasma confinement regime and plasma performance and the interaction with the materials interface needs to be conducted.
5) Material performance under extreme loads can be investigated in stellarators in realistic plasma boundary situations. In particular a high ratio between upstream and downstream plasma pressure can be obtained establishing realistic parallel gradients for impurity pinching by thermo-forces. Also, hot ions at the upstream position will allow establishing a representative magnetic pre-sheath, which define acceleration of the ions towards the target and hence the erosion processes and impurity source level from physical sputtering.

2.5. Energetic particle confinement and transport in stellarators

2.5.1. Introduction

Energetic particle (EP) confinement is an important issue for stellarators and remains a significant driver for 3D optimization strategies. Energetic particle populations in stellarators can arise from neutral beam heating, ICRF tails, runaway electrons, D-D produced tritons, and eventually alpha particles in D-T reactor plasmas. Due to their low collisionalities, EP components are more sensitive to deviations from quasi-symmetry than the thermal plasma species. Also, as a result of their non-thermal distributions, high energy densities, and high velocities (significant fraction of the Alfvén velocity), EP populations can drive instabilities through various resonant wave-particle interactions. In order to predict stellarator performance and choose optimization strategies, it will be important to understand EP transport both through direct classical orbit loss and from EP-driven instabilities. These issues will become especially important for stellarator reactor systems, due to the higher energy density of EP components. This leads to a greater potential for damage to plasma-facing components and negative impacts on the fusion ignition margin. Since existing stellarators have not achieved the levels of EP confinement optimization required in a reactor, addressing this issue will be critical to further development of the stellarator concept.

2.5.2. Classical Orbit Confinement

Improvement of EP classical orbit confinement in 3D configurations has been addressed using a variety of optimization approaches [50, 51, 52] . These have generally been effective; for example, losses of fusion-born alpha particles in reactor-sized stellarators can be reduced from 10 to 40% levels in un-optimized systems to a few percent in well optimized systems. The first line of attack would be to get as close as possible to one of the forms of quasi-symmetry. If quasi-symmetry were actually achievable, then one of the forms that minimized the bounce length of trapped orbits (i.e., helically or poloidally closed |B| contours) would likely provide the best confinement since this would result in smaller trapped particle banana widths. However, since precise quasi-symmetry has not been achievable, this banana width minimization likely is not relevant, and other strategies must be considered.

The next approach is quasi-omnigeneity (QO), or minimization of drift orbit displacements away from flux surfaces. There are a variety of strategies for implementing this, depending on the complexity of the magnetic field structure. For devices with more than about 5 field periods and aspect ratios > 5 to 6, there is typically one dominant well in $|B|$ over each field period. In such cases it is straightforward to compute the second adiabatic invariant $J_{||} == \oint dl\, v_{||}$; this will be a function of flux surface, poloidal angle, and the pitch angle parameter ε/μ. An optimization target can be readily constructed by minimizing the poloidal angle dependence of $J_{||}$ for a selected number of values of the pitch angle parameter ε/μ, encompassing both trapped and passing orbits. Minimization of the deviation of constant $J_{||}$ contours from flux surfaces [52,53] is equivalent to minimizing orbit deviations from flux surfaces and leads to improved EP confinement. A simplified version of this is possible if one focuses only on deeply trapped orbits, which often have the poorest confinement. In this case the contours of $J_{||}$ are equivalent to B_{min} contours, which are formed by recording the minimum value of |B| along toroidal angle for a set of fixed values of flux surface and poloidal angle locations. For example, in the case of heliotrons, such as LHD, shifting the plasma inward in major radius [54] improves centering of B_{min} contours on flux surfaces, leading to lowered trapped particle losses.

For lower field period/lower aspect ratio devices, while $J_{||}$ is still a valid invariant, its computation of becomes more complex, due to the presence of multiple wells in $|B|$ along field lines. In this case, a more direct minimization of guiding center drifts away from flux surfaces can be used. One method developed for this is the bounce/velocity averaged radial drift parameter of V. Nemov, et al. [55] For energetic ion confinement, this is analogous to the effective ripple parameter [56] ε_{eff} that has been used to characterize low collisionality neoclassical transport. Another approach is to simply follow a collection of guiding center orbits and measure their deviations from initial flux surfaces as a target for minimization; this has been used [57] in the NCSX and ARIES-CS designs. A final observation is that for devices near QO symmetry, diamagnetic currents from finite values of plasma β can improve EP confinement by minimizing the variation of B_{min} and, to some extent $|B|$, on flux surfaces. This effect has been predicted for W7-X [52,57,58], and in high β QO hybrid [59] configurations.

2.5.3. Instability driven particle transport

Fast ion transport driven by EP instabilities has been extensively studied in tokamaks and, in some regimes, can lead to 40-60% losses [60] of beam ions. This estimate is typically based on running with deuterium plasmas and beams, where the predicted level of DD neutrons (mostly from beam-plasma reactions) can be compared with measurements. The resulting neutron deficit then is a direct measure of fast ion transport levels. Such studies have not been carried out to the same degree in stellarators; however, now that LHD will be running in deuterium (starting March, 2017), and W7-X is certified for deuterium, this type of analysis will become possible on large stellarators. Any progress made regarding energetic particle confinement must thus involve international collaboration by U.S. scientists, as no domestic stellarator is capable of exploring such physics. Several models for the fast ion transport are emerging. One approach, the critical gradient model [61], assumes that fast ion profiles evolve to be close to marginal stability, and is based on the observed stiffness of the fast ion density pressure as heating deposition and power levels are changed. Other models attempt to calculate enhanced EP transport from the effects of Alfvén mode structures on orbit trajectories [62]. Understanding the causes and possibilities for reducing this transport requires a well-executed program of experiment/theory collaboration. EP instabilities have been readily observed and measured on all major stellarators (LHD, TJ-II, CHS, HSX, W7-AS) and generally fit with the theoretical frameworks that have been applied. However, much work remains to be done on developing and applying new models, especially in the nonlinear regime. Currently, mode structures and frequencies near marginal stability can be predicted using ideal MHD continuum and eigenmode codes (STELLGAP, CAS3D, AE3D), which take 3D effects taken into account. Linear instability can be addressed with both continuum (CAS3D-K), particle-based models (AE3D-K, EUTERPE), and fluid-particle hybrid models (MEGA, M3D-K). More recently, global gyrokinetic models (GTC, EUTERPE) have been applied. Tokamak experiments have developed several ways to suppress EP-driven instabilities and these methods should be applicable to stellarators. The use of ECH focused near regions in the q-profile where AE modes are present [63] is one method that has proven effective; it is thought that the effects of locally increasing electron temperature and pressure can close off accessible frequency ranges between low and higher frequency modes (GAMs and TAEs). Control of the q-profile and fast ion heating profile are other methods that can affect EP instabilities. There are also some options for suppressing EP-instabilities that are unique to stellarators. The simplest approach would be if the high density regimes seen on LHD and W7-AS can be sustained and utilized in a reactor. High density shortens the fast ion slowing-down time, lowering the fast ion beta, and the drive for EP instabilities. Future modeling could address this issue more quantitatively. A further way that stellarators could lower EP instability driven fast ion transport is to utilize 3D shaping to suppress AE modes. Since the Alfvén gap size is determined by mode coupling effects, this should be sensitive to shaping. This has not been attempted yet, but optimization target functions can be constructed that can guide a design in this direction. One approach would be to directly attempt to decrease the width of the Alfvén gaps by minimizing the variation of the coupling function ($g^{\rho\rho}/B^2$) within flux surfaces. Another is to use the fact that the density of Alfvén eigenvalues vs. frequency is lower for wider gaps. By targeting a more uniform eigenvalue density vs. frequency it should be possible to close off AE gaps, inducing stronger continuum damping. In experiments on NSTX with RMP coils, a suppression of several AE frequency lines was observed [64] associated with a partial closing off of the continuum gap [65,66] near the plasma edge. This may be related to increased continuum damping driven by the 3D field perturbations. While such tokamak experiments with 3D fields may help inform possible avenues of optimization, a significant gap exists in the U. S. fusion program with respect to stellarator EP physics. There are currently no operating U. S. stellarator experiments using injected or RF-driven fast ion populations for heating. While good connections have been maintained to international high performance stellarators (i.e., W7-X and LHD) this strongly limits the depth and flexibility with which U. S. researchers can study EP issues and optimization.

2.5.4. Research needs and opportunities

W7-X and LHD both have neutral beams and RF and, by virtue of their neutral beam arrangements, are well suited to study energetic particles.

- Can we acquire a predictive understanding of energetic particle losses, sufficient to establish requirements for power and particle handling at localized “hot spots”?
- Can we understand and control the fast ion redistribution due to Alfvenic and energetic particle driven mode activity?

2.6. MHD/High Beta Issues in Quasi-Symmetric Stellarators

MHD and high beta issues in many ways are rather different in stellarators relative to tokamaks. In tokamaks MHD instabilities provide rigorous bounds for plasma operation.

As external magnetic fields can provide the required rotational transform for confinement, plasma current inducing MHD instabilities can be avoided in stellarators. Indeed, conventional stellarators are not limited by a disruptive response to MHD instabilities (by disruptions, we mean the abrupt termination of the plasma discharge characterized by temperature collapse, current quench and runaway electron generation). The external rotational transform also provides an important centering force on the plasma whereby plasma induced displacements are countered by the interaction of plasma currents with the vacuum magnetic field. Moreover, stellarators are not subject to Greenwald level density limits.

Stellarators have been successfully operated at high beta and have tested predictions of pressure driven MHD instability boundaries. Generally, when linear ideal MHD stability limits for long wavelength pressure driven modes are breached, disruptions are not observed [67]. Low-n MHD instabilities do produce magnetic fluctuations. However abrupt termination of the discharge is largely averted with MHD activity simply providing weak confinement degradation.

2.6.1. MHD Equilibrium

Understanding how finite beta affects MHD equilibrium in 3D systems is a complex topic. A distinguishing feature of 3D equilibria is that they are generally topologically rich with a mixture of toroidal flux surfaces, magnetic islands and regions of magnetic stochasticity. The conventional model for estimating beta-limits in stellarators comes from MHD equilibrium considerations. At higher beta, large Shafranov shifts deform the flux surfaces. This deformation generates magnetic islands and stochasticity via Pfirsh-Schluter induced resonant magnetic fields. While much of this physics is clearly present in high beta stellarator operation, this is not the complete story as the conventional model can underestimate the equilibrium-beta limit in high shear LHD plasmas [68]. Quantitative calculations can be made with 3D MHD equilibrium tools that allow for magnetic island formation such as HINT, PIES, SIESTA or SPEC. However, detailed comparisons between these tools and experiments require careful treatment because observed edge pressure gradients can be seen in regions where the tools predict stochastic magnetic fields.

2.6.2. Stability and island physics

There are a number of effects outside of ideal MHD equilibrium theory that can affect this picture. Edge stochastic magnetic fields are complex and characterized by a mixture of short and long connection lengths. The allowance of finite parallel transport can give rise to profile gradients in these regions [69,70]. Additionally, finite beta can produce healing of magnetic surfaces via physics not accounted for in 3D MHD equilibrium tools. A particularly important example of this is the observed spontaneous healing of large vacuum magnetic islands on LHD [71]. In these experiments, large islands are observed to disappear as beta exceeds a threshold value that is empirically known to be a function of collisionality. A possible explanation of this physics is attributed to plasma flows that induce shielding currents that heal islands [72].

There are differences in the MHD properties of quasi-omnigenous (QO) and quasi-symmetric (QS) approaches to stellarator optimization. In QO, one can minimize the role of equilibrium bootstrap and Pfirsch-Schluter currents. This results a small Shafranov shift and a relatively robust equilibrium as beta varies. This strategy has been explicitly utilized in the W7X design. However, QS configurations will generally have Shafranov shifts, bootstrap currents and Pfirsch-Schluter currents with rising beta. All of these effects scale roughly with $q_{eff} = 1/(\iota - N)$ where $(N > 0)$ for quasi-helically symmetry and $N = 0$ for quasi-axisymmetry. Noting that typically the rotational transform scales roughly as ~ 0.2 N, we find that q_{eff} is small in QHS resulting in weaker bootstrap and Pfirsh-Schluter currents than in the tokamaks or QAS stellarator. However, it is may be possible to exploit pressure driven currents to improve flux surface quality. In particular, favorable neoclassical tearing mode (NTM) properties occur when the following inequality is satisfied [73]

$$\frac{1}{\iota - N}\frac{d\iota}{d\psi} > 0 \,.$$

While this criteria is violated in tokamaks and hence make them susceptible to NTMs, flexibility in stellarator design make it possible to provide a neoclassical healing effect on magnetic islands. From this criterion, we see QAS stellarators would benefit from stellarator-like averaged magnetic shear, whereas QHS stellarators benefit with tokamak-like averaged magnetic shear. Additionally, as QS allows for the possibility of plasma flows in the symmetry direction, it may be possible for QS devices to exploit flow healing of magnetic islands and/or regions of stochasticity.

In QS stellarators, substantial bootstrap currents may be present at high beta. Generally in QS, both external fields and plasma currents produce the needed rotational transform. Bootstrap currents can provide a free energy source for current driven ideal MHD instabilities and in principal MHD-induced plasma disruptions. A common metric employed to predict the disruptivity is the fraction of transform produced from external coils with typical values of order 10% distinguishing disruption from non-disrupting shots. The CTH program has dedicated a substantial portion of its research activities to addressing this question [74]. If QS stellarators are susceptible to current driven instabilities, methods for profile control may need to be developed.

Stellarators can also suffer from edge-localized MHD modes due to destabilization from edge pressure gradients and plasmas currents [75]. However, application of the conventional peeling criterion employed in tokamak plasmas indicates favorable peeling properties occur if $(J_{\|}/B)_{edge}$ $(d\iota/d\psi)^{-1}$ is negative. Note this is stabilizing for stellarator-like averaged shear in QAS and tokamak-like averaged shear in QHS. Hence, if the condition for favorable NTM is satisfied in QS, edge peeling mode properties will also benefit.

