Stellarators as a fast path to fusion energy

Allen H. Boozer
Columbia University, New York, NY 10027
ahb17@columbia.edu
(Dated: January 1, 2020)

The speed with which fusion can be developed is important for the mitigation of the physical and political risks associated with the continual increase in atmospheric carbon dioxide. Remarkably, a stellarator demonstration reactor (DEMO) could be based more reliably on the empirical information available now than a tokamak DEMO on the empirical information that ITER is likely to provide over the next twenty years. Waiting imposes major risks on society and is unnecessary. Readiness is sufficiently revolutionary that many will assume that it can be dismissed without thought. But, those who are willing to read and to think will find the arguments simple and self-evident. The paper explains not only how stellarators are better positioned now than any other concept for a fast path to fusion but also areas in which further improvements can be quickly made to stellarators through computational design.

I. INTRODUCTION

Fusion could have a major role in reducing the enormous financial costs associated with the physical and political risks of the continual increase in atmospheric carbon dioxide. Each person in the fusion community should consider how rapidly the feasibility of fusion energy could be demonstrated.

The time required to demonstrate fusion feasibility using stellarators is primarily determined by the availability of financial resources. Nonetheless, the required resources are several orders of magnitude smaller than those prominently discussed in the press for addressing the issue of carbon dioxide increase. The time required to demonstrate fusion feasibility using other concepts, inertial or magnetic, are problematic because of proof of principle issues that remain to be resolved.

The reason for the fundamental difference between stellarators and other fusion concepts is simple but profound. The plasma in a stellarator is externally controlled while the plasma in other fusion concepts, such as tokamaks, is self-organized. This difference leads to two distinct research paradigms: optimization using computer design for stellarators and extrapolation from one generation of experiments to another for tokamaks. The difference between the two paradigms is accentuated by the order of magnitude difference in the number of external magnetic field distributions that can be used to control and optimize fusion plasmas in stellarators versus tokamaks.

Anyone can compare the speed with which fusion feasibility can be demonstrated for stellarators versus tokamaks by making two lists: (1) the issues that must be addressed by tokamaks that are already addressed by stellarators and (2) the issues that must be addressed by stellarators that are already addressed by tokamaks.

The tokamak issue of disruptions with accompanying relativistic electrons is widely understood to require a solution before ITER can address its mission. Despite decades of effort, a solution that is considered reliable is yet to be found. Until a reliable solution is found, the time scale for a tokamak demonstration reactor cannot be credibly estimated. Stellarators have no similar issue.

Both tokamaks and stellarators have issues that must be addressed related to first walls and blankets. The time required to resolve these issues will be dependent on the time required to exchange components in the plasma chamber. Open access to this chamber allows alternative solutions to be efficiently tested. Stellarators can be designed with open access, but apparently tokamaks cannot.

This paper goes through issues that must be addressed related to first walls and blankets. The time required to resolve these issues will be dependent on the time required to exchange components in the plasma chamber. Open access to this chamber allows alternative solutions to be efficiently tested. Stellarators can be designed with open access, but apparently tokamaks cannot.

Section II discusses the role of fusion in addressing the issues of the increase in atmospheric carbon dioxide. Both carbon-free sources of energy and direct carbon dioxide removal are needed. The status of direct removal is summarized in Appendix A.

Section III on the stellarator as a path for rapid development of fusion energy is the core of the paper and summarizes the distinctions between stellarators and tokamaks.

The sections following Section III provide additional details on important topics and areas in which major improvements can be quickly made through
computational design. Section IV discusses coil issues including the importance of open access, which has had almost no mention in the fusion literature, possibly because open access does not appear to be possible in tokamaks. Section V discusses the space of stellarator configurations, which is so large that special strategies are required to determine those most suitable for fusion reactors. Section VI discusses strategies for dealing with the implications of microturbulent transport on reactor design. A method of assessing the implications is developed in Appendix B. This appendix shows that gyro-Bohm scaling fits the overall scaling laws of both tokamaks and stellarators with remarkable accuracy and derives implications for reactor design. Section VII considers divertors and the protection of the walls from alpha-particle damage in stellarators. Section VIII discusses technical developments in the areas of coils, liquid films for the walls, solid walls, and breeding blankets.

II. FUSION PROGRAM MISSION

Science could offer society options for addressing the continual increase in carbon dioxide in the atmosphere: new sources of carbon-free power, such as fusion energy, and direct removal of carbon dioxide from the air, Appendix A.

Deploying solutions is expensive; the replacement of the worldwide electrical generation capacity would require tens of trillions of dollars. Research to develop and demonstrate better solutions is far cheaper, a demonstration stellarator fusion reactor would cost tens of billions of dollars. The development of design concepts is orders of magnitude cheaper still.

The sometimes apocalyptic descriptions of the implications of the steady increase in carbon dioxide are unleashing powerful political forces but have nonetheless aroused little interest in the solutions that science could offer. Restoring the carbon-dioxide level on a credible time scale will require both direct removal of carbon-dioxide from the atmosphere and carbon-free energy sources. In the United States, the Department of Energy considers both to have the priority of basic-science issues, such as the 4% discrepancy between measures the proton radius on which recent progress has been made.

The carbon-dioxide problem must be solved worldwide, or it is not solved at all. The United States produces only 16% of the carbon dioxide released into the atmosphere, and a large reduction in that number would have little effect. Nevertheless, on average an American produces 3.7 times as much carbon dioxide as other persons in the world. Without carbon-free energy that can be reliably produced essentially anywhere, the concentration of carbon dioxide will increase ever more rapidly as worldwide living standards rise.

A vision: Develop fusion energy on the fastest possible time to ensure society has a safe carbon-free source of energy that is neither intermittent nor site specific.

The associated mission: Formulate plans for the construction of fusion reactors including the case in which the speed of achievement of a working power plant has priority over the annual budget. Proceed on research directed towards a power plant on the fastest possible time scale consistent with technical or budgetary limitations, whichever is more restrictive.

The continual increase in carbon dioxide is placing enormous financial risks on the annual world economy of eighty-seven-trillion dollars—whether one views the risks as primarily physical or political. The cost of developing fusion energy reactors to the point of deployment is extremely low in comparison. Not only is rapid fusion development an extremely cost effective method of mitigating the risks but would also drive the advancement of science and engineering.

Science and engineering can move rapidly to address societal risks and issues. The Manhattan project is the best known example. The splitting of uranium was discovered in Berlin in December 1938 and the first atomic bomb was exploded in New Mexico in July 1945, less than seven years later. The Apollo program to land and return a person from the moon is another example. The program was announced in May 1961 and reached it goal in July 1969.

III. STELLARATORS AS A PATH FOR THE RAPID DEVELOPMENT OF FUSION

The stellarator, among all fusion concepts, has properties that best open a fast and reliable path to reactors. These properties can be illustrated by comparing stellarators with tokamaks. Far more tokamak than stellarator experiments have been performed, but far more details about fusion plasmas are required for the design of a tokamak than of a stellarator reactor.

The extra information required for tokamak reactors makes the step from ITER to a demonstration power plant (DEMO) appear more difficult than go-
ing from our present understanding of stellarators to a stellarator DEMO. Open questions that ITER will address are summarized in a 2019 article by Hawryluk and Zohm in Physics Today [1] and the issues that are being considered for the European tokamak demonstration power plant are reviewed in a 2019 article in Nuclear Fusion [2].

1. No proof-of-principle issue, such as disruption avoidance in tokamaks, blocks rapid development of stellarators.

Disruptions are an existential threat to reactor-scale tokamaks, particularly the threat of strong currents of relativistic electrons. Nevertheless, disruptions receive only cursory consideration in the papers on open questions that will be addressed by ITER [1] and on the European tokamak demonstration reactor [2].

As discussed below, no solution that is generally perceived to be reliable is known for disruptions, which makes all tokamak planning problematic. A common presumption is that the disruption problem will be solved because it must.