A traditional metric used in stellarator optimization is ideal MHD ballooning stability. While local MHD instabilities may not directly drive disruptions, there is a still a virtue in optimizing for ideal ballooning. There are common geometric quantities (magnetic field line curvature, local magnetic shear) that are present is describing instability drives of both ideal MHD ballooning modes and a variety of turbulent transport inducing microinstabilities, including toroidal ITG, kinetic ballooning modes, collisionless trapped electron modes, etc. In QS configurations, generally there are favorable ideal ballooning properties if the following inequality is satisfied [76].

$$\frac{1}{\iota - N}\frac{d\iota}{d\psi} > 0$$

In summary, crucial high-beta MHD issues in stellarators include: Can we predict 3D MHD equilibrium properties? Understanding 3D equilibrium is a central question to stellarator confinement schemes. Do MHD equilibrium properties determine beta limits? What extended MHD physics needs to be included to accurately predict magnetic field topology? How do the extended MHD corrections scale as we go toward reactor conditions? Can we reliably operate a high-beta, high-bootstrap fraction QS stellarator without deleterious instabilities? The demonstration of this property is a necessity for the viability of the QS approach. Can we simultaneously optimize for good MHD, transport and edge properties of a QS stellarator? Quantification of the key MHD issues will allow us to assess tradeoffs in design studies.

2.6.3. Research needs and opportunities

The present US stellarator program has a number of tools available to test the questions described above. There exist a variety of 3D MHD equilibrium codes and linear ideal MHD stability tools. 3D MHD equilibrium reconstruction is an element in the CTH, CNT, and HSX domestic programs and a central element of the US involvement on W7-X. The CTH program is dedicated to addressing the effect of 3D fields on disruptions. However, the domestic stellarator experiments have no high beta/high bootrap current capabilities.

Needed elements to the US stellarator program include: experimental tests of ideal MHD stability in high beta/high bootstrap fraction QS stellarator. Additionally, there are a number issues that could benefit from the application of extended MHD modeling tools. These include understanding 3D MHD equilibrium physics, quantifying island healing physics and understanding the consequences of breaching instability boundaries in optimized stellarators. Increasing the speed of 3D equilibrium codes that can handle islands and stochastic regions would allow them to be used more routinely for analyzing experiments and for optimization studies. Additional physics should also be added to these codes, such as viscous torque, and shielding due to flow.

2.7. Impurity transport and accumulation

2.7.1. Introduction

Impurity control is a serious concern in stellarators. Some of the reasons for this concern are the same as in tokamaks. In the core, impurities radiate energy and would dilute the fusion fuel in a reactor. A particularly important impurity in a reactor will be the helium ash, which must be extracted somehow. On the other hand, impurities confined to the plasma edge can be beneficial, as impurity radiation reduces the peak heat fluxes on the divertor.

Stellarators generally have a robust inward neoclassical impurity pinch, detailed in the next section. Impurity content appears related to the stellarator density limit. For other differences between tokamaks and stellarators, it is not immediately clear which device is superior in terms of impurity behavior. First, the scrape-off layers and divertors in the two devices look quite different. For example, the distance between plasma and divertor in the poloidal plane is typically larger in a tokamak, while the distance along a field line is larger in a stellarator. Second, the variation of the electrostatic potential on flux surfaces, Φ_1, is driven by different mechanisms in tokamaks and stellarators, and high-Z impurities will be sensitive to this potential due to their large charge. In tokamaks, Φ_1 is mostly driven by the centrifugal force, whereas in stellarators a large Φ_1 can arise due to large drift-orbit departures from flux surfaces.

The issue of impurities is closely related to the issues of divertors and plasma-materials interactions. An effective divertor which reduces peak heat fluxes to surfaces is likely to result in lower impurity influx from sputtering. However, divertor improvements alone cannot address the need to extract helium ash from the core of a reactor.

Reviews of impurities in stellarators can be found in references [77] and [78].

2.7.2. Neoclassical pinch and symmetry

We next discuss the neoclassical impurity pinch mechanism in stellarators and point out differences in the case of quasisymmetry. Due to the linearity of the drift-kinetic equation, the neoclassical radial flux Γ_a of a species a in a general toroidal plasma is a linear function of the driving gradients [79]:

$$\Gamma_a = \sum_b \left[-\frac{c_1^{ab} n_b'}{n_b} + \frac{c_1^{ab} q_b E_r}{T_b} - \frac{c_2^{ab} T_b'}{T_b} \right] \quad (1)$$

where primes denote d/dr for some flux label r, n_a, T_a, and q_a denote the density, temperature, and charge, $E_r = -d\Phi/dr$ is the radial electric field, and the coefficients c_1^{ab} and c_2^{ab} depend on geometry and collisionality. Note that the flux of one species depends on gradients of all other species b due to inter-species collisions in the kinetic equation. However in a general stellarator, the cross-species terms are small compared to the $b = a$ terms. When a is an impurity species, the large q_a tends to make the E_r

term dominate, so $\Gamma_a \approx c_1^{aa} q_a E_r / T_a$. The coefficient c_1^{aa} is always >0 , hence the impurity flux tends to be in the same direction as E_r.

Unfortunately stellarators are typically in an ion-root regime in which $E_r < 0$, implying impurity accumulation. In the ion-root regime, E_r is determined by setting the main ion ($a = i$) terms in (1) to ≈ 0 , giving $E_r \approx q_i^{-1}\left[T_i n_i' / n_i + \left(c_2^{ii} / c_1^{ii} \right) T_i' \right]$. Since $c_2^{ii} / c_1^{ii} > 0$ for relevant collisionalities, peaking of either the main-ion density or temperature profiles makes E_r more negative, strengthening the impurity pinch. One solution may be to operate in an electron-root regime where $E_r > 0$, but access to this regime typically requires strong electron heating and/or very low collisionality, so it is not clear this approach is relevant to a reactor where the ions must be hot and high density is desired.

However, the situation is fundamentally different in the case of perfect quasisymmetry or axisymmetry. In this case, the cross-species terms in (1) cannot be ignored, and in fact $\sum_b c_1^{ab} q_a / T_a = 0$, so the E_r terms in (1) sum to 0. In true axisymmetry this cancellation can be understood from the existence of a rotating reference frame in which the inertial forces are negligible and the electric field has been transformed away, so fluxes cannot depend on E_r. Thus, in symmetric geometry the main impurity pinch mechanism is absent. Furthermore, c_2^{zi} can be positive so a peaked main-ion temperature profile drives impurities outward ("temperature screening"), a phenomenon that has been observed experimentally in DIII-D [80]. While these favorable properties apply to *perfect* quasisymmetry, it is unclear if they can be achieved in experimentally relevant plasmas that are *imperfectly* quasisymmetric, such as NCSX, ARIES-CS, ESTELLE, and HSX.

2.7.3. Impurities accumulating regimes

Core impurities can limit the density compatible with long pulse operation in stellarators. An

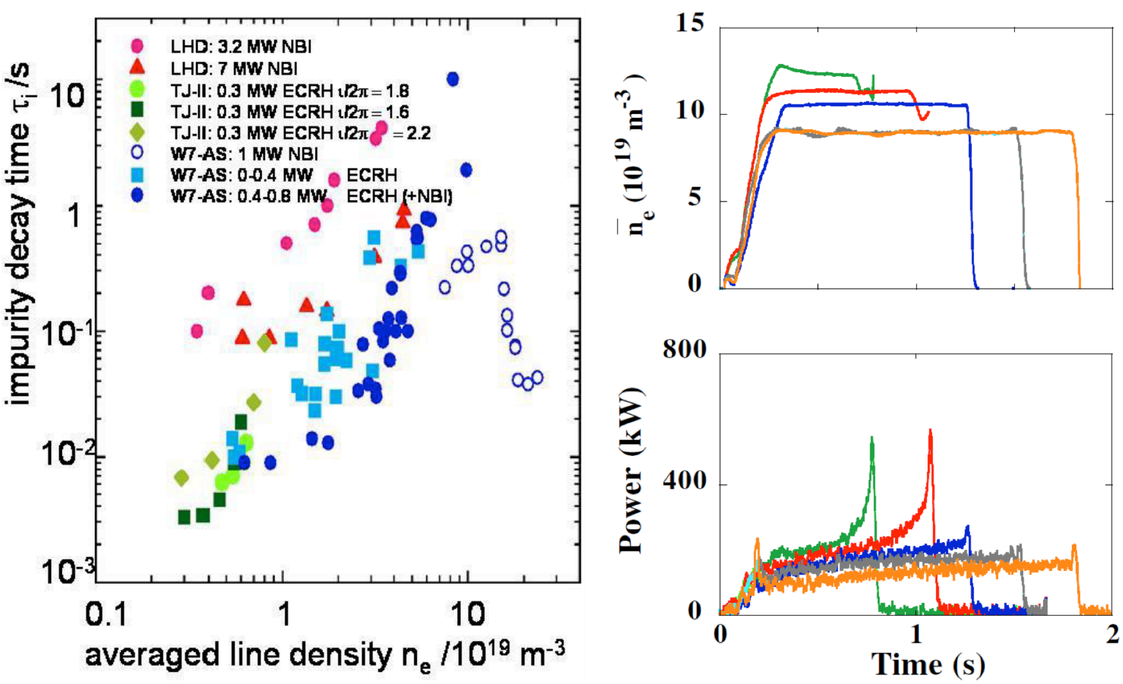


Figure 7 *(a) Stellarator impurity confinement is observed to increase with electron density, aside from special regimes like the HDH mode (hollow circles at right). (b)-(c) As shown by this set of W7-AS plasmas, above a threshold density, discharges typically terminate with a radiative collapse. Figures from [77] and [82].*

impoprtant experimental finding, seen in multiple experiments, is that the core impurity confinement time scales strongly with n_e. This trend, as seen in laser blow-off experiments, is shown in Figure 7.a. Energy confinement also increases with density, as in the ISS04 scaling $\tau_E \sim n_e^{0.54}$ [81], suggesting a common mechanism for energy and impurity confinement. Above a threshold density, impurity radiation typically increases with time until the discharge terminates in a radiative collapse, Figure 7.b-c [82].

In contrast to the core, high density in the scrape off layer (SOL) is believed to reduce the accumulation of impurities that originate in the edge [83, 84]. The reason is that the impurities experience friction with the main ions, which stream along **B** in the SOL towards the divertor. Increasing the SOL density increases this frictional coupling. Note this edge mechanism will be ineffective for purging helium ash born in the core if the core pinch keeps the ash from transporting out to the SOL.

2.7.4. Impurity expelling regimes

Besides the electron root regimes discussed above, two other low-impurity regimes have been observed experimentally, neither of which is well understood theoretically. One regime is the high-density H-mode (HDH) observed in W7-AS [85]. This regime is accessed with neutral beam heating and rapid initial gas puffing such that n_e exceeds a threshold around 1-2x10^{20}m^{-3}. Radial profiles of impurity radiation are hollow, steady in time, and generally much lower than for non-HDH discharges, as shown in figure 3 of [85]. In HDH mode, impurity confinement times measured by laser blow-off are strongly reduced compared to the expected scaling with n_e in "normal" discharges, as seen in Figure 7.a above.

The other noteworthy regime is the "impurity hole" in LHD [86, 87, 84], which in contrast to HDH mode is found at low rather than high electron density. The regime is associated with neutral beam heating and a peaked T_i profile. Core impurity radiation is seen to decrease in time, so there is evidently outward impurity convection in the core, despite measurements of a negative core E_r, and predictions of inward turbulent impurity transport [88]. Multiple impurity species are seen to have hollow profiles, with the hollowness increasing with Z.

Both the HDH and impurity hole regimes have only been observed on a single experiment. In both regimes the electron density profile is flat, consistent with the aforementioned property of the neoclassical pinch that peaking of the density profile drives impurities inward.

2.7.5. Research needs and opportunities

1) The transition from a stellarator-like impurity pinch to tokamak-like impurity screening near axi/quasisymmetry should be studied. In designs like NCSX/ARIES-CS/HSX/ESTELLE that are nearly but not perfectly quasisymmetric, are the departures from symmetry small enough to reduce or eliminate the impurity pinch? Impurity neoclassical transport in asymmetric and symmetric geometry can be computed using the codes PENTA [89] and SFINCS [90,91], which include the crucial physics of momentum-conserving cross-species collisions.
2) Related to item 1, one could directly target the neoclassical temperature screening coefficient in STELLOPT.
3) Gyrokinetic simulations can be carried out with impurities to model experiments and understand turbulent impurity transport in general. The first gyrokinetic study of impurity transport in a stellarator was recently carried out by [88].
4) There is a general theoretical need for ideas to decouple energy and particle transport. The energy confinement time must be long, but ideally the particle confinement time is short so impurities (particularly He ash) can be purged. Perhaps this could be accomplished by using shaping to control the phase between the fluctuating density and potential in microinstabilities, building on gyrokinetic investigations as described in item 3.
5) On both LHD and W7-X, the US has invested in XICS spectrometer diagnostics [92] which measure impurity density, flow, and temperature. These hardware investments should be supported with modeling.

6) It is now becoming possible to compute the effect of Φ_1 (variation of the electrostatic potential on flux surfaces) on impurity transport [78, 93]. Can Φ_1 be manipulated - either by manipulating the plasma shape or through ICRF power [94] - to produce an outward impurity flux?
7) There is a need is for increased collaboration between the various laboratories to compare the scaling of impurity confinement, particularly the scaling with respect to heating power P, A scaling $\tau \sim P^{-0.8}$ has been reported for W7-AS [77] whereas a very different scaling $\tau \sim P^{-3}$ has been reported for TJ-II [95].
8) Experiments should be conducted on W7-X looking for impurity-expelling regimes, and for similarities with the LHD and W7AS regimes. Careful comparison of measurements with simulations should be used to verify our understanding and ability to predict how to achieve impurity control by design.
9) Experiments should document the transport of both light and heavy impurities, both to clarify the physical mechanisms and guide choices of PFC materials.