2. The stellarator is unique among all fusion concepts, magnetic and inertial, in not using the plasma itself to provide an essential part of its confinement concept.

This allows stellarators to be designed computationally with far more reliability than any other fusion concept.

The alternative to computational design is extrapolation from one generation of experiments to another, as is traditional in tokamaks. The abstract of the original paper on the ITER Physics Basis emphasized extrapolation [3]. The paper that introduced the scientific basis of W7-X emphasized computational optimization of designs [4].

Four disadvantages of extrapolation in comparison to computational design are:

(a) Experiments build in conservatism—even apparently minor changes in design are not possible and therefore remain unstudied.

(b) Experiments are built and operated over long periods of time—often a number of decades.

Multiple experiments carried out at the same time do not delay development, but extrapolation using consecutive generations of experiments does. The need for consecutive generations of experiments should be minimized.

(c) The cost of computational design is many orders of magnitude smaller than building a major experiment, as well as having a much faster time scale.

(d) Extrapolations are dangerous when changing physics regimes. Examples are (i) plasma control in ignited versus non-ignited plasmas and (ii) the formation of a current of relativistic electrons during a disruption.

In existing tokamaks, external heating provides plasma control that is not present when the heating is dominated by DT fusion. As will be discussed, a stellarator has far more available degrees of freedom for control but requires fewer than a tokamak.

Although relativistic electrons are an existential threat to tokamak reactors, the danger is removed in standard stellarator designs. In stellarators, the plasma is robustly centered within the confinement chamber, and the effect of plasma currents on the rotational transform, which is the derivative of the poloidal relative to the toroidal flux, is minimized. The loop voltage, which accelerates electrons, is the rate of slippage of the poloidal relative to the toroidal magnetic flux.

In tokamaks, a loss of position control accompanies disruptions and each megaampere of decay in the plasma current can increase the current in relativistic electrons by a factor of ten. As noted in Chapter 3, Table 5, of the ITER physics basis [5], this implies an amplification of a current of relativistic electrons a trillion times greater in ITER than in JET.

Recent theoretical work [6] indicates the amplification may be far larger than formerly expected.

The danger of electrons running away to relativistic energies has been prominent in the literature for more than twenty years, but no method has yet been devised of mitigating the danger in a way that is perceived to be reliable. The last
paragraph of the 2019 Nuclear Fusion review of the physics of runaway electrons (RE) noted: With ITER construction in progress, reliable means of RE mitigation are yet to be developed. Sections 10.8 and 11.3 of this review discussed the wall damage that runaway electrons can produce. A runaway current as small as 300 kA could cause the melting limits of a wall panel roof to be exceeded. This is a worse-case number, and several megaampere instabilities of relativistic electrons striking the walls may be required for extreme damage.

The severity of the damage that can be produced by even a single relativest-electron incident implies: (i) The achievement of the ITER mission will be difficult when more than one such incident occurs in a year. (ii) The strategy for avoidance must be fundamentally based on theory and computation.

Tokamak disruptions are often said to result from exceeding operating boundaries\[1\]. Unfortunately, methods of steering tokamak plasmas away from operating boundaries during a fusion burn are extremely limited and slow, seconds for the temperature and density profiles and minutes for the current profile.

Steering alone does not provide complete protection against disruptions. A disruption can be initiated if a part of a wall tile or even a tiny flake from the tungsten divertor targets were to enter the plasma. A tile falling into a stellarator plasma would also cause a rapid drop in the plasma pressure. Stellarators should be designed so currents associated with the diamagnetic effect of the plasma and the quick loss of the plasma energy through radiation would be tolerable. These requirements are far less demanding than those of a tokamak.

3. Stellarator reactor designs are only weakly dependent on the plasma pressure profile.

(a) The sensitivity of tokamaks to the profile of the current density makes them highly sensitive to the pressure profile, through the bootstrap current, and in non-steady-state reactors to the temperature profile, through the resistivity.

(b) Microturbulent transport is an issue for all magnetic-fusion systems. The insensitivity of stellarators to the pressure profile implies that only the overall level of the transport is of central importance. For tokamaks, not only is the overall level important, but also the radial dependence of the transport.

(c) The overall effective transport rate can be normalized to gyro-Bohm transport by a coefficient $D$, Appendix 2. The $H$-factor of the tokamak literature scales as $H \propto 1/D^{2/5}$.

Too large a $D$ implies that either the power output of a single reactor or the magnetic field strength become excessively large. Too small a $D$ implies the plasma radius is small compared to the thickness of the blankets and shields, which means the power production is too small compared to the reactor cost. The problem of too small a $D$ could be addressed by reducing the plasma reactivity by departing from a 50/50 deuterium-tritium mixture. $D$ of order but somewhat smaller than unity is optimal.

(d) Stellarators do not have constraints such as the Greenwald Limit on the plasma density or the high electron temperature required in tokamaks for current maintenance. The higher the electron temperature the greater the number of energetic alpha particles, which increases the sensitivity to energetic particle instabilities. As discussed in Appendix 2 the degradation of confinement with power seen in empirical scaling laws implies a degradation of confinement with temperature.

4. Stellarators offer far more freedom of control than do tokamaks.

Approximately fifty externally produced distributions of magnetic field are available for plasma control in stellarators; approximately five are available in axisymmetric tokamaks. Unlike in stellarators, these require careful time-dependent control.

The plasma profiles in tokamaks require far more control than in stellarators, but the available degrees of freedom to provide that control are far fewer.
5. The coil systems in stellarators, unlike those in tokamaks, can be designed for open access to the plasma chamber, Section [V.A]

If fusion is to be developed rapidly, a demonstration reactor (DEMO) must be designed to allow first wall components to be changed quickly—too many uncertainties remain in first wall materials [8,9], in concepts such as walls being covered by liquids [10], and in blankets for breeding tritium [11] for it to be otherwise.

Open access also shortens maintenance times in operating reactors.

Stellarators do have the disadvantage of a larger aspect ratio, the ratio of the major to the minor radius, \( R/a \), of the torus. The power density on the walls \( p_w \), megawatts per square meter, should be as large as is consistent with a reasonable wall lifetime to minimize the cost of fusion per kilowatt hour. The total power output \( P_T \) of a single fusion reactor would ideally be small to maximize flexibility and minimize the capital required for the construction of a single unit. The obvious relation \( P_T \propto (R/a)a^2p_w \) couples a high power density with a high total power output. The aspect ratio \( R/a \) is determined by the fundamental properties of a fusion concept, and the aspect ratio of stellarators is several times larger than that of tokamaks. The minor radius \( a \) cannot be too small compared to the thickness of blankets and shield around a fusion plasma, but otherwise it is determined by transport.

Appendix B shows that the empirical energy-confinement scaling of tokamak and stellarator experiments closely match what would be expected if the diffusion coefficient were a factor \( D \) times gyro-Bohm transport. The required plasma radius squared scales with the quality of confinement \( 1/D \), the magnetic field strength \( B \), the power density on the walls \( p_w \), and the central plasma temperature \( T_0 \) as

\[
a^2 \propto \frac{D^{4/5}T_0^{6/5}}{B^{8/5}p_w^{2/5}}.
\]

For a given quality of confinement, wall loading, and aspect ratio, the total power output \( P_T \propto a^2 \) can be made smaller by using a larger magnetic field and a lower plasma temperature—as long as \( T_0 > 10 \) keV. The higher central temperature in tokamak reactors offsets the advantage of a smaller aspect ratio for allowing fusion power plants to have a smaller total power output \( P_T \). Tokamak reactor designs often require a significant fraction of this power output to be used for maintenance of the plasma current and for control while stellarators do not. The larger recirculating power fraction is a significant burden on the economic viability of tokamak reactors.