2.8. Power Plant Issues

Since the early 1980s, seven conceptual stellarator power plants have been designed in the US (MSR [96], UWTOR-M [97], ASRA-6C [98], SPPS [99], ARIES-CS [100]), Germany (ASRA-6C [98], HSR [101]), and Japan (FFHR series [102]) – all with modular coils, except the FFHR (refer to Figure 8). The first attempt to simplify the plasma and in-vessel components delivered the 1982 UWTOR-M design with large 24-m major radius and high aspect ratio (A= 14). To reduce the machine size and cost, subsequent designs focused on modified coil configurations with non-uniform plasma shapes driven by non-planar coils. The 2006 ARIES-CS, with the lowest A of 4.5, is the only study that evaluated the most compact stellarator where the design point was pushed to the limit to examine the constraints imposed by

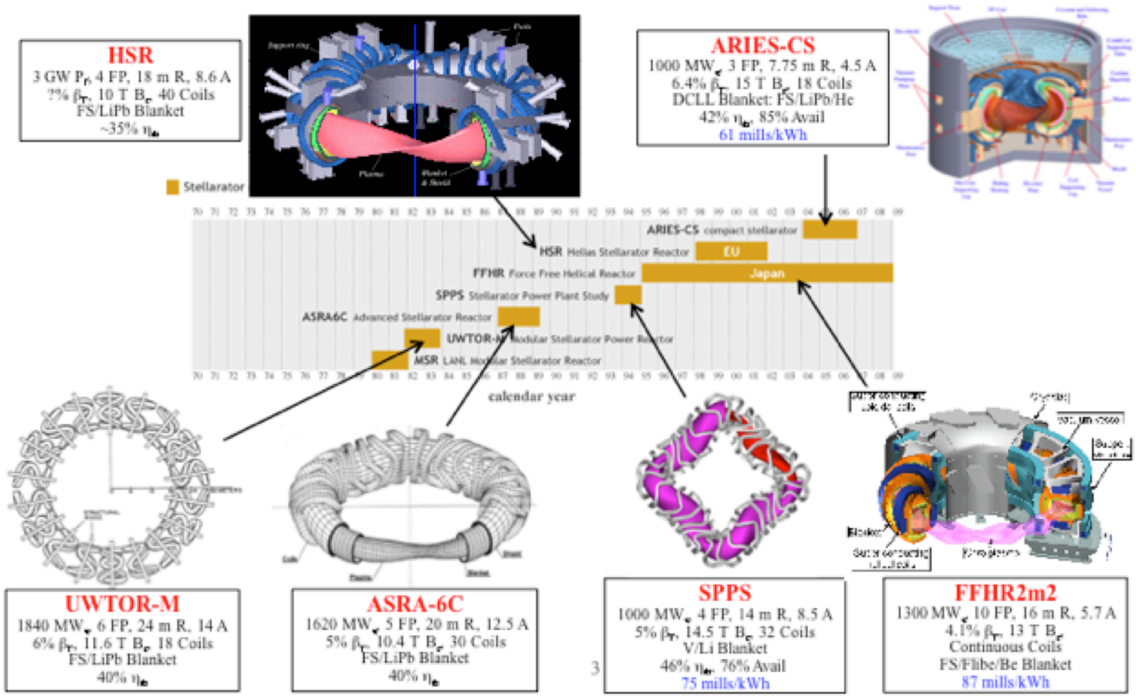


Figure 8 *Timeline of stellarator power plant conceptual designs developed since the early 1980s.*

compactness and potential tradeoffs. One of the main goals is to make compact stellarators comparable in size to advanced tokamaks to reduce capital costs, but this compactness introduces complexity to all power core components, causing difficulties in fabrication, assembly, and maintainenance. Several engineering issues unique to stellarators are outlined below focusing on ARIES-CS. Future stellarator power plants should optimize the configuration from the engineering viewpoint, not only for physics.

2.8.1. Complexity of coil configuration and in vessel components

Modular, highly shaped, non-planar coils for stellarators can have quite complex geometries and result in a non-uniform coil bore and more complex in-vessel components. The structural support for the coil is challenging as the electro-magnetic forces act both outward and to the side of each coil. An innovative solution proposed in ARIES-CS is to place the winding packs within grooves on the internal surface of large supporting tube that accommodates all coils. However, this solution is only compatible with a port-based maintenance scheme. Additive manufacturing may be the only economical fabrication method.

Changes to the stellarator engineering are recommended to simplify the coil design (to reduce bending and coil interlocking) and deliver a simpler, more practical configuration that satisfies the engineering and technology constraints. In more recent years since the ARIES-CS study was completed in 2008, improvements in coil design codes have made it possible to give greater weight to engineering metrics in the optimization. New machine configurations have been developed that are compatible with sector maintenance of in-vessel systems and more conventional support structures (see Section 2.9). Increasing the aspect ratio from 4.5 to 6.0 also helps make the coil geometry simpler, but at the expense of larger machine size. Further improvements in optimization methods are highly recommended in order to find configurations that simultaneously provide required plasma properties, are practical from a fabrication and maintenance standpoint, and are as compact as practically possible.

The first wall (FW) and surrounding in-vessel components conform to the non-uniform plasma. Within each field period, the configuration changes from a bean-shape to a D-shape, and then back to a bean-shape, continually switching the surfaces from convex to concave over a toroidal length of ~20 m. This means the FW and in-vessel component shapes vary toroidally and poloidally, representing challenges to 3-D modeling, fabrication, design integration, and maintenance.

2.8.2. Blanket concept and tritium breeding

Several liquid breeders and molten salts [Li, LiPb, Flibe ($LiF-BeF_2$), Flinabe ($LiF-NaF-BeF_2$), and LiSn] and ceramic breeders [Li_2O, Li_4SiO_4, Li_2ZrO_3, Li_2TiO_3, and $LiAlO_2$] were developed for fusion applications [103]. All six stellarator designs noted employed liquid metal and molten salt blankets, which are more suitable to flow liquids in complex geometries than ceramic breeder blankets with numerous alternating layers of beryllium, structure and ceramic breeder. In the US, the most prominent liquid metal concept is the dual-cooled PbLi (DCLL) blanket. Limited R&D activities for the DCLL blanket concept exist in the US, Europe, Japan, and China.

Any large fusion device must generate its own tritium (T) fuel (e.g., 136 kg per full power year in ARIES-CS) as T does not exist in nature in any appreciable quantity. A limited amount of T can be created outside the fusion plant, but this is an extremely costly process. The tritium breeding ratio (TBR) is the metric for T self-sufficiency. Accurately predicting the TBR for the complex stellarator geometry is a daunting task involving high fidelity, 3-D nuclear analysis modeling of the non-uniform blanket. A new computational tool was developed specifically for ARIES-CS by the University of Wisconsin-Madison to couple the CAD geometry directly with the 3-D neutronics code, using the DAGMC code [104]. Such a coupling is essential to preserve all geometrically complex design elements and speed up feedback and iterations. The main deliverables are the TBR and the toroidal/poloidal distribution of the radiation flux profile and neutron wall loading needed for the shielding analysis. In ARIES-CS, the TBR requirement and DCLL breeding capacity mandated that the minimum major radius should exceed 7.5 m [105].

2.8.3. Magnet shielding

Stellarators with non-planar coils have unique features that requires special shielding attention. In each field period, there are a few critical regions where the magnets are closer to the plasma, constraining the distance between the plasma edge and middle of the coil. This minimum distance (Δ_{min}) should accommodate the scrapeoff layer, FW, blanket, highly efficient shield, vacuum vessel, assembly gaps, coils case, and half of the winding pack. Being the most influential parameter for stellarator's size and cost, the optimization of this distance is crucial to the overall stellarator design. An innovative approach was developed in ARIES-CS to locally downsize the blanket in this region and utilize a highly efficient tungsten carbide-based shield. This approach placed a premium on the remaining blanket to supply the majority of the tritium needed for plasma operation, forcing the major radius to exceed 7.5 m in order to limit the coverage of the reduced-size blanket.

2.8.4. Plasma heating, divertor design and heat loads

Similar to advanced tokamaks, a large fraction of the plasma power should be radiated in order to limit the heat load on the stellarator divertor plates to less than the current 10 MW/m^2 engineering limit. Alpha losses contribute to the divertor localized heat load (e.g., 13 MW/m^2 peak without alpha losses and 18 MW/m^2 peak with 5% alpha losses in ARIES-CS). These localized heating levels would likely damage the solid materials through blistering and erosion. Further effort to reduce the alpha loss fraction below 5% along with further optimization of the divertor materials and configuration should be pursued. Besides careful tailoring of divertor plate shape and orientation to reduce the peaking factor, an optimized alignment of plates (as proposed in ARIES-CS) could lead to an acceptable solution.

Heating systems and technology with minimal port size should be emphasized to alleviate the streaming problem and enhance the TBR. Several penetrations for plasma heating (NBI, ICRF, LH, etc.) protrude through all in-vessel components, requiring careful integration with non-planar coils and power core components. Such penetrations compete for the available space for breeding and allow neutrons to stream through, putting the shield efficiency in jeopardy. NBI ports in particular have large footprint at the FW and degrade the TBR the most. Stellarators have an advantage in not requiring current drive, so they can operate close to ignition and their heating systems and launch structures can be simplified.

2.8.5. Fabrication, integration, and maintenance

Fabrication of stellarator components (FW/blanket, shield, vacuum vessel, coils and their supporting structures) with conventional means would be very challenging and costly as such components vary in shape and curvature throughout the field period. New fabrication technology (additive manufacturing, nano-structured metals, precision casting, advanced joining technologies, etc.) can create unique shapes directly from the CAD definition with reduced labor and final machining [103,106]. This could be a "game changer" for stellarators in particular.

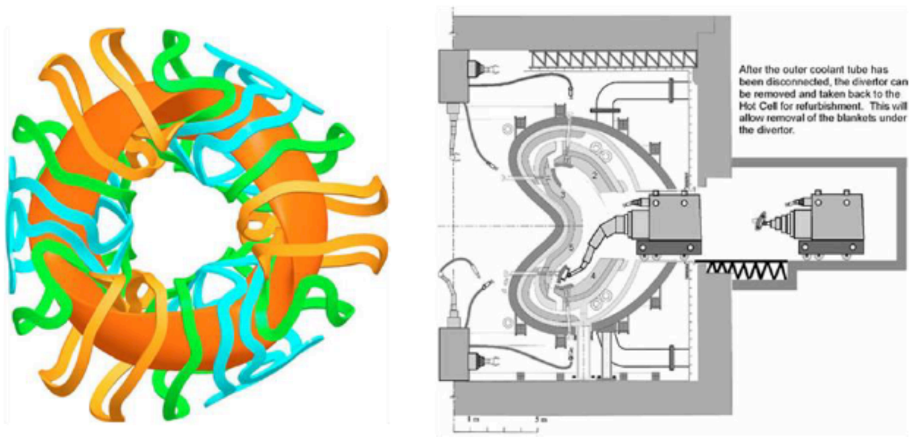
Maintaining and replacing stellarator components can be considerably more challenging compared to tokamaks if the access to internal elements is limited by the lateral space between modular coils [107]. This space determines the maximum dimensions of blanket and divertor modules. A port maintenance scheme is generally more complex and time-consuming (compared to a sector maintenance) resulting in a negative impact on availability. Therefore, accommodating a sector maintenance approach has been a high priority aim of coil design improvement studies in recent years, and the latest compact stellarator machine configurations are based on sector maintenance of a relatively small number of large modules (see Section 2.9).

2.9. Stellarator Coil Simplification

One of the issues that has been raised regarding the utility of stellarators as a steady state confinement device is that "stellarators are too complicated". In order to address this issue we first need to define what makes stellarator coils complex. The nonplanar aspect of the coil leads to large bending forces on the coil that necessitate a more complex support structure to accommodate the loads. Large non-planar excursions can lead to interlocking coils that hamper assembly. Two items that have been clearly identified as issues for future stellarators are 1) access for maintenance, and 2) the fact that stellarators often require a small distance between the plasma and coils.

2.9.1. Maintenance and access

The maintenance access issue is illustrated by the figure of ARIES-CS below, taken from T. Brown et al., submitted to the 26th IEEE Symposium on Fusion Engineering [108]. As can be seen in the figure, the modular coils cover much of the surface of the device. As a result, the ARIES team adopted a port based maintenance scheme. The availability of the device would be severely limited by this maintenance approach.



ARIES-CS 4.5 AR, 7.75-m R_{axis}

ARIES-CS port maintenance approach

Figure 9 *a) top-down view of the ARIES-CS stellarator showing the tight space between coils, and b) the port-based maintenance scheme envisioned for this device.*

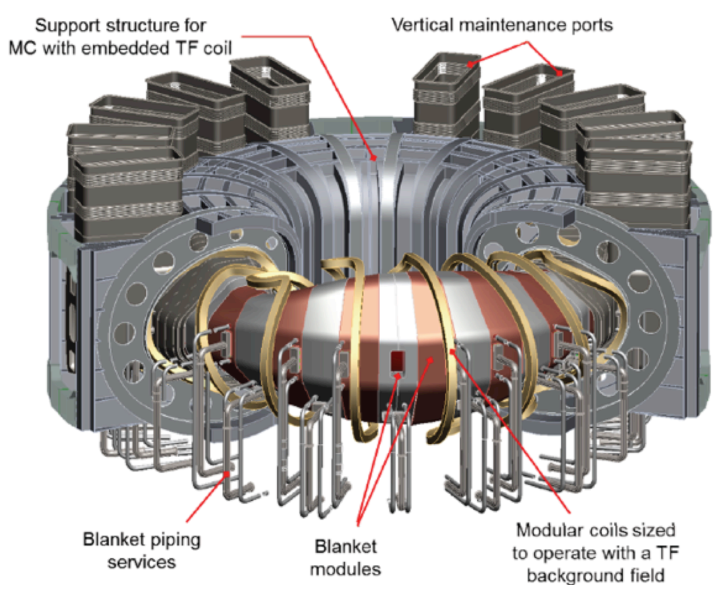


Figure 10 *Cut-away view of a maintainable stellarator with the outer half of the modular coils constrained to be vertical for sector maintenance access.*

In order to address the issue of maintenance access a new code, COILOPT++, has been developed with the following features 1) a spline based representation of the coils, 2) the ability to target coil penalties and freeze coil geometry for individual coils 3) freedom to straighten MC outer legs over asymmetric distances above and below the outboard mid-plane; 4) the inclusion of nested saddles with enforced minimum coil-to-coil separation distances was implemented; 5) self-symmetric saddles, necessary if saddles are to straddle toroidal symmetry planes, were included; 6) saddles can now be constrained to lie within a chosen u-v patch on the control winding surface and finally, 7) coil winding surface geometry can be accepted from the Pro-E CAD program. The code features listed above have enabled the achievement of self-consistency between large-sector-maintenance requirements and plasma equilibria with attractive physics.

An A = 6.0 quasi-axisymmetric stellarator plasma was considered, based on the work of Ku & Boozer [109]. In moving from ARIES-CS parameters (A=4.5, R = 7.75m, B = 5.7T) to an aspect ratio A=6.0 configuration while retaining the values for fusion power, beta, plasma volume, and toroidal magnetic field leads to a major radius of 9.39m. The plasma current, I_p, is scaled to keep I_p/RB = 0.045, leading to I_p = 2.6MA. Plasma beta is set to be 4.0%. Fourier coefficients describing the target plasma boundary of the A=6.0 configuration are taken from Table 1 of reference [110], and scaled appropriately.

The resultant stellarator coil design with a large sector maintenance scheme is shown in Figure 10. In the vacuum configuration using the coil currents from the high beta case there are no flux surfaces, although at 4% beta the magnetic surfaces are close to the targeted design. The ideas described in this section are the first attempt to include constraints on the physical location of the coils for an optimized stellarator. COILOPT++ supports the possibility of adding saddle coils. Given that a tool has been developed that can aid the design of a more maintainable stellarator, our next steps are to add saddle coils to the optimization so that the vacuum transform is close to the 0.3 value given in the original Ku & Boozer design. Other goals include minimizing the bootstrap current and optimizing the aspect ratio.

2.9.2. Coil efficiency

A second important aspect of stellarator coil complexity is the fact that stellarator coils often need to be relatively close to the plasma. The reason for this small plasma-coil separation in stellarators is that the shaping components of the magnetic field created by coils decay through space, so for a given stellarator plasma shape, the non-planar excursions of modular coils must grow exponentially as the plasma-coil separation is increased. Therefore, practical constraints on the coil shapes (such as minimum bend radius and absence of intersections) provide an upper limit to the plasma-coil separation. Plasma-coil separation can be increased if the plasma and coils are scaled up together by the same factor, but this approach leads to large expensive facilities. As discussed in [111], the small engineering margins associated with the small coil-plasma separation in W7-X were a significant driver of schedule delays and cost increases in that facility. The issue of small plasma-coil separation becomes even more important in a reactor, because a blanket and neutron shielding must fit between the plasma and coils. Indeed, in the ARIES-CS reactor study, plasma-coil separation was identified as "the most influential parameter for the stellarator's size and cost" [112]. However, the maximum feasible plasma-coil separation is a strong function of the plasma shape. For example, plasma shapes with concave regions tend to require very close coils, whereas plasma shapes with convex cross-sections permit the coils to be more distant. Any theoretical or numerical advances that lead to plasma shapes that permit larger plasma-coil separation will have a significant impact on the cost of future stellarator experiments and reactors.

In order to find plasma shapes compatible with more distant coils, or equivalently to find plasma shapes produced by simpler coil shapes at a fixed separation from the plasma, innovations are needed in the optimization framework. Stellarator optimization has typically been performed in two stages. In the first stage, the shape of the plasma boundary is varied to optimize various physics properties of the plasma, such as neoclassical transport and MHD stability. The primary code used in the US for this purpose is STELLOPT [113]. In the second stage, coil shapes are optimized to yield a plasma of approximately the shape that results from stage 1 by minimizing B_{normal}, the magnetic field component normal to the desired plasma boundary. This second stage can be done using the codes COILOPT or COILOPT++. This two-stage approach has several advantages: it is computationally robust and fast, and helps ensure good flux surface quality, since generic coils shapes will yield magnetic islands and volume-filling field regions. But as described above, the non-planar coil excursions and maximum feasible plasma-coil separation are strong functions of the plasma shape, so it is crucial to somehow consider coil complexity issues concurrently with the physics optimization, not only in a later stage.

There are two possible improved optimization frameworks, both of which are important research opportunities. The first option is a combined STELLOPT-COILOPT++ single-stage optimization, in which the independent variables are the coil shapes, and the physics figures-of-merit are targeted, with no B_{normal} target. A challenge for this approach is to ensure good flux surface quality, perhaps by devising new optimization targets that minimize islands and stochastic volumes. The second option is to retain the two-stage optimization framework (first STELLOPT, then COILOPT++), but to include some form of coil complexity target in the first stage. Preliminary investigations of this approach were made during the NCSX optimization, penalizing various quantities [114] computed with the NESCOIL code [115]. More recently, Landreman & Boozer [116] defined and explored several new magnetic field 'efficiency' metrics, called the efficiency sequence and feasibility sequence. Much more could be done to systematically compare these various coil complexity figures of merit, compare how effective they are as optimization targets, and explore how much the plasma-coil separation can thereby be increased. Such research could significantly reduce the size and cost of future stellarators.

2.9.3. Research needs and opportunities

The ideas described in this section are the first attempt to include constraints on the physical location of the coils for an optimized stellarator to enable better engineering and maintenance features of the final device. Given the ease with which an attractive solution was found, it seems clear that additional physical constraints could be added they are deemed advantageous. Additional constraints that could be included in the plasma chape optimization are plasma-coil separation and shaping efficiency [116]. Additionally, scans of plasma shaping and aspect ratio should be carried out. Another possibility is to include the forces on the coils as constraints in the design. Eventually the goal of this coil optimization using geometric constraints is to include it in a complete optimization activity, which includes all of the metrics described in the preceding sections of this document. In addition, the trade off between coil complexity coming from the need for external rotational transform and the disruption risk associated with transform generated by the bootstrap current needs to be understood explicitly.

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3. An invigorated US stellarator program

This section presents series of research options that could be used as elements of a strategically designed national stellarator research plan. How many of the options described could be addressed depends on the level of resources available, but there are many areas where the US has the ability to be world leaders in stellarator physics. A section is also included that describes areas that are either already being addressed overseas or would be best addressed on such facilities.

3.1. Needs and priorities in Analytic theory

3.1.1. Background

Analytic theory has played a foundational role in magnetic confinement physics. The stellarator has been uniquely impacted by the role of analytic theory. The central means by which the concept has been improved is through the development of optimization schemes defined by sets of metrics derived from analytic theory. For example, the theoretical notions of quasi-symmetry and quasi-omnigeneity were developed with the goal of improving neoclassical transport. As we seek to further improve the stellarator concept, there are a number of topical areas where new ideas for optimization require insight. In the following we outline a few possible areas where analytic theory can provide a sound basis for these advances.

Nearly all plasma physics properties in stellarators are influenced by the properties of the three-dimensional magnetic fields. 3-D magnetic field perturbations are also present in tokamaks but largely it is the resonant magnetic field components that garner the most interest as these fields can induce island formation and/or shielding responses that are require sophisticated modeling. While resonant fields are also of interest to stellarators, the bulk of the 3-D fields present are non-resonant and can be used to manipulate the rotational transform profiles to avoid resonances. Since non-resonant 3-D fields are far easier to model than resonant components, it may very well be that 3-D equilibrium of interest to stellarator applications are far more robust than 3-D tokamak equilibria.

Because stellarators do not require large plasma currents, they are less susceptible to the complex self-organization physics that permeates other magnetic confinement concepts. This makes it possible to make large steps in stellarator development for fusion applications relative to the tokamak. By exploiting this feature of the stellarator, the time and cost of developing fusion power could be greatly reduced. Of course, these large steps are only possible if credible theoretical understanding of crucial plasma physics issues can be obtained and verified.

3.1.2. Areas of interest

There is a growing interest in the stellarator community to understand how 3-D shaping can be used to optimize design with respect to microinstabilities and the associated anomalous transport. The same class of microinstabilities present in tokamaks (ITG, TEM, KBM, etc) are also predicted to be active in stellarators. Simplified analytic models need to be developed to understand the fundamental role 3D geometry has on influencing the linear stability properties of these modes. Zonal flows are known to have a crucial role in establishing turbulent transport levels. Their role in the nonlinear saturation of microinstabilities in 3D systems needs some attention. Understanding how 3D geometry impacts nonlinear saturation processes of microinstabilities is an area that has received almost no attention in theory community. Additionally, it would be advantageous to develop schemes for decoupling energy and particle (particularly impurity) confinement. The question is can we figure out means to encourage impurity pump out while maintaining good energy confinement.

Obtaining solutions to MHD equilibrium equations in 3-D has been a central question of stellarators since their invention. A generic feature of equilibria in 3-D is the presence of mixed magnetic topologies with magnetic islands and regions of magnetic stochasticity present. As is known from tokamak studies, magnetic island physics is quite intricate with a number of effects playing a role in determining magnetic island onset, growth and saturation. Analytic insight is needed to understanding the role of various non-ideal MHD effects on stellarator islands including the roles of plasma rotation, anisotropic heating conduction, polarization currents, neoclassical effects etc.

One of the advantages of low current stellarator over the tokamak is the lack of disruptions. Stellarators have exceeded stability limits for pressure driven MHD instabilities without deleterious effects. Understanding how ideal MHD instabilities evolve nonlinearly in stellarator configurations is an open question that needs attention. Current driven instabilities can produce disruptions in 3-D hybrid configurations. Understanding how and what level of 3-D shaping is needed to avoid disruptivity is an open question that would benefit from analytic insight. Transport barriers can lead to situations in tokamak plasmas where MHD instabilities can be excited. Understanding how this behavior translates to stellarators is another area where analytic insight can be applied. In particular, one question would be to ask if stellarators can operate with edge pedestals that avoid peeling/ballooning instabilities by using 3-D shaping/rotational transform control.

Controlling impurities is a crucial issue for stellarators. Conventional neoclassical transport theory predicts impurity accumulation due to the formation of a self-consistent radial electric field. However, there are examples in stellarator experiments where impurities are expelled from the core in sharp contrast to the neoclassical picture. At the moment, there are no compelling models to explain these observations. Current possibilities for future studies include understanding the role of potential variation within the flux surface and the possible role of turbulent transport. An open question is to understand what level of asymmetry is required for neoclassical theory to dominate. Can a quasi-symmetric stellarator be designed to be sufficiently symmetry to appeal to the tokamak solution for impurity transport? An important goal would be to generate analytic criteria for use in optimization with impurity control in mind.

One of the most elusive areas of study in the stellarator configuration is the edge/divertor region. Part of the reason for this is the complicated properties of the magnetic field where there may be no nested magnetic surfaces and field lines generally have a range of connection lengths. There is a need to develop analytic techniques to describe the properties of edge magnetic field. It would be useful to develop some analytic metrics of how the magnetic field properties affect edge physics properties. Additionally, cross-field flows and transport processes may play a more prominent role in stellarator edges relative to tokamaks. There is very little analytic understanding of the role of these effects on the edge phyiscs. Finally, there is a continuing need to develop better divertor configurations for stellarators. Analytic insight is required to guide improved designs. One potentially intriguing idea is the notion of a non-resonant divertor. This idea relies on the presence of sharp edges in the magnetic surface shape of the outer part of the plasma. These edges tend to dictate the helical strike point pattern in a manner that is insensitive to the plasma profiles and rotational transform value. Nearly all optimized stellarator designs have the sharp edged features in the magnetic surfaces and hence may have a natural non-resonant divertor solution. This divertor concept contrasts the island divertor concept of W7-X that requires detailed control of the edge rotational transform value. Traditional numerical tools used to describe MHD equilibrium properties use a Fourier representation. Since this is not convenient for describing sharp edges, a different theoretical representation would need to be developed.

Energetic particle confinement continues to be an issue in optimized stellarators. While perfect quasi-symmetry guarantees good confinement of fast particles, inevitably there are deviations from QS in actual configurations. However, not all deviations from symmetry cause energetic particle loss. There is a need to understand how one can modify the magnetic field structure to minimize energetic particle loss.

A broad area of interest would be to develop theoretical models for understanding tolerances. The tight tolerance requirements impact fabrication and coil assembly costs. More broadly, we need to understand how close is close enough when optimizing for a particular physics goal. Stellarator designs are affected by trade-offs from different desirable design features. While we have some idea of how to characterize deviations from quasi-symmetry in deducing thermal particle transport in the low collisionality regime, we do not have an appreciation for how close to quasisymmetry (i.e., how small off-diagonal elements of the magnetic field spectrum) is required to take advantage of the beneficial effects of flow and flow shear for suppression of turbulent transport, island healing and impurity transport.

In optimization approaches, it would be beneficial to formulate an approach that simultaneously addresses plasma physics and coil needs. There is a need to develop approaches that only targets the "important" modes of the normal component of the boundary magnetic field. Additionally, there is a need to develop coil sets that are flexible rather than optimized for a single point design and incorporate the needs of fabrication and assembly tolerances. Finally, stellarator design would benefit by incorporating modern optimization methodologies and algorithms from applied mathematics and other engineering fields.

3.1.3. Summary of program elements for analytic theory

- Work describing the role of 3D geometry in determining zonal flows
- Insight into the role non-ideal MHD effects
- How much 3D shaping is required to prevent disruptions?
- Can quasisymmetry alone control impurity influx?
- What is the relationship between the deviation from quasisymmetry and flow damping?
- Are there simple metrics for divertors that can be included in stellarator optimization?
- How can we integrate the new optimization metrics successfully?

3.2. Needs and priorities in code development and computation

3.2.1. Background

Stellarator research has, essentially from its inception, taken full advantage of theory and computation to tackle a wide range of physics issues. Driven by challenges posed by, for example, high neoclassical transport and low macroscopic stability limits, stellarator physicists have been pro-active in using predictive simulations to design concepts that overcome these challenges. Several experiments have then been designed and constructed to validate that the simulation-based optimization did indeed result in plasmas with the targeted properties. Overall this has resulted in a very healthy relationship between theory and experiment, geared towards solving problems and avoiding them altogether rather than simply studying them after the experiments have occurred.

The magnetic configurations forming the basis of stellarator designs are typically arrived at using an optimization suite, with STELLOPT being the standard in the US, comprising of a set of codes that calculate MHD equilibrium, the transport and stability properties of that equilibrium (among other properties), and then navigate the parameter space defining the magnetic configuration to optimize the calculated properties towards the desired targets. This sort of optimization essentially embodies our understanding and predictive capability in stellarator physics, and further simulation efforts are generally geared towards developing tools that, either directly or by informing reduced models, can be added to the optimization suite to target new physics properties and further improve on possible stellarator designs.

Each of the physics areas discussed in this report has a significant associated simulation effort. The list of simulation tools described in the previous sections will not be repeated in detail here. Rather, we focus on the set of high-priority code development and application areas that are viewed as most at-risk of being understaffed. While further research and development is clearly needed in, for example neoclassical impurity transport physics, this are currently the subject of vigorous exploration and is relatively likely to mature as simulation efforts without additional intervention. In contrast, the specific areas of energetic particle physics, gyrokinetics, divertor optimization, integrated scenario optimization, and extended MHD studies including calculations of 3D equilibria with islands and stochastic regions, are identified as high priority research areas which would significantly benefit from additional resources.