Early operations of the W7-X stellarator achieved \( D = 0.13 \) in ten-second steady-state plasma conditions, which yields attractive reactor designs, and \( D = 0.05 \) during short intervals. DIII-D has carried out long-pulse tokamak experiments that achieved \( D = 0.31 \). Both the stellarator and tokamak results for \( D \) are discussed in Appendix B.4.

IV. COILS FOR STELLARATOR REACTORS

Three properties of the coils that produce the external magnetic field are of particular importance to a rapid and reliable development of fusion energy: (1) Coils that offer easy access to the plasma chamber. (2) Coils that are relatively easy to manufacture because the magnetic fields that they produce are not unnecessarily strong nor rapidly varying in space. (3) Coil systems that maximize the flexibility of plasma control [12].

A. Coils with easy chamber access

Coils systems for both tokamaks and stellarators have been designed in a manner that makes access to the plasma chamber extremely constrained. Though this is probably required for tokamaks, there is no
known reason why this must be true for stellarators.
The red coil in Figure 1, which is from [13], is the
only coil that need encircle a stellarator plasma and
limit plasma access. Mathematics ensures the rest of
the magnetic field could be produced by coils, each
shaped like a windowpane, with some embedded in
the removable sections of the walls [12]. There is
no necessity for the removal of wall sections to be
more restricted than that produced by the red coil
of Figure 1.

Rapid changes in the components that surround
the plasma are critical for the fast development of fu-
sion energy and would minimize maintenance time
in a reactor. Despite the obvious importance, coil
concepts such as the one illustrated in Figure 1 re-
main largely unexplored.

B. Efficient magnetic field distributions

A curl-free magnetic field decays with the distance
x from the coil that produces it as $e^{-kx}$ where k
is the wavenumber of the field. All possible exter-
nal magnetic field distributions can be ordered by
their efficiency of production [12, 14]. Stellarator
optimizations could be constrained so only magnetic
fields that can be produced efficiently at a distance
are included, which are the only magnetic fields that
can be produced by practical coils.

The benefits of limiting the design to the ef-
ciently produced external field distributions are
largely unexplored.

C. Coils needed for plasma control

The important stellarator control parameters are
the efficient magnetic field distributions. It is known
that the importance of the various magnetic field
distributions for plasma control varies widely [12].
What has not been done is to assess which of these
distributions are the most important and how the
control of these distributions can be incorporated in
coil design.

The speed and the completeness with which a
given machine allows fusion to be developed is
largely determined by its available control.

V. STELLARATOR CONFIGURATIONS

The space in which stellarators are designed has
about fifty degrees of freedom—far too many for an
optimization code to ensure that a global optimum
has been found. Although a direct and complete
optimization is impossible, practical numerical opti-
mizations can (1) refine an initial guess or (2) main-
tain the optimization of a curl-free magnetic field as
the plasma pressure is increased.

The large size of this space compared to what has
been explored is illustrated by Figure 2.

A. Identification of states for optimization

The attractiveness of an optimized stellarator is
largely determined by the initial state used in the op-
timization, which makes makes methods of choosing
an initial state of great practical importance. Two
concepts for finding advantageous initial states are
(1) a Taylor expansion around the magnetic axis and
(2) the optimization of an outer magnetic surface of
a curl-free magnetic field.

1. Expansion around the axis

The original idea of defining equilibria using a
Taylor expansion around the central field line in a
toroidal plasma, the magnetic axis, [15] is due to
Mercier in 1964. Taylor expansion methods were
found in 1991 to set important constraints on stel-
larators [16], and recent advances have been made
[17]. Unlike axisymmetric systems, the achieve-
ment of adequate particle confinement is the primary
physics issue in stellarators. Methods of achieving
particle confinement, quasisymmetry and omnigen-
ity, are discussed in [18–20].

2. Optimization of an outer surface

An outer magnetic surface of a curl-free magnetic
field can be found [21] that has desirable confinement
properties such as exact quasisymmetry. Quasisym-
metry gives tokamak-like confinement of individual
particles. This method is particularly important
in conjunction with the concept of annular design,
which is discussed in Section V B 3 and in Appendix
B 3.

The shape of the outer magnetic surface is deter-
mined by three functions of two angles that must
satisfy constraints. Using $(\hat{R}, \hat{\zeta}, Z)$ cylindrical co-
dinates, the three functions are $\hat{R}(\theta, \varphi)$, $\hat{\zeta} = \varphi + \omega(\theta, \varphi)$, and $Z(\theta, \varphi)$, where $\theta$ and $\varphi$ are the
FIG. 2: Matt Landreman and his group have used analytic expansions around the magnetic axis to survey the landscape of possible quasisymmetric stellarators. The figure shows a database of $2.4 \times 10^8$ quasisymmetric stellarator configurations. The few designated points on the right side of the figure indicate previously known stellarator configurations.

Poloidal and toroidal angles in Boozer magnetic coordinates in which the constraint of exact quasi-symmetry is easily specified. One function of the two angles is required to obtain magnetic coordinates. Obtaining well confined particle trajectories constrains half of the freedom of another function of the two angles. Quasiaxisymmetry is obtained when the field strength $B$ has the property that $B(\theta, \varphi) = \int B(\theta, \varphi) d\varphi / 2\pi$. The curl-free solution can be extended throughout the volume enclosed by the surface by choosing efficient magnetic field distributions so that the magnetic field perpendicular to the optimization surface zero. Maximizing the coil efficiency is equivalent to placing another constraint on a function of the two angles. There are only two-and-a-half functions of constraints on the three functions of $\theta$ and $\varphi$. Consequently, there is additional freedom in the properties of the magnetic field.

This method of defining curl-free states for optimization is unexplored.

### B. Annular Design

An optimal design for a stellarator may have low plasma transport in the outer half of the minor radius, but such rapid transport in the inner half that the pressure is essentially constant there, Appendix B.3. The implications are essentially unexplored, but there are advantages to having the confinement produced by an outer annulus. (1) A spatially constant pressure $p$ maximizes $\int p^2 d^3x$ for a fixed maximum pressure, which maximizes the fusion power. The confinement time of the plasma is the ratio of the total plasma volume to the volume of the annulus longer than the confinement time of the annulus. (2) Impurities tend to be flushed out more readily the narrower the confinement annulus compared to the total confining volume. (3) The injection of fuel is easier. (4) The optimal 50/50 DT ratio can be maintained in the fusing plasma, which is not trivial when transport coefficients are small in core. (5) Control of the width of the annulus would provide an important control of the plasma.

The situation in tokamaks with good confinement is related but different. The pressure drops across the core of a tokamak, but there is a narrow region right at the plasma edge, where a transport barrier arises that creates a pedestal, which raises the plasma pressure everywhere inside. See Figure 2 in [1] and the related discussion. This annulus naturally has periodic instabilities called Edge Localized Modes (ELM’s), which must be mitigated to avoid unacceptable damage to the chamber walls, [5]. The extent to which such pedestals arise in stellarators, or whether it is even desirable that they occur, is unclear. The self-organized state of an axisymmetric tokamak plasma implies the control over important features such as the pressure profile is
limited. Carefully designed non-axisymmetric perturbations that preserve the quasisymmetry of the tokamak core could ameliorate this limitation [22]. An example is the control of ELM’s by long wavelength non-axisymmetric magnetic fields [23].

VI. MICROTURBULENCE STRATEGIES

An uncertainty in the design of both stellarator and tokamak fusion reactors is microturbulent transport. The physics of microturbulence in stellarators was reviewed in 2015 by Helander et al [24]. The two most important types of microturbulence are the ion-temperature-gradient (ITG) mode and the trapped electron (TE) mode. The TE instability has much greater stability when the trapped electrons are primarily in a region of good magnetic field line curvature as in W7-X. In tokamaks and quasi-axisymmetric and quasi-helically symmetric stellarators the trapped electrons are primarily in a region of bad curvature.

ITG microturbulence appears to have a beneficial effect of expelling impurities, which implies some level is desirable. But, in a reactor ITG turbulence can not be so large that it unacceptably degrades ion confinement. Unlike the situation in tokamaks, the details of the pressure profile in stellarators are of little relevance. ITG turbulence need only be kept at a level that is consistent with an adequate fusion product, \( n \tau_E T \), where \( n \) is the number density of the deuterium and tritium ions, \( \tau_E \) is the energy confinement time, and \( T \) is the temperature.

As discussed in Appendix B, power-law scaling relations hold with remarkable accuracy for tokamaks and stellarators. Nevertheless, scaling relations do not provide the certainty that is wanted for a reactor design—even in stellarators. The effect of microturbulence is not well understood in either tokamaks or stellarators. Non-linear calculations of microturbulence using the GENE code [24] show a W7-X case with a transport enhancement of twenty times the characteristic gyro-Bohm value and a DIII-D case with an enhancement of two-hundred times.

In designing a stellarator reactor, the most important information on microturbulence is what factors are beneficial in obtaining an adequate \( n \tau_E T \). A higher magnetic field strength \( B \) appears to be beneficial. The ion temperature gradient, \( d \ln T_i / dr \) times a spatial scale is an instability factor, but what that spatial scale is is not agreed upon. It could be related to the average magnetic field curvature, the local shear, or the global shear in the magnetic field. A density gradient is stabilizing, so \( d \ln T / d \ln n \) should not be too large, but a sufficiently weak temperature gradient can be stable even when the density profile is flat.

To avoid impurity accumulation, it is not clear that stabilizing the trapped electron mode when ion temperature gradient mode is unstable is beneficial. This might make ion heat transport rapid compared to particle transport, which is bad. What is needed is a rapid transport of non-hydrogenic ions relative to the heat transport in a plasma that primarily has hydrogenic ions.

What seems to be agreed upon is that linear instability theory is a poor surrogate for relative levels of microturbulent transport [25]. The ITG mode can be stabilized by zonal flows [20], though in stellarators wave coupling in other forms than zonal flows may be more important [27].

Stellarators can be designed for optimal microturbulent transport [28], but such optimizations require a surrogate. Full simulations of microturbulence are too time consuming to be practical. The reliability of full gyrokinetic simulations is debated, but to the extent that they can be taken to be reliable they could be used (1) to test whether a given stellarator configuration has acceptable transport properties and (2) to determine which features of the magnetic configuration have the greatest effect on the microturbulent transport. The microturbulence codes GENE, XGC, and GTC have been primarily developed for tokamaks but stellarator versions, such as XGC-S [29], are being developed.

Plasma confinement can be greatly enhanced by transport barriers. The best known is the H-mode enhancement of tokamak confinement by approximately a factor of two by the formation of a narrow edge pedestal [1]. The formation and the stability of this transport barrier can be strongly influenced by tokamak shaping and in particular by having negative triangularity, which has the mid-plane point of the triangle on the small major radius side of the plasma [30]. Transport barriers can form not only at the edge but also in the body of the plasma [31]. Internal transport barriers are associated with rational magnetic surfaces, low or negative magnetic shear, a strong local magnetic shear, such as that produced by the Shafranov shift, and \( \vec{E} \times \vec{B} \) flow shear, which has a far stronger effect on the ion than on the electron transport.

Stellarators offer much more freedom to change the properties that control internal transport barriers than do tokamaks. Freedom from the details of the profile of the net plasma current, including the disruptions caused by that profile, imply transport
barriers are an important area for exploration. Nevertheless, existing W7-X results, Appendix B.4 imply such explorations are probably not required to build an attractive stellarator reactor other than to increase the certainty that the transport in a given design is acceptable.

There should be a focus on experiments and theory that can contribute over a time scale of years, not decades, to clarify the constraints of microturbulence on reactor design.

VII. EDGE CONTROL

A. Divertors

The particle exhaust from plasmas should be concentrated to the location of pumps, but this concentration makes the power loading on the walls intolerably high unless a large fraction of the power is radiated away.

A divertor is a magnetic structure that directs the plasma particles to the locations of pumps. A detached divertor means that radiation removes essentially all of the energy from the plasma before it contacts the wall.

Two types of magnetic structures are being considered for divertors in stellarator reactors: resonant and non-resonant.

1. Resonant divertors

A resonant divertor locates a chain of islands at the plasma edge, which requires extremely accurate control of the edge rotational transform, \( \iota = 1/q \), which is the twist of the magnetic field lines; \( q \) is the safety factor.

W7-X has a resonant divertor [32, 33], so this concept is being studied as part of the W7-X program. In particular, W7-X has demonstrated that a resonant divertor can maintain stable detachment and radiate most of the plasma energy before the plasma reaches the walls.

2. Non-resonant divertors

Non-resonant divertors use the Hamiltonian mechanics concepts of Cantori and turnstiles. Magnetic field lines obey exactly the equations of one-and-a-half degree of freedom Hamiltonian mechanics, \( H(p,q,t) \), although for field lines the three variables of the Hamiltonian mechanics are three spatial coordinates. Beyond the outermost confining magnetic surface, a double magnetic flux tube is formed in each period of the stellarator (1/2 the flux comes in and 1/2 goes out). The two parts of these tubes strike the wall at remarkably robust locations.

Non-resonant divertors have been explored less than resonant divertors, but there are several theoretical papers on non-resonant divertors [12, 34–36]. Unlike resonant divertors, non-resonant divertors place no constraint on the edge rotational transform and the width of the escaping flux tube that carries plasma to the pumps can be adjusted.

B. Protection of the walls from \( \alpha \) particles

Helium ions (alpha particles) produced by the nuclear reactions can become deeply embedded in the walls if they strike while still energetic. The accumulation of helium gas in crystal lattices creates blisters and fuzzy regions, which destroys the structural integrity of the walls.

Three strategies have been proposed for addressing this issue: (1) Apply whatever constraints are necessary on the variation of the magnetic field strength on the magnetic surfaces to limit the loss of alpha particles. (2) Design the edge magnetic field so the energetic trapped alpha particles, which are the problem, strike the wall in a location in which they harmlessly go into a liquid, such as lithium or tin, not a solid wall. The feasibility of doing this is essentially unexplored. (3) Avoid alpha-particle damage altogether by covering plasma facing components with a thin liquid film, Section VIII 2.

VIII. IMPLICATIONS OF TECHNICAL DEVELOPMENTS

Technical developments are of particular importance in four areas (1) coils, (2) liquid films for covering first walls, (3) solid first wall materials, and (4) breeding blankets for tritium. The design of a stellarator reactor that has open access to the plasma chamber requires a suitable choice for the coil system. But, when this is done, a fast development of fusion requires only that an appropriate space allocation be made for the first wall, the blankets, and shields. Several versions of these systems should be made to test various designs. The replacement of inadequate components must be part of the research on the test reactor.
1. Developments for coils

Technical developments in high-temperature superconducting coils for fusion applications was the subject of a 2018 Nuclear Fusion review [37]. This review included a discussion of use of joints in coils. The construction of both tokamaks and stellarators could be faster and cheaper if coils could be delivered in pieces that are joined during device construction.

Commonwealth Fusion Systems [38] has placed a strong focus on developing coils for fusion systems that can operate at much higher magnetic fields than those in existing tokamaks. This work is important for stellarators as well as tokamaks. The required minor radius of a plasma will be found to scale as \( a \propto D^{2/5}/B^{4/5} \) while the total power output of a reactor \( P_T \) for a given wall loading \( p_w \) scales as \( a^2 \). Higher magnetic fields allow power plants to be built with a smaller unit size and allow compensation for poor confinement, a large \( D \).