3.2.2. Areas of interest

Although energetic particle (EP) confinement is a key issue for the stellarator concept, and although computational tools in this area are well developed, the present level of effort in this area worldwide and nationwide is low. Alpha confinement was found to be one of the most serious concerns for a stellarator reactor [Najmabadi (2008)], and energetic particle confinement in the nominally optimized W7-X configuration is expected to insufficient to demonstrate the confinement required for a reactor [Drevlak (2014)]. At present there is no organized activity in the U.S. to improve stellarator EP confinement (although some individual activities do exist, for example as part of the GSEP SciDAC effort). As described in section 2.5, good codes exist to compute EP transport, and indeed the tracking of alpha guiding-center trajectories is a particularly 'clean' problem for which the equations are robust and modeling should be highly reliable. Also as described in section 2.5, ideas exist for ways to optimize configurations to have improved EP confinement. More resources towards a coordinated effort are needed in this critical area.

Another need of the stellarator program is a 3D equilibrium code that handles islands and stochastic regions and is sufficiently fast to be used routinely in the analysis of experimental data, in optimization studies, and in the planning of experimental campaigns. The code needs to be able to correctly handle unflattened and partially flattened pressure profiles across magnetic islands, as well as flattened profiles, and it must be able to handle the resulting modification of the Pfirsch-Schlüter and bootstrap current in and near the islands. Additional physics such as flow shielding of rational surfaces, and viscous torque on the magnetic islands, should also be incorporated in such a code. It would also be desirable to couple such a code to transport and stability codes. Stellarator equilibria have intrinsic islands. The vacuum field is generally tuned to minimize the size of the islands. As beta is increased, nonaxisymmetric pressure-driven currents produce resonant magnetic field components that increase the island widths. In a device such as W7X, where the rotational transform (q) profile has been tuned to avoid low order rational surfaces, the bootstrap current will modify the q profile, as well as generating resonant field components, and flux surface breakup to produce islands and stochastic regions may be an issue.

The field of gyrokinetic simulation has quite broad applicability in terms of the problems that can be addressed. The genesis of the field and still the main application has been in understanding and predicting core turbulence and transport. In addition, new efforts have extended gyrokinetic studies to include fast-ion driven instabilities and the resulting energetic particle transport, as well as increasing efforts using gyrokinetic simulations to predict the transport of impurities. Significant progress has been made in the past several years in extending these studies to include 3D systems, and exploring the ability to use 3D shaping to tailor the turbulent transport properties much in the way that neoclassical transport is controlled. An initial optimization has even been performed with STELLOPT using proxy functions for turbulence that were developed based on gyrokinetic calculations, with the properties of the resulting configuration confirmed by more detailed simulations at the optimized design point. At the present time, the various codes—notably the code GENE—have been developed to the point that they are fully applicable to stellarators (or will be imminently). The primary need is in staffing resources dedicated to applying these codes to stellarator problems, be they application to existing experiments for the purposes of validation, or in further optimization studies targeting gyrokinetic properties. Overall the potential of this field is extremely high, as new simulation tools offer the possibility of designing the full transport properties—both neoclassical and turbulent—into the configuration and producing systems with significantly better plasma confinement than is presently accessible in either tokamaks or stellarators.

A second area where further resources are needed is in stellarator divertor physics. In this case the code set requires further development, as the existing tools have significant shortcomings in their capabilities. The end goal of divertor physics studies is again to produce an optimization that maximizes desirable properties. The geometric optimization for divertors has made significant strides recently, with methodologies developed to tailor component shapes for a given magnetic field line geometry in order to produce designs that minimize surface heat loading (this is embodied in the Scraper Element as designed for W7-X). The next step along this research path is to incorporate this optimization within codes like STELLOPT in order to vary the magnetic geometry in addition to component shaping. At present this area is understaffed, to the point that significant progress is unlikely to be made without additional dedicated resources. The major following piece of divertor optimization lies in controlling the plasma transport characteristics to produce a desired divertor plasma state, much as is done in tokamaks that target a partially detached, low temperature, highly-dissipative divertor. The EMC3-EIRENE code is the state of the art 3D edge plasma-neutral transport code aiming at providing the predictive capability required to perform such an optimization. While this code is very capable and has proven extremely useful, it shows several shortcomings in its physics basis (for example, the lack of drifts and flux-limiter corrections for kinetic effects) that prevent it from being more fully predictive. While development of EMC3-EIRENE is ongoing, further resources in this area would help to address these shortcomings and produce a more capable code more quickly. Further, this is at present the only code available that solves the full set of transport equations in 3D, and so is lacking the healthy cross-fertilization of ideas and benchmarking that comes from having multiple codes solving similar problems, the benefits of which have been shown for example by the variety of core gyrokinetic codes.

Similar to gyrokinetics, the field of extended MHD simulation has made significant strides mainly within the tokamak community, with multiple codes that will become capable in the near term of being applied to stellarators (namely, NIMROD and M3D-C1). Recent development and application within tokamak research is highly applicable to stellarators, with the penetration of resonant perturbations and formation of islands being the subject of intense research due to the observed but so-far unexplained suppression of ELMs. High-priority research for stellarator includes exploring the formation of islands and understanding how they can be avoided as part of the optimization process. Further, the observed resilience to macroscopic stability limits as seen in experiments should be subjected to much more intensive study to test if physics behind this and the general robustness off stellarator against catastrophic instabilities (i.e., disruptions) extrapolates to reactor-scale devices. As in the case of gyrokinetic simulation, the primary need for extended MHD studies are the resources and staffing to apply these codes to stellarators. A useful code base already exists in part due to efforts geared towards tokamak research, and efforts are ongoing to leverage these capabilities by making these codes also applicable to stellarators.

A final priority area of simulation-based research lies in integrated scenario optimization. This in part includes extending the optimization tools such as STELLOPT to include further physics (such as the turbulence optimization described above). Improving the incorporation of engineering constraints into the configuration optimization process should be a high-priority, as producing systems that manage the unavoidable clash between physics desires and engineering realities from the beginning is clearly needed. In addition, further effort should be put into exploring novel, stellarator specific operational regimes in order to draw out their full potential. For example, the high-density core that is evidently made possible by removing the Greenwald density limit could also benefit energetic-particle confinement by reducing the fractional fast-ion pressure. As another example, if core turbulence and transport could be controlled such that an attractive, high-confinement regime could be accessed without the need for an edge pedestal, the divertor scenario could be much more tractable due to the lack of ELMs and the increased capacity for core radiation as power flowing through the edge to sustain the H-mode would presumably no longer be needed. Again in this case, the primary need is resources to explore these possibilities and test the potential to deliver significantly improved scenarios through optimization.

A major focus of simulation efforts in the near future should be on validation against experiments. This should take advantage of both the modestly-sized existing domestic devices, as well as the various international stellarators--especially the large W7-X experiment now beginning operations. Emphasis should be placed on validating the physics that could form the basis of next-generation optimizations; for example, the ability to predict core turbulence or divertor plasma conditions should be tested. These studies can take advantage of experiments that may not be optimized for these aspects at present, being aimed more at establishing the physics basis for future optimizations. To better lay out the possibilities in this area, a more in-depth study of physics areas that could benefit from validation studies taking advantage of the existing array of experiments should be performed, along with exploring how such cross-machine experiments and analyses could be coordinated.

3.2.3. Summary of program elements for code development and computation

- A concerted effort to minimize fast particle losses in QS stellarators is required

- The advent of stellarator gyrokinetic codes presents a unique opportunity for improving the confinement even further in optimized stellarators

- The development of a new modern 3D edge physics code would be very beneficial

- A 3D equilibrium code that handles islands and stochastic regions, including flow shielding of rational surfaces and viscous toque on magnetic islands could clarify pressure limits

- A fully 3D extended MHD code development activity may relax some stability constraints broadening the available stellarator design space

- An integrated optimization program will be the basis for an invigorated US experimental program

3.3. Needs and Priorities for Stellarator Technology

3.3.1. General need: Optimize engineering along with physics

At the most general level, the highest priority for technology is to better integrate the engineering design with the physics design at the earliest possible stage. Till now, typically the physics design is optimized and the engineering design is adjusted to maintain the physics performance near its optimal level. This may make for an excellent physics optimization, but practical implementation of the component engineering may be highly compromised.

For example, since stellarator and helical devices are 3D entities, many of the major system components must accommodate the 3D complexity, including among others, the magnets, divertor, vacuum vessel, blanket, shield, plasma facing components, port configuration and access for diagnostics and plasma heating system, not to mention, assembly, maintenance, and replacement of components internal to the vacuum vessel.

In the following sections, these issues are discussed for several major engineering components or systems. Note that some of these issues are addressed more completely in the sections of this report specifically on divertors, PMI, and reactor issues.

3.3.2. Magnet Technology

The magnet system is among the first components to be considered during the design and optimization stage. A few of the critical items to be considered are:

The 3D geometry increases the difficulty and costs of coil winding and manufacture, as well as assembly. A 3D support structure is required. In particular the large electromagnetic Lorentz forces lead to loads in bending on the winding and the structure must support these in tension, compression, bending, and shear. Compared with a symmetric planar coil system as found in tokamaks, the 3D structure must take much greater bending and shear loads, which inevitably leads to larger structural thicknesses or more intercoil bracing.

Shielding the magnets from the neutron flux is always a formidable problem, and especially so if the magnet system is superconducting, operating in a cryogenic environment. Shielding and blanket thicknesses are very important parameters for determining the machine size on the inboard side. Lessons can be learned from the tokamak program, and from system studies of both tokamak and stellarator plants (see Section 2.8), but, again, the 3D geometry could lead to further problems due to irregular and smaller sections within the vacuum vessel, and a greater difficulty in achieving complete coverage. Extensive nuclear analysis is required to highly optimize the blanket and shield. A major consideration should be determination of the minimum plasma to coil distance, and the overall machine size and economics.

The problems described for the coil, structure, and the shield will gain the most by engineering optimization, along with the physics optimization, especially if a greater volume of these components can be made straighter or more planar. Progress along these lines is discussed in Section 2.9 but clearly more needs to be done. Another great improvement could be made if these components can be made modular, fabricated in a factory and assembled in the field. This might require the use of demountable joints in the superconducting coils, an issue presently under consideration in the U.S. for tokamak magnets. But if it can be done reliably it would go a long way to alleviating many design and assembly constraints, and lead to a much more maintainable device.

3.3.3. Divertor Technology And PMI

There are many issues to be considered for divertor technology. Although stellarators can take advantage of the vast body of research that exists for tokamaks, that will probably be insufficient due to plasma physics issues and the 3D geometry required here. Many of the issues with choice of materials, high heat flux, cooling, and pumping, are similar, but the complex shapes are a challenge for analysis, manufacturing, assembly and operation. A particular area that could be investigated for this application would be 3D printing or other additive manufacturing of metallic components. This could resolve some of the complex fabrication issues, especially for embedded cooling channels, and mounting and support points, as well as in-vessel assembly. Divertor performance modeling is much more of an issue for 3D devices than in tokamaks, and there are few validated tools available now. Thus the need for integrated modeling of 3D divertor physics with complex component geometry and cooling.

The major issues for Plasma Material Interface (PMI) include erosion and redeposition, implantation and impurities, tritium retention. The question to be answered is, are these issues different than in a tokamak? The Wendelstein 7-X experiment will open new opportunities for studying these issues once it is configured for 30-minute-long high-performance plasma pulses starting in about 2020. This will become a major focus of the U.S. collaboration on W7-X.

3.3.4. Blanket Concepts and Tritium Breeding Capacity

Tritium breeding capacity is largely a function of the blanket design, but a major consideration in calculating the TBR depends on high fidelity neutronics computations. Some efforts are already underway to enhance the coupling of 3D CAD modeling of any fusion device with 3-D neutronics codes, using DAGMC (UW). With regard to geometry, ARIES studies have shown that larger stellarator machines breed more tritium because the non-uniform blanket coverage decreases with increasing radius. These studies and 3-D analysis activities should continue to model the fine details of the blanket using UW state-of-the-art neutronics tools.

Blanket size and shape must follow the first wall shape, and these geometry and fabrication issues are already discussed in the above sections. In the U.S a liquid blanket based on a liquid metal/salt is preferred to a solid blanket. Under consideration are LiPb, Li, or FLiBe. A prominent candidate is a Dual Coolant Liquid Lead Lithium (DCLL). So far very limited R&D has been carried out on these concepts and increased effort must be given to this development for future nuclear fusion devices.

3.3.5. Materials, Fabrication, Integration, Maintenance

Stellarators are expected to have challenging engineering issues with fabrication and assembly of components, machine integration, and maintenance. As described above for the coil, structure, and shield, these areas will gain the most by engineering optimization, along with the physics optimization, especially if a greater volume of these components can be made straighter or more planar. A discussion of how this is presently being addressed is given in Section 2.9.

Aspect ratio is another important consideration. It is likely that sector maintenance will be preferred to port maintenance because it gives higher plant utilization and efficiency. Coil demountability could lead to major reduction in maintenance downtime and interchange of in-vessel components.

These areas require more study and if the R&D program results in demountable magnets, and 3D printed or advanced manufactured components, these need to be well-integrated into the overall machine design, operation, and maintenance plans.

Many of the materials issues are similar for stellarators as for tokamaks including choice of structural materials, first wall, divertor materials, and blanket and shield materials. Issues and technology needs for these are discussed in the sections above. Tokamak studies should be followed closely for development and understanding of these materials, but analysis and testing should be done for geometry, nuclear environment and plasma conditions and behavior, relevant to operation in a stellarator. Particular attention must be paid to development of low activation materials, and exclude materials containing Nb, Mo, Re, Ag, etc. New concepts for materials recycling and radwaste minimization are needed.

3.4. Issues best addressed experimentally on international facilities

In planning how to address stellarator research needs that we identify in the U.S., it is important to take into account the opportunities available to us through collaboration on overseas facilities. The main opportunities reside with Germany's Wendelstein 7-X and Japan's Large Helical Device (LHD).