2. Development of liquid films

Even a thin layer of liquid on plasma-facing components can address four issues [10]. First, the layer can eliminate the degradation of wall materials that can be produced by fusing plasma plasmas. Examples are alpha particle degradation and the sudden flash of radiative energy that would occur if a piece of a tile fell into the plasma. Second, flowing liquids can remove the surface heat load. Third, somewhat thicker liquid layers can reduce the nuclear damage. Fourth, liquid layers can reduce gradients such as temperature and stress.

3. Development of solid walls

Although liquids can mitigate issues associated with plasma-facing components, solid walls are required even if there are liquids covering the walls. Issues that must be addressed relative to materials for the first wall are discussed in [8, 9].

4. Development of tritium breeding blankets

Major challenges and fundamentally different design choices exist for the blankets that breed the tritium burnt in fusion systems. These are reviewed in [11].

Acknowledgements

This material is based upon work supported by the U.S. Department of Energy, Office of Science, Office of Fusion Energy Sciences under Award DE-FG02-95ER54333 and by grant 601958 within the Simons Foundation collaboration "Hidden Symmetries and Fusion Energy."

Appendix A: Direct carbon removal

Thermodynamics implies that it is feasible to remove the carbon-dioxide from the atmosphere using energy sources that do not produce carbon dioxide.

If the excess carbon dioxide were all from burning coal, approximately 4% of the energy that has been produced from burning coal would be required to remove it. If the excess carbon dioxide were all from burning natural gas, about 2.5% of the energy that has been produced from burning natural gas would be required.

Though there are articles on carbon-dioxide removal [39–42], they do not state how close existing processes come to the minimum energy requirement from physics, the thermodynamic limit. The implication of these articles is that efficiency of existing processes is not extremely small compared to thermodynamic limits, nor would one expect them to be.

In principle no external energy source is required, carbon dioxide can be dealt with by exothermic reactions with minerals. A review of these processes, both in nature and in the laboratory, is given in [43].

The intrinsic cost of carbon-dioxide removal is proportionate to the efficiency. The cost of deploying a carbon-dioxide removal system is enormous compared to the cost of developing systems. A consensus that the present level of carbon-dioxide presents either physical or political risks should imply an aggressive research program. It is also true that certain specialized applications, such as intercontinental air travel, cannot easily use carbon-free energy sources, no matter how cheap they may be. Carbon dioxide removal could relatively easily offset the carbon dioxide produced by these specialized applications.

The present concentration of carbon dioxide is \( \frac{N_{\text{CO}_2}}{N} = 407.4 \times 10^{-6} \). Since \( N_{\text{CO}_2}/N \ll 1 \), the energy required to remove one molecule of carbon dioxide from the atmosphere while holding the temperature and pressure fixed is given by the chemical potential \( \mu \approx T \ln(N_{\text{CO}_2}/N) \), or more precisely the negative of the chemical potential. Since 20° Centi-
grade is 0.0272 eV and ln(N/N\textsubscript{CO\textsubscript{2}}) = 7.806, the energy required to remove one molecule of carbon dioxide is 0.212 eV. The number of moles of carbon dioxide released per 10\textsuperscript{6} joules of energy production is given in [44] for a number of fuels. For coal the number is 2.0, and for natural gas the number is 1.2.

Avogadro’s constant 6.022 × 10\textsuperscript{23} is the number of molecules in a mole and 1 eV = 1.60218 × 10\textsuperscript{-19} J, so 10\textsuperscript{6} J/mole ≈ 10.365 eV/molecule.

To remove the carbon dioxide produced by burning coal, the thermodynamic limit on the required energy is (0.212 eV)/(10.365 eV/1.2) = 2.45% of the energy produced by burning the coal.

To remove the carbon dioxide produced by burning natural gas, the thermodynamic limit on the required energy is (0.212 eV)/(10.365 eV/2) = 4.1% of the energy produced by burning the gas.

Appendix B: Fusion power and transport

A small unit size for fusion reactors, measured by the total power output \( P_T \), is in conflict with having a high power density on the walls \( p_w \) since \( P_T \propto R a \). The basic fusion concept sets the aspect ratio \( R/a \), but the minor radius \( a \) is determined by transport as long as the minor radius is sufficiently large compared to the thickness of the blankets and shields surrounding the plasma. When transport would allow a minor radius smaller than this, the DT fuel mixture could be degraded from the optimal 50/50 mixture for the reactor design to be consistent with an adequate \( a \). The diffusion coefficient below which transport becomes too small is comparable to gyro-Bohm with an enhancement factor \( D \approx 0.1 \).

The units that are used are 10 keV for temperature, 10\textsuperscript{20}/m\textsuperscript{3} for number density, Tesla for magnetic field, megajoules for energy, and seconds for time. In these units, the Boltzmann coefficient, which converts 10\textsuperscript{20} particles/m\textsuperscript{3} times 10 keV into mega-Joules per cubic meter, is \( k_B = 10^{20} \times 1.602 \times 10^{-21} = 0.1602 \). The permeability of free space \( \mu_0 = 4\pi \times 10^{-7} \) in standard scientific units becomes \( \mu_0 = 0.4\pi \) in the units that are used in this paper.

The radial coordinate \( r \) is defined so the volume enclosed by a magnetic flux surface is \((2\pi R)(\pi r^2)\) with \( R \) the major radius. The edge of the plasma is at \( r = a \), which is the standard stellarator definition of the minor radius. The standard definition of the minor radius of a tokamak, \( a_t \), has a plasma volume \( \kappa_c (2\pi R)(\pi a_t^2) \), where \( \kappa_c \) is the elongation. That is \( a_t = a/\sqrt{\kappa_c} \).

1. Deuterium-Tritium power density

John Wesson [45] gave a convenient expression for the power density of DT fusion, which holds with 10% accuracy for temperatures between 10 keV and 20 keV,

\[
p_{DT} = 0.77n^2T^2. \tag{B1}\]

with 1/5 of the energy in alpha particles and 4/5 in neutrons. The power density in alpha particles is

\[
p_{\alpha} = 0.154n^2T^2 \tag{B2}\]

\[
c_{DT} = \frac{p_{DT}(0)}{(nT)^2} = 0.154 \tag{B3}\]

\[
\kappa_c \approx 0.0272 \text{ eV} \text{ molecule}^{-1}. \tag{B4}\]

The derivation of the power density, megawatts per meter cubed, begins with Equation (1.4.2) of Wesson’s book Tokamaks [45]. The power density in alpha particles is \( p_{\alpha} = n^2 <\sigma v > \varepsilon_\alpha/4 \). The energy released in alpha particles per reaction is \( \varepsilon_\alpha = (3.5 \text{ MeV}) \times (1.60 \times 10^{-19} \text{ MJ/MeV}) = 5.6 \times 10^{-19} \text{ MJ} \). The velocity weighted cross section with Maxwellian ions is approximated within 10% accuracy for 10 keV < \( T < 20 \text{ keV} \) in Equation (1.5.4) as \( <\sigma v > = 1.1 \times 10^{-22}T^2 \) when the units of \( T \) are 10 keV. This calculation gives \( p_{\alpha} \), Equation (B2); a multiplication by five gives the full power density, \( p_{DT} \), Equation (B1).

The required energy confinement time to achieve ignition is

\[
\frac{3k_BnT}{\tau_E} = p_{DT} \quad \text{so} \quad \frac{nT\tau_E}{c_{DT}} \approx 3.12. \tag{B6}\]

The minimum of \( nT\tau_E \) is at \( T = 1.4 \), which means at 14 keV. What is precisely meant by \( nT \) is not clear since both \( n \) and \( T \) depend on radius. When central values are used in an analytic transport model, the constant 3.12 becomes 4.39, Equation (B3).

2. Transport model

The equilibrium between heat transport and fusion power in alpha particles is

\[
\frac{1}{r} \frac{d}{dr}(rQ) = p_{\alpha}(0)f, \text{ where} \tag{B7}\]

\[
f \equiv \left( \frac{nT}{n_0T_0} \right)^2, \tag{B8}\]

11
and $p_{DT}^0(0)$ is the central power density provided by the fusion-produced alpha particles. The heat flux is

$$Q(r) = -3k_B D \frac{dnT}{dr}, \quad (B9)$$

and $D$ is the diffusion coefficient for plasma pressure.

An integration of the transport equation across the plasma $0 < r < a$ gives

$$aQ(a) = p_{DT}^0(0) \int_0^a f r dr. \quad (B10)$$

### a. Analytic model

An analytic model is obtained by letting

$$D(r) = D_0 \sqrt{f}, \quad \text{so} \quad (B11)$$

$$Q(r) = \frac{3}{4} p_0 D_0 \frac{df}{dr}; \quad (B12)$$

$$p_0 = 2k_B n_0 T_0, \quad (B13)$$

where $D_0$ is a constant and $p_0$ is the central plasma pressure.

The solution to the equation for $f$ given by Equations (B7) and (B12) is

$$f(x) = J_0(x), \quad \text{with} \quad (B14)$$

$$x \equiv kr; \quad (B15)$$

$$f(0) = 1, \text{ and } f(ka) = 0. \quad (B16)$$

$J_0(x)$ is the zeroth order Bessel function of the first kind, which has its first zero, $J_0(\lambda_0) = 0$, at $\lambda_0 = 2.405...$, $dJ_0/dx = -J_1(x)$, and $d(xJ_1)/dx = xJ_0(x)$. The boundary condition $f(ka) = 0$ implies

$$ka = \lambda_0. \quad (B17)$$

The energy flux at the plasma edge is

$$Q(a) = \frac{3}{4} p_0 D_0 \frac{f_0}{a}; \quad (B18)$$

$$f_0 \equiv \lambda_0 J_1(\lambda_0), \quad \text{and} \quad (B19)$$

$$k^2 = \frac{p_{DT}^0(0)}{\frac{3}{4} p_0 D_0}, \quad \text{or} \quad (B20)$$

$$D_0 \frac{a^2}{\lambda_0^2} = \frac{p_{DT}^0(0)}{\frac{3}{4} p_0 \lambda_0^2}; \quad (B21)$$

$$= \frac{2c_{DT}}{3k_B \lambda_0 n_0 T_0}. \quad (B22)$$

Using Equation (B22), Equation (B22) gives the required confinement for the power from alpha heating to balance the thermal losses at the plasma edge.

The thermal energy in the plasma $W_{th}$ is the integral of $3p/2$ over the volume of the plasma:

$$W_{th} = \frac{(2\pi)^2 R}{2} \int_0^a \frac{3}{2} p_0 \sqrt{f r dr} \quad (B23)$$

$$= 3p_0 \pi^2 R a^2 \bar{m}_0 \text{ where} \quad (B24)$$

$$\bar{m}_0 = \frac{\int_0^{\lambda_0} \sqrt{J_0(x) x dx}}{\frac{1}{2} \lambda_0^2} \approx 0.608. \quad (B25)$$

When performed numerically $\int_0^{\lambda_0} \sqrt{J_0(x) x dx} \approx 1.76$.

The energy confinement time is

$$\tau_E = \frac{W_{th}}{(2\pi)^2 \pi R a Q(a)} \quad (B26)$$

$$= \frac{\bar{m}_0 a^2}{\bar{F}_0 D_0} \quad (B27)$$

$$\approx 0.488 \frac{a^2}{D_0}. \quad (B28)$$

The requirement for alpha heating to balance the thermal losses is

$$\frac{W_{th}}{\tau_E} = \frac{(2\pi)^2 R}{2} \int p_{DT}^0 r dr \text{ or} \quad (B29)$$

$$\tau_E = \frac{\int_0^{\lambda_0} \sqrt{f r dr}}{p_{DT}^0(0) \int_0^a p^2 r dr} \quad (B30)$$

$$= \frac{1}{n_0 T_0 c_{DT} \int_0^a p^2 r dr} \quad (B31)$$

$$= \frac{1}{n_0 T_0 c_{DT} \int_0^{\lambda_0} \sqrt{f r dr}} \quad \text{so} \quad (B32)$$

$$n_0 T_0 \tau_E \approx 3.12 \times 1.409 \approx 4.39. \quad (B33)$$

### b. Gyro-Bohm diffusion

The implications of transport on designs of toroidal magnetic fusion systems, stellarators and tokamaks, requires a normalizing transport model. Gyro-Bohm diffusion will be used for two reasons:

1. An empirical scaling law $\tau_{EISS04}^\star$, Equation (B64), describes a broad range of stellarator and tokamak experiments, Figure (3). This scaling law is accurately approximated by gyro-Bohm scaling, Equation (B41), with a dimensionless multiplying factor $D$. (2) Even non-turbulent transport models, Appendix B.5, can differ by a number of orders of magnitude from gyro-Bohm transport models, either larger or smaller, but gyro-Bohm transport with $D \approx 1$ gives optimal reactor designs. A heuristic
derivation of the gyro-Bohm diffusion coefficient is given in Appendix B2c.

The gyro-Bohm diffusion coefficient is
\[ D_B = \frac{\rho_i^2 C_s}{a} \] (B34)
where \( \rho_i \equiv C_s/\omega_{ci} \) is the ion gyroradius using the speed of sound \( C_s \equiv \sqrt{T/m_i} \), and \( a \) is the plasma minor radius. When \( D_B \) is evaluated at an ion mass of 2.5 times the proton mass,
\[ D_B = \frac{c_B T^{3/2}}{a B^2}; \] (B35)
\[ c_B \approx 162 \] (B36)
The speed of sound is \( C_s \equiv \sqrt{T/m_i} = 6.19 \times 10^5 \sqrt{T} \) and the ion gyroradius is \( \rho_i \equiv C_s/\omega_{ci} = 1.62 \times 10^{-2} \sqrt{T/B} \).

The analytic model of Appendix B2a is obtained when the density profile has the form \( n \propto \sqrt{T} \), then \( D_B/\sqrt{T} \) is constant.

To study the effect of enhanced or reduced transport, a dimensionless coefficient \( D \) is introduced so
\[ D_0 = D D_B(0), \] (B37)
\[ = D \frac{c_B T_0^{3/2}}{a B^2}, \] (B38)
where the constant \( c_B \) is given in Equation (B36).

c. Heuristic derivation of gyro-Bohm diffusion

The heuristic derivation of the gyro-Bohm diffusion coefficient starts with general expression for a radial diffusion coefficient, \( D \approx \Delta^2/\tau_{co} \), where \( \Delta \) is the radial scale of the microturbulence, and \( \tau_{co} \) is the correlation time of the flow \( v_r \) that gave the radial scale; \( \Delta \approx v_r \tau_{co} \). The radial velocity in electrostatic turbulence is \( v_r = \tilde{E}_\theta/B \). The radial motion produces a change in the electric potential \( \phi \approx \Delta(T/ea) \) where \( T \) is the plasma temperature and the minor radius \( a \) is the scale over which the temperature varies. The poloidal variation in the potential is \( \delta \phi \approx \delta \Delta(T/ea) \), where \( \delta \Delta \) is the poloidal variation of the radial scale. Consequently, \( \tilde{E}_\theta \approx (\delta \phi / \Delta \phi)(T/ea) \approx T/ea \); the radial scale varies by roughly the poloidal scale over the poloidal scale of the turbulence, \( \Delta \phi \). That is, \( D \approx (T/ea) \Delta \), where \( T/ea = \rho_i C_s/a \), the ion gyroradius times the speed of sound with both calculated using the temperature \( T \). Therefore, one can let
\[ D = \Delta \frac{\rho_i C_s}{a}, \] (B39)
where the approximations are absorbed into the radial scale size of the turbulence \( \Delta \).