3.4.1. Available resources

Wendelstein 7-X (W7-X)

Wendelstein 7-X, which began operating in late 2015, is a large, modern facility based on an optimized magnetic configuration design and superconducting magnet technology. It has a mission to validate its physics optimization strategy, which relies on alignment of vacuum magnetic flux surfaces and energetic particle drift surfaces to reduce neoclassical transport; such a configuration is called "quasi-omnigenous" (QO). As beta increases, terms cancel such that, in principle, bootstrap and Pfirsch-Schlüter currents are not generated and there is no Shafranov shift, making the configuration robust to beta changes. Currently the device is configured for pulsed operation, but it will be reconfigured 4-5 years from now with an actively-cooled divertor system to support its mission to demonstrate high-performance plasma operation for pulse lengths up to 30 minutes.

The U.S. has a growing national team collaborating on W7-X, already involving seven U.S. institutions and significant investments in equipment and analysis capabilities. The U.S. team played key roles in the first W7-X operating campaign, which ended in March 2016, leading important physics experiments and producing key data and results. Preparations for the next campaign, scheduled to begin in mid-2017, are under way. With a strong partnership now well established, it can be expanded it to address additional topics that are compatible with the W7-X capabilities and schedule.

Large Helical Device (LHD)

The Large Helical Device is of a traditional design, featuring a continuous helical winding and supplementary coils, all superconducting. Operating since 1998, it is a mature facility with good heating and diagnostics and a strong team. Typically, U.S. collaborations with LHD develop from contacts between individual researchers, in contrast to the national team collaboration model being followed on W7-X. Starting in April 2017, LHD will begin operating with deuterium plasmas for the first time, offering opportunities to investigate isotope effects through comparisons with a vast data base from years of hydrogen-only operation.

3.4.2. Opportunities to address research needs

W7-X offers the best opportunities to advance the physics and technology of 3D divertors, a topic of paramount importance for steady-state operation. The magnetic configuration features an island divertor, i.e. a chain of islands at the plasma edge intersecting a system of divertor targets and baffles designed to safely remove the heat and particle exhaust from the plasma. The U.S. has prepared diagnostics and a system of low-order field perturbation coils (so-called "trim coils") designed to vary the load distribution among the ten W7-X divertor chambers. U.S. and IPP scientists collaborate in the application of state-of-the-art edge transport modeling tools, e.g. the EMC3-EIRENE code, to design experiments and make predictions. These tools have already been used to study heat loading of plasma facing components and impurity transport in the core plasma during the first W7-X operating campaign. [Schmitz et al., Wurden] Currently the U.S. is preparing an instrumented divertor "scraper" that will be installed during the next W7-X campaign and used to test our models of edge transport in diverted plasmas. Collaboration with W7-X in this area will enable the U.S. to evaluate the divertor flux widths and the potential advantages of long connection lengths that characterize the W7-X island divertor. More generally, we can improve our understanding and models of edge plasma physics and impurity transport in diverted stellarator plasmas. In the longer term, we will collaborate in the extension of these studies to steady-state conditions.

W7-X will be equipped with carbon plasma-facing components for the next several years, a choice which facilitates robustness of the material interface during this period when high-performance plasma operating conditions are being developed. However, transformation of the device to a full metal wall environment, once a high performance scenario is established, is presently being discussed. This step will address the important question of the compatibility of such a high-Z metallic first wall and divertor material choice with the features of the divertor design and the optimized plasma core performance. The development time of such a transformation of the wall and divertor material is substantial and a sufficient scoping of basic characteristics of the plasma edge and the PMI at Wendelstein 7-X is critical to make informed decisions for this transformation. In the near term, divertor physics and core impurity transport are appropriate research foci for the U.S. A modest investment in basic capabilities for plasma-material interaction (PMI) studies, such as modeling codes and material sample exposure probes, can prepare the U.S. to take advantage of W7-X once it becomes a unique facility for PMI research in a steady-state toroidal confinement facility with relevant materials.

W7-X provides the first opportunity to experimentally test a physics-optimized stellarator configuration under high-performance plasma conditions. Though stellarator optimization methods have advanced greatly since the W7-X magnetic configuration was designed (around 1990), much can be learned by collaborating in the validation of the W7-X QO optimization strategy. Currently, W7-X relies on a U.S.-provided diagnostic for time-dependent ion temperature and impurity profile measurements. Clearly such measurements will continue to be central to assessing core plasma performance as heating power and pulse length are increased over time. In addition, the U.S. is preparing core and edge fluctuation diagnostics which will contribute to understanding of how magnetic configuration changes affect turbulence, and how turbulent transport affects overall performance.

There are good opportunities for the U.S. to use its modeling capabilities to leverage its investment in diagnostics. Predictions for W7-X from new impurity transport modeling tools can be tested using U.S. diagnostics. Gyrokinetic codes have already been used by U.S. scientists to investigate the cause of "impurity hole" conditions in LHD, where there is a net transport of impurities outward from the core. Those codes can also be used to help target and interpret measurements using U.S. fluctuation diagnostics.

Though U.S. stellarator research follows the more tokamak-like quasi-symmetric (QS) strategy for design optimization, collaboration with W7-X in evaluating their configuration enables the U.S. to develop the experimental methods for testing an optimized stellarator configuration, as well as advance the science.

Modern stellarators are designed and constructed with careful attention to accurate realization of stellarator symmetry in the magnetic configuration, and to be robust against magnetic surface breakup at high beta. However construction imperfections, even within tight tolerances, can have important effects such as island generation and unequal load distribution among the discrete divertor chambers. The W7-X trim coils can be used to systematically vary the magnitude and phase of applied $n = 1$ or $n = 2$ (toroidal mode number) perturbations. Already, these coils have been used by U.S. researchers in vacuum field mapping experiments to estimate the magnitude of intrinsic low-order field errors and show that it is consistent with expectations based on measurements of the as-built magnet geometry. Also, U.S. scientists have collaborated in work on LHD showing that, under appropriate conditions, plasmas can spontaneously heal magnetic islands that are present in the vacuum configuration. If plasma flow healing proves viable as suggested by theory, it means that magnetic surfaces in stellarators could be more robust than the conventional 3D equilibrium tools predict. Collaboration on W7-X and LHD offer opportunities to explore multiple strategies for realizing good magnetic surfaces, e.g. accurate construction, field error compensation with trim coils, and self-healing.

Both W7-X and LHD are able to operate in deuterium. In the near future, the best opportunity to investigate main-ion isotope effects is through collaboration with LHD. Until now, LHD has been limited to hydrogen operation only, but deuterium plasmas will become available for the first time in April 2017, and will become the focus of LHD experiments for the foreseeable future. Topics for collaboration identified by the LHD team at a recent meeting include isotope effects on: 1) confinement, 2) H-mode power threshold, 3) hysteresis in L-H and H-L transitions, and 4) long-range turbulence correlation.

LHD and W7-X offer capabilities to generate energetic ion populations with neutral beam injection and ion cyclotron heating. These capabilities are available now on LHD and will become available on W7-X starting with the next campaign. Energetic particle instabilities will be accessible with U.S. fluctuation diagnostics now in preparation. Lost particle measurement techniques developed in the U.S. and used in collaborations with JET can be applied to stellarator experiments as well, for example to study fusion products from DD reactions.

3.4.3. Summary of research opportunities on international devices

- A strong international collaborative research program on the large super-conducting international devices is an excellent near term opportunity for the US

- Divertor, impurity transport, and plasma-material interaction research is the primary focus of the current collaborative program on W7-X and this should continue

- Neoclassical confinement and MHD stability optimization are the design bases of the W7-X device and the US should also continue participate in the core research program

- Magnetic flux surface quality and 3D magnetic control experiments on W7-X using the US-built trim coils and power supplies should continue

- Isotope effects in stellarators is a new topic of research on LHD and the US would benefit from participation in this new experimental campaign

- Energetic particle physics studies on W7-X can help validate theoretical predictions providing a solid basis for fast particle optimization studies

3.5. Major challenges and opportunities can be addressed in a U.S stellarator initiative engaging domestic experiments.

3.5.1. The case for a reinvigorated US experimental stellarator program

In providing the basis for inherent steady-state sustainment, avoidance of deleterious transients, and minimization of recirculating power, stellarators mitigate long-standing risks to the feasibility of magnetic fusion. U.S. research and development has pioneered the study of quasi-symmetric magnetic geometries for stellarators, a concept that offers some remarkable advantages to the stellarator path to magnetic fusion energy, namely high neoclassical confinement and the possibility of large flows. For over two decades, the U.S. fusion science community participating in stellarator research has consistently advanced a program to pursue quasi-symmetric approaches to magnetic confinement. Fundamentally, these approaches seek to capture both the inherent advantages of the stellarator and the good confinement and reduced size of the idealized symmetric tokamak. Success in this line of research, which is at the frontier of fusion science, could lead to magnetic configuration solutions that are optimum for practical fusion systems. The U.S. is currently the world leader in this area. This community has kept its vision current with advances in stellarator understanding and tools and with new opportunities, most notably those afforded by Wendelstein 7-X (W7-X), an optimized (but not quasi-symmetric) stellarator which has now come into operation and stimulated renewed interest in stellarators worldwide. Our collaboration with W7-X, described in Sec. 3.4, has become a mainstay of the U.S. stellarator program, attracting newcomers to the field and providing new opportunities to work at the frontiers of stellarator research. Nevertheless, quasi-symmetric principles, which have been tested in the design and construction of NCSX, and the successful operation of HSX, provide an exciting opportunity to pursue in a U.S. stellarator initiative, fulfilling a gap in the overall world fusion research enterprise and addressing issues which cannot be examined in existing or planned facilities

The most important and challenging scientific issues of interest in stellarators are summarized in Sec. 2 of this report. A more detailed exposition of most of these issues that motivate current stellarator research can also be found in the series of white papers and presentations [1] to the Fusion Energy Sciences Advisory Committee in 2014. Taken together, they identify the key scientific and technical issues that set the direction for a U.S. initiative in pursuit of fusion in 3D confinement systems. These issues include challenges that have a strong impact on the stellarator approach, such as energetic particle confinement, neoclassical impurity inflow, and three-dimensional divertor design. Recent theoretical and technical advances in optimization of turbulent transport and in the design of coils and divertors are also discussed, and should be exploited to realize the steady-state advantages of the stellarator. This section outlines the needs and priorities of experimental activities emerging from the discussion of the predominant issues in the further development of stellarators. Coincident with work carried out on existing U.S. facilities and ones abroad, an inclusive program at diverse levels is needed to guide the design of prospective new quasi-symmetric U.S. facilities.

3.5.2. The Existing US experimental program

The U.S. domestic stellarator experimental program is currently addressing several of the research topics discussed in section 3 at the so-called “concept exploration” scale. These physics topics range from magnetic configuration and transport studies, 3D MHD equilibrium and stability, suppression of disruptions, and basic measurements and model validation of 3D divertor physics.

Investigation of quasi-helically symmetric neoclassical transport and reduced flow damping has been shown on HSX at the University of Wisconsin, with current research focused on turbulent transport optimization, impurity transport, and measurement of 3D divertor properties. The CTH device at Auburn University is able to span the operational space from a stellarator to a tokamak/stellarator hybrid modified by significant amounts of internal ohmic plasma current for the investigation of 3D MHD equilibrium reconstruction and stability and also is investigating 3D divertor physics. In addition, there are basic physics studies of ECRH and divertor transport planned on CNT at Columbia University and 3D PMI on the HIDRA device aimed at liquid metals at the University of Illinois.

Focused experiments at the concept explanation scale will continue to have an essential role in furthering the understanding of 3D magnetic confinement science. A forward looking program balances the existing roles of these devices and would consider additional small-scale facilities as appropriate going forward.

3.5.3. National stellarator design project.

A national stellarator design project should be established as soon as possible to guide the design of the proposed new experimental facilities. A similar joint effort launched in the late 1990's produced large advances in stellarator analysis and design tools, deepened the understanding of quasi-symmetric stellarators, and produced two machine designs- for NCSX and QPS. In the intervening years there have been advances in design tools, providing new capabilities to improve coil designs and reduce turbulent transport, resulting in better designs. Continued progress in stellarator research has broadened understanding of stellarator physics and engineering. These advances all represent opportunities to improve both experiments and stellarator reactor designs. At the same time, the design goals have become more challenging– new configuration designs must integrate the core, divertor, and coils in the optimization; and reactor-relevance metrics such as alpha losses and maintainability must have greater weight in the design process. Candidate designs will be subjected to rigorous physics and engineering evaluation before proceeding with more detailed engineering development.

In order to pool capabilities and develop designs for new experiments in the most efficient manner, the task of advancing stellarator designs is best carried out by a national team, This design process must be tightly integrated to the optimization efforts described in Section 3.2. The goals and designs of new experiments must be carefully coordinated to ensure that they will work together as elements of a coherent national program with a common mission.

3.5.4. Proposed new devices

Two new U.S. experiments, including a world-class facility comparable in its scientific impact to W7-X and LHD along with a mid-scale device, will be necessary to assess the physics and demonstrate the utility of the quasi-symmetric stellarator and its role in international fusion energy development. Two devices at disparate scales are required because a flexible research device will be required to support the operational and design decisions - such as first wall material design and divertor operational parameters for the eventual flagship device (similar to the JET/ASDEX-U relationship).

A new mid-scale facility

While the design and detailed planning of a large quasi-symmetric experiment will take place over an extended period of time and take account of ongoing results from W7-X, a new medium-scale facility, intermediate between HSX and the major experiment described below, is needed now. Research on the exploratory HSX facility has demonstrated the promise of quasi-symmetry in reducing transport in stellarators, and it is advisable to capitalize on the success of the quasi-symmetric concept in a timely fashion. Therefore, the scope of U.S. experimental research on stellarators at the medium-scale should specifically be broadened to:

1. examine the physics of quasi-symmetric confinement in regimes more relevant to fusion energy.
    a. Ion-dominated neoclassical transport with hot thermal ions and low collisionality
    b. Higher density plasmas with lower neutral penetration
2. focus on issues that are not addressed in W7-X or LHD due to lack of quasi-symmetry
    a. Low flow damping
    b. Effect of nearly perfect symmetry on energetic particle confinement, impurity pinch
3. implement innovative design choices based on ongoing optimization, for example…

a. develop options for a more flexible stellarator divertor to match to an appropriate range of plasma equilibria
b. integration of quasi-symmetric stellarator core physics with a scalable divertor and plasma-material interaction (PMI) strategy.