In gyro-Bohm diffusion, \( \Delta = \rho_s \), which is a typical scale of fluctuations in ITG turbulence. In Bohm-like diffusion, \( \Delta \approx a \), which is as large as it can be.

The enhancement factor of gyro-Bohm diffusion has the interpretation
\[ D = \frac{\Delta}{\rho_s}, \] (B40)
but can also differ from unity because the plasma is not microturbulent; Appendix B3c, or turbulence is present in only part of the plasma.

d. Gyro-Bohm scaling of \( \tau_E \)

The scaling of the energy confinement time will be studied using Equation (B27) with \( D_0 \) replaced by the gyro-Bohm-scaled diffusion coefficient, Equation (B38),
\[ \tau_E = \frac{\bar{m}_0 B^2 a^3}{c_B D_0 F_0 D T_0^{3/2}}. \] (B41)

The convention is to replace the temperature dependence of \( \tau_E \) with a thermal power \( P_{th} \) dependence. Since \( P_{th} = W_{th}/\tau_E \) and the central pressure is \( P_0 = 2 k_B T_0 a^3 \), Equation (B24) implies
\[ \frac{1}{T_0} = 6\pi^2 k_B \bar{m}_0 \frac{Ra^2}{P_{th}}, \] and (B42)
\[ \tau_E = \frac{\bar{m}_0 \left( \frac{1}{T_0 c_B} \right)^{2/5} \left( 6 \pi^2 k_B \right)^{3/5}}{D^{2/5} P_{th}^{3/5}} \times a^{12/5} R^{3/5} n_{th}^{3/5} B^{4/5}. \] (B43)
\[ \approx 0.281 a^{12/5} R^{3/5} n_{th}^{3/5} B^{4/5}. \] (B44)

The parameter dependencies is this formula reproduce those of the \( \tau_E^{BS04} \) scaling law, Equation (B64) with surprising accuracy. The \( \tau_E^{BS04} \) scaling law represents both tokamak and stellarator experiments, Figure 3.

e. Gyro-Bohm scaling of non-ignited experiments

The magnetic field that is required to reach a central density \( n_0 \) and temperature \( T_0 \) can be calculated in terms of the thermal power, \( P_{th} = (2\pi)^3 Ra Q(a) \),
supplied, the enhancement over gyro-Bohm diffusion, \( D \), and the major \( R \) and minor radius, \( a \). Equation (B18) for the edge heat flux, \( Q(a) \), \( p_0 = 2k_B n_0 T_0 \) for the central pressure, Equation (B38) for the relation between \( D_0 \) and gyro-Bohm diffusion imply

\[
B = \sqrt{\frac{3}{2} k_B c_B F} \sqrt{\frac{D_{n_0} T_0^{5/2}}{a^2 Q(a)}} \tag{B45}
\]

\[
\approx 6.97 \sqrt{\frac{D_{n_0} T_0^{5/2}}{a^2 Q(a)}} \tag{B46}
\]

\[
= \sqrt{3\pi^2 k_B c_B F} \sqrt{D_{n_0} T_0^{5/2}} \frac{R/a}{P_{th}} \tag{B47}
\]

\[
\approx 31.0 \sqrt{D_{n_0} T_0^{5/2}} \frac{R/a}{P_{th}}. \tag{B48}
\]

Equivalently,

\[
D = \frac{1}{3\pi^2 k_B c_B F} \frac{B^2 P_{th}}{n_0 T_0^{5/2} R/a} \tag{B49}
\]

\[
= 1.043 \times 10^{-3} \frac{B^2 P_{th}}{n_0 T_0^{5/2} R/a}. \tag{B50}
\]

\[f. \text{ Gyro-Bohm scaling of ignited experiments}\]

When the plasma is undergoing a steady fusion burn, Equation (B22) gives an expression for the re-

\[
\left( \begin{array}{cccc}
\alpha & 0 & 0.5 & 0.6 & 0.7 & 0.8 \\
\lambda(\alpha) & 2.405 & 2.554 & 2.703 & 2.963 & 3.455 \\
\bar{m}(\alpha) & 0.6 & 0.7 & 0.8 & 0.8 & 0.9 \\
\mathcal{F}(\alpha) & 1.248 & 1.390 & 2.400 & 3.211 & 4.859
\end{array} \right)
\]

\textbf{TABLE I:} The argument \( \lambda \) of the Bessel functions at the plasma edge, \( r = a \), the ratio \( \bar{m} \) of the average to the central pressure, and the enhancement of the fusion power \( \mathcal{F} \) are given as a function of \( \alpha \), which is the fraction of the plasma radius in which diffusion is assumed to go to infinity.

\[
\text{required } D_0/a^2. \text{ The expression obtained for } D/a^2 \text{ from gyro-Bohm scaling is given by Equation (B38). Equating these two expressions provides an expression for the central density, } n_0 = n_b, \text{ with}
\]

\[
n_b = \frac{3k_B c_B \lambda_0^2 D \sqrt{T_0}}{2c DT_0} \tag{B51}
\]

\[
= 252.8 \lambda_0^2 \frac{D \sqrt{T_0}}{B^2 a^3}. \tag{B52}
\]

One less parameter is required to describe ignited than non-ignited experiments.

Equation (B18) for the edge heat flux, \( Q(a) \), \( p_0 = 2k_B n_0 T_0 \) for the central pressure, Equation (B38) for the relation between \( D_0 \) and gyro-Bohm diffusion, and \( n_0 = n_b \) using Equation (B52) imply that in an ignited plasma

\[
B = \left( \frac{9 c_B^2 k_B^2 \lambda_0^2 \mathcal{F}_0}{4 c_D T_0} \right)^{1/4} \left( \frac{D^2 T_0}{a^5 Q(a)} \right)^{1/4} \tag{B53}
\]

\[
= 9.96 \left( \lambda_0^2 \mathcal{F}_0 \right)^{1/4} \left( \frac{D^2 T_0^3}{a^5 Q(a)} \right)^{1/4}. \tag{B54}
\]

\textbf{3. Transport with a confining annulus}

A confining annulus means that the diffusion coefficient \( D(r) \) is extremely large in the central part of the plasma \( 0 < r < a \), so \( f = 1 \) there, but within the confining annulus \( a a < r < a \), the diffusion coefficient has the same form as in Appendix B2 \( D(r) = D_0 \sqrt{T} \). The solution for \( f \) in the confining annulus \( a a < r < a \) is

\[
f(x) = \frac{Y_0(\lambda) J_0(x) - J_0(\lambda) Y_0(x)}{J_0(\alpha \lambda) Y_0(\lambda) - J_0(\lambda) Y_0(\alpha \lambda)}, \tag{B55}
\]

where \( k^2 = 4 p_{DT}(0)/(3 p_0 D_0) \) as before, Equation (B20). The boundary conditions are \( f(\alpha \lambda) = 1 \) and
\[
\begin{pmatrix}
\frac{\tau}{a} & 0 & \frac{d \ln p}{d \ln r} & \frac{\rho}{\rho_0} & \frac{d \ln \rho}{d \ln r} & \frac{\rho}{\rho_0} & \frac{d \ln \rho}{d \ln r} \\
0.5 & 0.818 & 0.448 & 1 & 0.408 & 1 & 0 \\
0.6 & 0.737 & 0.729 & 0.907 & 0.695 & 1 & 0 \\
0.7 & 0.638 & 1.191 & 0.789 & 1.163 & 1 & 1.075 \\
0.8 & 0.518 & 2.084 & 0.642 & 2.063 & 0.822 & 2.000 \\
0.85 & 0.445 & 2.952 & 0.553 & 2.936 & 0.7105 & 2.887 \\
0.9 & 0.361 & 4.66 & 0.448 & 4.647 & 0.577 & 4.613 \\
\end{pmatrix}
\]

**TABLE II:** The radial profiles of the pressure and the logarithmic derivative of the pressure with respect to radius are given for three values \(\alpha\), which is the fraction of the plasma radius in which diffusion is assumed to go to infinity. As \(\alpha\) becomes larger, the stability measure \(d \ln p/d \ln r\) becomes smaller at a given radius, but the pressure at which \(d \ln p/d \ln r\) reaches a certain value becomes larger.