A world-leadership class facility

A definitive international assessment of the potential of quasi-symmetric experiment requires an integrated experiment, one that can answer equilibrium, stability, divertor, and energetic-particle related issues simultaneously and self-consistently. The exact requirements can only be determined by carrying out a multi-disciplinary conceptual design activity, but examples of this class of facility abound. One can anticipate that a plasma radius in the ≥ 0.4 m range, magnetic field strength in the 2 to 4 T range, and multi-10s of MW of plasma heating will be needed. Pulse length requirements are not so easily anticipated; much can be learned about divertors and plasma evolution in ~10 s pulses, but a convincing demonstration of reliable steady-state performance will likely require minutes to hours. The design may or may not include capability for DT operation, but nonetheless must be shown to be on a path to steady state nuclear facilities that are practical with respect to engineering issues such as fabrication and maintainability.

The design and planning of this "flagship" experiment will build on advances from W7-X, LHD, and U.S. programs, and will exploit the most up-to-date understanding of stellarator physics and optimization capabilities. Its scientific basis will be developed through efforts in theory and computation, collaboration on W7-X, and well-targeted domestic experiments. U.S. stellarator theory and W7-X collaboration programs already in place can and should be readily expanded, commensurate with the needs of a major initiative. A mission need determination and approval of CD-0 for this facility should be secured as soon as possible.

Two types of quasi-symmetry, quasi-helical (QH) and quasi-axisymmetric (QA), are of primary interest to U.S. researchers and both have been extensively studied. Key characteristics of these are summarized in Section 2.1. The two approaches have a common physics basis in their approach to neoclassical optimization, but their differences give rise to different advantages and disadvantages in the context of a particular facility design. The selection between QH and QA for a given experiment must follow from a careful comparison of their relative merits in light of the experiment goals, a comprehensive process of optimization, and an open process that provides for review and acceptance by the research community.

The scientific case for a world-class U.S. stellarator initiative is based on opportunities to be exploited and gaps to be narrowed or closed as described in the preceding sections of this document. The major topics are summarized here:

Quasi-symmetric (QS) magnetic configurations

- The US leads the world in the design of quasisymmetric stellarators, which are consistent with steady state operation and that possess good neoclassical confinement and are stable at high plasma pressure

Non-resonant Divertors

- The non-resonant divertor concept has great potential and represents an opportunity for US leadership in divertor science and is complimentary to other divertor concepts being investigated on LHD and W7-X

Turbulence and Transport

- Advances in gyrokinetic codes have furnished new capabilities for nonlinear simulation of microinstabilities in the fully 3D toroidal equilibria of stellarators. The ability to simultaneously optimize for both neoclassical and turbulent transport is an important advance for magnetic fusion.

Fast particle confinement

- Newly designed experiments will demonstrate the required fast particle confinement to gain the required confidence for the design of a quasisymmetric fusion reactor system.

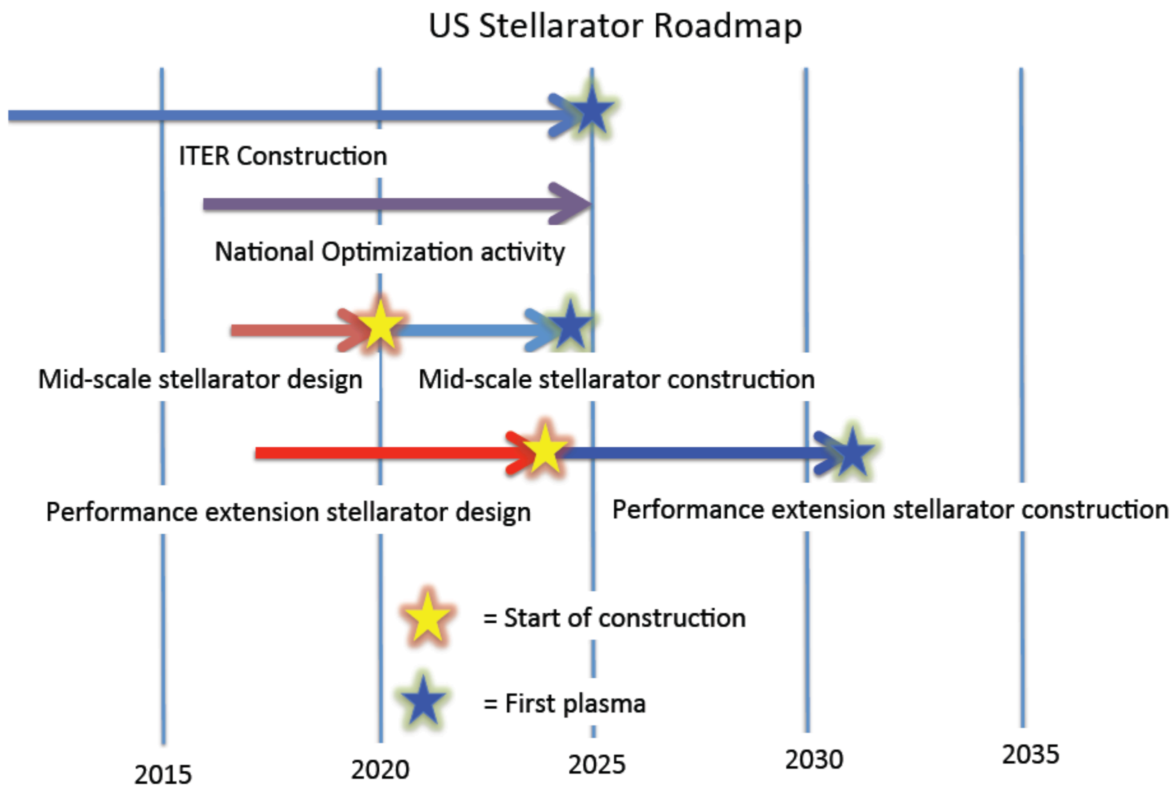

Figure 11 Possible timeline for the major elements of the proposed reinvigorated US stellarator

3.5.5. Summary

Implementation of the proposed U.S. stellarator initiative will improve the prospects for developing a suitable magnetic configuration for magnetic fusion. A possible timeline for such a program is shown Figure . A successful U.S. effort to develop quasi-symmetric stellarators could open a path to an attractive pilot plant or fusion nuclear science facility and arguably the optimum DEMO. In order to ensure success in this endeavor and impact major fusion policy decisions in time for the ITER era, it is essential to move ahead with a vigorous U.S. stellarator program starting now.

References Section 3

[1] The U.S. community stellarator initiative proposed to FESAC in 2014 may be found at http://advprojects.pppl.gov/home/stellarator-r-d/stellarator-community-initiative-2014.

4. Summary

This document outlines the physics basis and an outline for a broad-based renewed stellarator physics program for the US domestic fusion energy sciences program. It is based on a consensus view of the important outstanding issues for developing the stellarator concept as a fusion energy system that was developed in part at the "Stellcon" meeting held at MIT in February of 2016 and incorporates program elements described initially to the FESAC strategic planning subcommittee in 2014 [1]. The report is clearly aligned with the strategic priorities of FES as articulated in "Fusion Energy Sciences: A Ten-Year Perspective (2015-2025)" [2] and as outlined in the preamble to this report.

4.1. Research opportunities summary

The following important research elements have been highlighted:

- *The U.S. leads the world in the development of the quasi-symmetric stellarator concept.*

  There are three types of optimized stellarators that may lead to reactor designs, with the types distinguished by the method of obtaining confinement for trapped particles. Passing particles are well confined in all stellarator types. Quasi-symmetry (QS) is one concept for obtaining trapped particle confinement. Quasi-omnigeneity (QO), the type of stellarator that was the target of the W7-X design, is a more general concept for obtaining trapped particle confinement, which has quasi-symmetry as a special case. There are two types of quasi-symmetry: quasi-helical symmetry (QH) and quasi-axisymmetry (QA). A QH stellarator has $|B|$=constant along a helical trajectory, whereas a QA stellarator has $|B|$=constant along a toroidal trajectory. QH stellarators are apparently uniquely able to confine collisionless particles in a way that is arbitrarily close to the way they are confined in exact symmetry. A full range of QA configurations is possible, from a perturbed tokamak to a device where most of the rotational transform is produced by the 3D geometry, rather than the toroidal plasma current. All three types can be designed to have the important features of stellarators: (1) External control rather than plasma self-organization, which allows accurate computer design to speed the development of fusion energy. (2) Robust positional stability, which prevents tokamak-like disruptions. (3) No apparent limit on plasma density other than power balance. (4) Intrinsic MHD stability, including to neo-classical tearing modes. (5) A net plasma current that can be restricted to whatever level is required to avoid major runaway-electron issues. (6) An intrinsically steady-state magnetic configuration.

- *Turbulent transport optimization presents an important new development that could increase the viability of fusion energy. Strong flows expected in quasi-symmetric designs should be synergistic, adding to the interest in QS.*

  The level of turbulence in toroidal devices is strongly dependent on their shape, and through this, their magnetic field structure. In view of the strong dependence of the cost of fusion power on the level of turbulent transport, understanding and if possible reducing these levels is an issue of considerable importance. For this purpose, two powerful numerical tools for stellarators have emerged. The first of these are gyro-kinetic codes capable of simulating micro-instabilities in the 3D toroidal equilibria of stellarators. The second critical numerical tool is the stellarator optimization code. The numerical tools are now in hand to evolve stellarator (or tokamak) designs via shaping to ones with substantially reduced levels of turbulent transport, often without degrading the neoclassical transport, resulting in "turbulence-optimized" designs analogous to the "neoclassically-optimized" concepts which emerged in the 1980s and 1990s. A systematic, apples-to-apples assessment of which stellarators can achieve the lowest turbulent transport is needed, as one component of determining what an optimal reactor design should be. Additionally, the changes in turbulence with plasma shape, which are the basis for turbulent optimization, need to be tested experimentally, and compared with theoretical and numerical expectations.

- *The non-resonant divertor concept should be investigated as a complementary path to the resonant island divertor being investigated on W7-X and the helical divertor on LHD.*

All fusion reactors requires a sub-system – called the divertor - which enables them to exhaust helium as the ash of the fusion process, to control plasma density and impurity levels and to handle the heat exhaust from the plasma edge without overloading the material surfaces. The eventual heat and particle fluxes to the divertor material have to be controlled such that an acceptably long material lifetime is accomplished while enabling compatibility with good core confinement (density and impurity control). The configurational flexibility for optimized stellarators provide a large degree of freedom for design and optimization of custom-made divertor configurations. However, one challenge in stellarator edge science is to design a divertor magnetic structure, which is insensitive to effects of the confined plasma equilibrium. There a three distinct types of divertors that have been identified: 1) *Helical divertors* (e.g., LHD). The helical divertor is continuous, located between the helical coils and follows the helical geometry of the device, 2) *Island divertors* (e.g., W7-AS, W7-X, CTH). The edge transform is configured to provide a resonant value at the plasma edge. A magnetic island is formed on this low order rational surface by matching harmonics in the magnetic field structure of the device, and 3) *Intrinsic divertor* (e.g., NCSX, HSX). Generally stellarator systems have well defined exit pathways for field lines from the last closed flux surface (LCFS). The resulting interaction with the vessel wall will have the form of intrinsic 'stripes' of magnetic field line intersections when mapped from the LCFS to a vacuum vessel designed as magnetic surface offset from the LCFS and conformal to it. To date no stellarator has been optimized with full consideration of divertor and edge transport as target parameters. Hence, a fully integrated optimization, which includes optimization of the divertor system is an innovative and yet not systematically tackled research goal of stellarator edge physics. The existing expertise in the U.S. program on 3-D equilibrium modeling, plasma core optimization and stellarator divertor physics makes the U.S. program capable of acting as a leader in this field of great generic relevance to the success of stellarator reactors. The state of the art model available is the EMC3-EIRENE fluid plasma and kinetic neutral transport code. Validation of EMC3-EIRENE simulations on W7-X will be an important test of predictive capabilities, but a new code incorporating additional physics would be advantageous.

- *Stellarators naturally solve the steady-state problems of tokamaks and decouple the relationship between the heat-flux, edge density, and the current-drive power - making the stellarator an excellent configuration for PMI experiments.*

Stellarators, like tokamaks and other toroidal confinement devices, need to develop solutions for the plasma facing components, which withstand the harsh environment around a thermonuclear plasma. The assembly of relevant PMI conditions (collisionality, density and pulse length), which will be obtained first at W7-X, inherently distinguishes PMI research on long-pulse stellarators altogether. Any long pulse stellarator can deliver the relevant high flux, long pulse conditions that are required of a PMI experiment, so the need for explicit long pulse plasma exposure experiments is not a distinct priority for a PMI program in stellarators. Rather, dedicated test facilities are needed which serve as concept exploration tools to make informed decision for long pulse, high flux PMI experiments such as W7-X or on a new U.S. leadership class stellarator. This could encompass the use of metal material for the divertor target – an issue not addressed anywhere in the world program in combination with an optimized stellarator core. Similarly, development of the technology to deploy liquid lithium as a plasma facing material in steady state, such as being is tackled on HYDRA, could impact onto the two new facilities proposed in this report.

- *Energetic particle confinement issues are important for stellarators and many of them can be addressed using existing facilities like LHD and W7-X. Energetic particle optimization is an outstanding issue to be addressed by a larger optimization program.*

  Energetic particle (EP) confinement is an important issue for stellarators and remains a significant driver for 3D optimization strategies. Energetic particle populations in stellarators can arise from neutral beam heating, ICRF tails, runaway electrons, D-D produced tritons, and eventually alpha particles in D-T reactor plasmas. Since existing stellarators have not achieved the levels of EP confinement optimization required in a reactor, addressing this issue will be critical to further development of the stellarator concept. Improvement of EP classical orbit confinement in 3D configurations has been addressed using a variety of optimization approaches. These have generally been effective; for example, losses of fusion-born alpha particles in reactor-sized stellarators can be reduced from 10 to 40% levels in un-optimized systems to a few percent in well-optimized systems. W7-X and LHD both have neutral beams and RF and, by virtue of their neutral beam arrangements, are well suited to study energetic particles. Verification of numerical predictions of fast-ion losses on the large international stellarators will increase confidence in the fast-particle confinement optimizations that will be done for new stellarator designs.

- *Stellarators can be designed to be absolutely stable to ideal MHD modes up to high plasma β. Experimental verification of high beta stability can be addressed on W7-X and the optimization techniques applied to future designs.*

  MHD and high beta issues in many ways are rather different in stellarators relative to tokamaks. In tokamaks MHD instabilities provide rigorous bounds for plasma operation. Conventional stellarators are not limited by a disruptive response to MHD instabilities (by disruptions, we mean the abrupt termination of the plasma discharge characterized by temperature collapse, current quench and runaway electron generation). The external rotational transform also provides an important centering force on the plasma whereby plasma induced displacements are countered by the interaction of plasma currents with the vacuum magnetic field. Moreover, stellarators are not subject to Greenwald level density limits. The conventional model for estimating beta-limits in stellarators comes from MHD equilibrium considerations. At higher beta, large Shafranov shifts deform the flux surfaces. This deformation generates magnetic islands and stochasticity via Pfirsh-Schlüter induced resonant magnetic fields. There are a number of effects outside of ideal MHD equilibrium theory that can affect this picture. For example, finite beta can produce healing of magnetic surfaces via physics not accounted for in 3D MHD equilibrium tools. Additionally, flexibility in stellarator design make it possible to provide a neoclassical healing effect on magnetic islands. Also, if the condition for favorable neoclassical tearing mode NTM behavior is satisfied in QS, edge peeling mode and ballooning mode properties will also benefit. Needed elements in the US stellarator program include: experimental tests of ideal MHD stability in high beta/high bootstrap fraction QS stellarator. Additionally, there are a number issues that could benefit from the application of extended MHD modeling tools. These include understanding 3D MHD equilibrium physics, quantifying island healing physics and understanding the consequences of breaching instability boundaries in optimized stellarators.

- *Impurity accumulation is an important topic for stellarators to study both in existing machines and in future designs.*

  Impurity control is a serious concern in stellarators. Stellarators generally have a robust inward neoclassical impurity pinch. The situation is fundamentally different in the case of perfect quasisymmetry or axisymmetry. In symmetric geometry the main impurity pinch mechanism is absent. While these favorable properties apply to *perfect* quasi-symmetry, it is unclear if they can be achieved in experimentally relevant plasmas that are *imperfectly* quasisymmetric, as is inevitably the case. It is experimentally observed that core impurities can limit the density during long pulse operation in stellarators. However, two low-impurity regimes have been observed experimentally, neither of which is well understood theoretically. One regime is the high-density H-mode (HDH) observed in W7-AS and the other is the “impurity hole” regime in LHD. The transition from a stellarator-like impurity pinch to tokamak-like impurity screening near axi/quasisymmetry needs to be studied. Experiments should be conducted on W7-X looking for impurity-expelling regimes, and for similarities with the LHD and W7-AS regimes. Careful comparison of measurements with simulations should be used to verify our understanding and ability to predict how to achieve impurity control by design.

- *The large design space of stellarators should be explored more fully considering device engineering and maintenance, applying constraints on the coils and their locations to make stellarators more attractive as potential power plants.*

  From the early 1980s to the mid-2000s, seven conceptual stellarator power plants have been designed in the US, Germany and Japan. A major conclusion of these designs was that coil design and simplification – particularly with a focus on maintenance and construction – would improve the attractiveness of the stellarator concept. Maintaining the plasma coil separation was also a major cost driver. Recent code development efforts aimed at improving maintenance access have increased interest in coil simplification for stellarators. Including engineering constraints inside the coil design algorithms is an important conceptual step towards achieving attractive designs. Another important aspect of stellarator coil design is the fact that stellarator coils often need to be relatively close to the plasma. The issue of small plasma-coil separation becomes even more important in a reactor, because a blanket and neutron shielding must fit between the plasma and coils. Any theoretical or numerical advances that lead to plasma shapes that permit larger plasma-coil separation will have a significant impact on the cost of future stellarator experiments and reactors. Several new magnetic field ‘efficiency’ metrics have been identified and may be useful in simplifying coil designs. Finding better coil designs that are developed using both engineering constraints and physics optimization metrics is a promising research opportunity.

4.2. Program summary

- *Stellarator advances are strongly linked to a vigorous program in analytic theory*

  There is a growing interest in the stellarator community to understand how 3-D shaping can be used to optimize design with respect to microinstabilities and the associated anomalous transport. The same class of microinstabilities present in tokamaks (ITG, TEM, KBM, etc) are also predicted to be active in stellarators. Simplified analytic models need to be developed to understand the fundamental role 3D geometry has on influencing the linear stability properties of these modes. A generic feature of equilibria in 3-D is the presence of mixed magnetic topologies with magnetic islands and regions of magnetic stochasticity present. Analytic insight is needed to understanding the role of various non-ideal MHD effects on stellarator islands including the roles of plasma rotation, anisotropic heating conduction, polarization currents, neoclassical effects etc. Stellarators have exceeded stability limits for pressure driven MHD instabilities without deleterious effects. Understanding how ideal MHD instabilities evolve nonlinearly in stellarator configurations is an open question that needs attention. Understanding how and what level of 3-D shaping is needed to avoid disruptivity is an open question that would benefit from analytic insight. Conventional neoclassical transport theory predicts impurity accumulation due to the formation of a self-consistent radial electric field. However, there are examples in stellarator experiments where impurities are expelled from the core in sharp contrast to the neoclassical picture. A model is needed to explain these observations. Traditional numerical tools used to describe MHD equilibrium properties use a Fourier representation. Since this is not convenient for describing the sharp edges expected in intrinsic divertors, a different theoretical representation would need to be developed. There is a need to understand how one can modify the magnetic field structure to minimize energetic particle loss. While we have some idea of how to characterize deviations from quasi-symmetry in deducing thermal particle transport in the low collisionality regime, we do not have an appreciation for how close to quasisymmetry (i.e., how small off-diagonal elements of the magnetic field spectrum) is required to take advantage of the beneficial effects of flow and flow shear for suppression of turbulent transport, island healing and impurity transport. Each of the topics listed here are elements of a vigorous stellarator theory program.

- *Improved stellarator physics understanding depends on high fidelity simulation tools*

The magnetic configurations forming the basis of stellarator designs are typically arrived at using an optimization suite, with STELLOPT being the standard in the US. Good codes exist to compute energetic particle transport, and indeed the tracking of alpha guiding-center trajectories is a particularly ‘clean’ problem for which the equations are robust and modeling should be highly reliable so efforts aimed at optimization of energetic particle confinement are likely to yield improved designs. Another need is a 3D equilibrium code that handles islands and stochastic regions and is sufficiently fast to be used routinely in the analysis of experimental data, in optimization studies, and in the planning of experimental campaigns. Significant progress has been made in the past several years in extending gyro-kinetic studies to include 3D systems, and exploring the ability to use 3D shaping to tailor the turbulent transport properties much in the way that neoclassical transport is controlled. Overall the potential of the computationally intensive field of turbulent transport optimization is extremely high, as new simulation tools offer the possibility of designing the full transport properties—both neoclassical and turbulent—into the configuration and producing systems with significantly better plasma confinement than is presently accessible in either tokamaks or stellarators. The code set for stellarator divertor physics requires development, as the existing tools have significant shortcomings in their capabilities. The geometric optimization for divertors has made significant strides recently - the next step along this research path is to incorporate such optimization within codes like STELLOPT. High-priority research for stellarator includes exploring the formation of islands and understanding how they can be avoided as part of the optimization process. Further, the observed resilience to macroscopic stability limits as seen in experiments should be subjected to much more intensive study to test if physics behind this and the general robustness of the stellarator against catastrophic instabilities (i.e., disruptions) extrapolates to reactor-scale devices. A major focus of simulation efforts in the near future should be on validation against experiments. This should take advantage of both the modestly-sized existing domestic devices, as well as the various international stellarators--especially the large W7-X experiment now in operation. In the near term, emphasis should be placed on validating the physics that could form the basis of next-generation optimizations; for example, the ability to predict core turbulence or divertor plasma conditions should be tested. Each of the topics listed here are elements of a vigorous computational program that would enable development of improved stellarators.

- *Technology development for stellarators largely parallels the requirements of other fusion reactors, with specific development needs for tools that combine engineering and physics optimization*

  At the most general level, the highest priority for technology is to better integrate the engineering design with the physics design at the earliest possible stage. Up to now, typically the physics design is optimized and the engineering design is adjusted to maintain the physics performance near its optimal level. The problems described for the coil, structure, and the shield will gain the most by engineering optimization, along with the physics optimization, especially if a greater volume of these components can be made straighter or more planar. Another great improvement could be made if these components can be made modular, fabricated in a factory and assembled in the field.

- *Given the billion-dollar level investments in W7-X by Germany and in LHD by Japan, international collaboration is the only way to participate in stellarator fusion science at the performance extension scale for the near term.*

  W7-X offers the best opportunities to advance the physics and technology of 3D divertors, a topic of paramount importance for steady-state operation. W7-X provides the first opportunity to experimentally test a physics-optimized stellarator configuration under high-performance plasma conditions. Collaboration with W7-X in the area of divertor physics will enable the U.S. to evaluate the divertor flux widths and the potential advantages of long connection lengths that characterize the W7-X island divertor. More generally, we can improve our understanding and models of edge plasma physics and impurity transport in diverted stellarator plasmas. W7-X will be equipped with carbon plasma-facing components for the next several years, a choice which facilitates robustness of the material interface during this period when high-performance plasma operating conditions are being developed. However, transformation of the device to a full metal wall environment, once a high performance scenario is established, is presently being discussed. This step will address the important question of the compatibility of such a reactor relevant first wall and divertor material choice with the features of the divertor design and the optimized plasma core performance.

  The Large Helical Device (LHD) is a mature facility with good heating and diagnostics and a strong team. Starting in April 2017, LHD will begin operating with deuterium plasmas for the first time, offering opportunities to investigate isotope effects through comparisons with a vast data base from years of hydrogen-only operation.

  Collaboration on both W7-X and LHD provides opportunities to explore multiple strategies for realizing good magnetic surfaces, e.g. accurate construction, field error compensation with trim coils, and self-healing, as well as contrasting divertor concepts. LHD and W7-X offer capabilities to generate energetic ion populations with neutral beam injection and ion cyclotron heating. Lost particle measurement techniques developed in the U.S. and used in collaborations with JET can be applied to stellarator experiments as well, for example to study fusion products from DD reactions.

- *A targeted US effort aimed at developing new optimized stellarator designs that combines the ideas described in the research opportunities section of this document are very likely to lead to more attractive stellarator reactor concepts.*

A national stellarator design project should be established as soon as possible to guide the design of the proposed new experimental facilities. A similar joint effort launched in the late 1990's produced large advances in stellarator analysis and design tools, deepened the understanding of quasi-symmetric stellarators, and produced two machine designs- for NCSX and QPS. In the intervening years there have been advances in design tools, providing new capabilities to improve coil designs and reduce turbulent transport, resulting in better designs. At the same time, the design goals have become more challenging– new configuration designs must integrate the core, divertor, and coils in the optimization; and reactor-relevance metrics such as alpha losses and maintainability must have greater weight in the design process. In order to pool capabilities and develop designs for new experiments in the most efficient manner, the task of advancing stellarator designs is best carried out by a national team.

- *Attractive opportunities for investments in research tools are identified at both the medium-scale and large-scale that could place the US in a world-leadership position in stellarator research.*

Two new U.S. experiments, including a world-class facility comparable in its scientific impact to W7-X and LHD along with a mid-scale device, will be necessary to assess the physics and demonstrate the utility of the quasi-symmetric stellarator and its role in international fusion energy development. Two devices at disparate scales are required to operate simultaneously because a flexible research device will be required to support the operational and design decisions - such as first wall material design and divertor operational parameters for the eventual flagship device (similar to the JET/ASDEX-U relationship).

The scope of U.S. experimental research on stellarators at the medium-scale should specifically be broadened to:

1. examine the physics of quasi-symmetric confinement in regimes more relevant to fusion energy.
    - a. Ion-dominated neoclassical transport with hot thermal ions and low collisionality
    - b. Higher density plasmas with lower neutral penetration
2. focus on issues that are not addressed in W7-X or LHD due to lack of quasi-symmetry
    - a. Low flow damping
    - b. Effect of nearly perfect symmetry on energetic particle confinement, impurity pinch
3. implement innovative design choices based on ongoing optimization, for example…
    - a. develop options for a more flexible stellarator divertor to match to an appropriate range of plasma equilibria
    - b. integration of quasi-symmetric stellarator core physics with a scalable divertor and plasma-material interaction (PMI) strategy.

A definitive international assessment of the potential of quasi-symmetric experiment requires an integrated experiment, one that can answer equilibrium, stability, divertor, and energetic-particle related issues simultaneously and self-consistently. The scientific case for a world-class U.S. stellarator initiative is based on opportunities to be exploited and gaps to be narrowed or closed as described in the preceding sections of this document. The major topics are summarized here:

Quasi-symmetric (QS) magnetic configurations

- The US leads the world in the design of quasisymmetric stellarators, which are compatible with steady state operation and that possess good neoclassical confinement and are stable at high plasma pressure

Non-resonant Divertors

- The non-resonant divertor concept has great potential and represents an opportunity for US leadership in divertor science and is complimentary to other divertor concepts being investigated on LHD and W7-X

Turbulence and Transport

- Advances in gyrokinetic codes have furnished new capabilities for nonlinear simulation of microinstabilities in the fully 3D toroidal equilibria of stellarators. The ability to simultaneously optimize for both neoclassical and turbulent transport is an important advance for magnetic fusion.

Fast particle confinement

- Newly designed experiments will demonstrate the required fast particle confinement to gain the required confidence for the design of a quasisymmetric fusion reactor system.

4.3. Conclusion

With a focused effort to complete the described research plan, the stellarator concept could solve many of the difficulties faced by tokamak devices and offer a viable path to fusion energy production. The advent of the Wendelstein W7-X has created a unique window of opportunity for action by raising the profile of the stellarator research field. It is hoped that this document can serve as the basis for continued discussions within the fusion community and with the Department of Energy aimed at advancing stellarator research in the US. Timely investment in a well-coordinated national program could help the US regain the lead in the world stellarator research program.

References section 4

[1] The U.S. community stellarator initiative proposed to FESAC in 2014 may be found at http://advprojects.pppl.gov/home/stellarator-r-d/stellarator-community-initiative-2014.

[2] http://science.energy.gov/~/media/fes/pdf/program-documents/FES_A_Ten-Year_Perspective_2015-2025.pdf

5. Appendix A: National Stellarator Coordinating Committee (NSCC) Membership

NSCC Chair: David A. Gates, Princeton Plasma Physics Laboratory
NSCC Co-chair: David Anderson, University of Wisconsin-Madison

Members:
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