\(f(\lambda) = 0\). \(J_0(x)\) and \(K_0(x)\) are the Bessel functions of the first and second kind. Both obey the relations \(d J_0/d x = -J_1(x)\) and \(d(x J_1)/d x = x J_0(x)\).

The function \(\lambda(\alpha)\) is given implicitly by Equation (B1), which is obtained by equating the total exiting heat flux \(2 \pi a Q(a)\) per unit length of the plasma to the toroidal direction with the total alpha-heating power per unit length.

\[
\int_0^\lambda f x dx = \frac{(\alpha \lambda)^2}{2} + F(\lambda) - \alpha \lambda \frac{J_0(\alpha \lambda)J_1(\alpha \lambda) - J_0(\lambda)Y_1(\alpha \lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)}; \quad (B56)
\]

\[F(\alpha) = \lambda \frac{J_1(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_1(\alpha \lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)}; \quad (B57)\]

\[
ka \left( \frac{df}{dx} \right)_\lambda = -F(\alpha). \quad (B58)
\]

The implication is that Equation (B18) for \(Q(a)\) holds when \(F_0\) is replaced by \(F(\alpha)\). Indeed, \(F\) can be defined by Equation (B18).

Equation (B10) for energy balance is satisfied when

\[
0 = \frac{\alpha \lambda}{2} + \frac{Y_0(\alpha \lambda)J_1(\alpha \lambda) - J_0(\lambda)Y_1(\alpha \lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)} \lambda a Q(a) = \rho_{DT}^2 \frac{\lambda}{k^2} F(\alpha). \quad (B59)
\]

Equation (B59) implicitly gives the function \(\lambda(\alpha)\), Table II. In the absence of a region of rapid transport, \(\alpha = 0\), the solution vanishes, \(f(\lambda) = 0\) at \(\lambda = \lambda_0 \approx 2.405\) and \(F = J_0(\lambda_0)\).

The equation for energy balance \(\int_0^a \rho_{DT}^2 r dr = a Q(a)\) can be used to define \(\lambda\) for any pressure profile that satisfies the transport equation as

\[
\lambda^2 = \frac{\rho_{DT}^2(0)a^2 F}{\int_0^a \rho_{DT}^2(r)r dr}; \quad (B61)
\]

Similarly \(\bar{m}\) can be defined as \(\bar{m} = \frac{\bar{m}}{p(r)dr/p_0 a^2}\). These definitions give the equations derived in Appendix B2 general validity.

Plasmas are generally unstable to microturbulence when the logarithmic gradient of the pressure becomes large compared to unity;

\[
\frac{d \ln p}{dr} = \frac{\rho}{2} \frac{df}{dx} = \frac{\rho}{2} \frac{\rho}{2} \frac{d \ln r}{dx} \quad \frac{-J_0(\lambda)Y_1(\lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)} - \frac{J_0(\alpha \lambda)Y_1(\alpha \lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)}; \quad (B62)
\]

\[
\frac{d \ln p}{dr} = \frac{\rho}{2} \frac{df}{dx} = \frac{\rho}{2} \frac{\rho}{2} \frac{d \ln r}{dx} \quad \frac{-J_0(\lambda)Y_1(\lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)} - \frac{J_0(\alpha \lambda)Y_1(\alpha \lambda)}{J_0(\alpha \lambda)Y_0(\lambda) - J_0(\lambda)Y_0(\alpha \lambda)}; \quad (B63)
\]

The pressure profile and the profile of the logarithmic derivative of the pressure are given in Table II.

### 4. Comparison with experiments

The observed global energy confinement in stellarator experiments is summarized by the scaling

\[\tau_E^{ISS04} = 0.134 \frac{0.28^{0.64}}{P_0^{0.61}} \rho^{-0.54} B^{0.84} r^{0.41}; \quad (B64)\]

where the energy confinement time is in seconds, the minor \(a\) and the major radius \(R\) are in meters, the volume averaged magnetic field \(B\) is in Tesla, the volume-averaged electron density \(\bar{n}\) is in \(10^{20}/m^3\), the effective heating power \(P\) is in mega-Watts, and the rotational transform \(\tau_{3/2}\) is at a radius \(r = 2a/3\). The minor radius \(a\) is defined so the plasma volume is \((2\pi R^2a^3)\).

The \(\tau_E^{ISS04}\) scaling law represents both tokamak and stellarator experiments, Figure 3, though tokamak H-mode experiments have up to a factor of two better confinement than predicted. Paradoxically, the radial dependence of the transport seen in W7-X, as reported in the article from which this figure was taken [17], does not agree with that expected for gyro-Bohm transport. Nevertheless, the overall dependencies of the \(\tau_E^{ISS04}\) scaling law are given by gyro-Bohm transport, Equation (B44). The coefficient in the stellarator scaling is a factor of 2.99 times smaller than in gyro-Bohm scaling, which can be counterbalanced by \(D = 6.3\). The rotational-transform dependence of \(\tau_E^{ISS04}\) can be interpreted as \(D \propto 1/\tau\).

The detached divertor experiments in the Large Helical Device (LHD) that were reported in 2018 [18]
had \( a = 0.55, R = 3.90, n_0 = 0.7, B = 3, P_{th} = 9, \) and a stored plasma energy \( W_{th} = 0.35. \) These results were said to be agreement with Equation (B64) for stellarator scaling. The central temperature is related to the thermal energy content in the analytic model by

\[
T_0 = \frac{W_{th}}{3\pi^2 n_0 E_B n_0 Ra^2} \quad \text{(B65)}
\]

\[
\approx 0.147. \quad \text{(B66)}
\]

Consistency with Equation (B48) is obtained for \( D = 2.1. \)

If \( D \) were 2.1 for stellarators, but the energy confinement time were factor of two longer, as in the case in H-mode tokamaks in Figure 3, then \( D \) would be 0.37 for H-mode tokamaks. Smaller values of \( D \) have been seen in tokamak and stellarator experiments.

A study of long-pulse DIII-D results published in 2018 [49] had \( T_0 = (T_e + T_i)/2 = 0.45 \) and \( n_0 = 0.5, B = 1.6, R = 1.7, a_i = 0.6, \) and \( P_{th} = 15.6. \) The elongation was \( \kappa_e = 2, \) which makes the stellarator definition \( a = \sqrt{\kappa_e a_i} = 0.849. \) A fit gives \( D = 0.31. \)

Early results from W7-X [32,33] demonstrate that excellent confinement can be obtained, \( D = 0.05, \) though this confinement rapidly degrades, possibly because continual pellet injection is not yet available. The central plasma has \( T_i = T_e = 3.5 \) keV and \( n_0 = 0.8 \times 10^{20}/\text{m}^3, B = 2.5 \) T, \( R = 5.5 \) m, \( a = 0.5 \) m, and \( P_{th} = 5 \) MW. W7-X was able to maintain plasma parameters for ten seconds [50] with \( T_i = T_e = 1.9 \) keV and \( n_0 = 1.6 \times 10^{20}/\text{m}^3, B = 2.5 \) T, \( R = 5.5 \) m, \( a = 0.5 \) m, and \( P_{th} = 5.9 \) MW. These results give \( D = 0.13, \) a value that yields attractive reactor designs.

Burning plasma experiments in ITER seem to require a value of \( D \) consistent with those seen in DIII-D. For example, the burning-plasma scenario outlined in Table 1 of [51] for ITER had \( P_T = 500, \) \( a_t = 2, R = 6.2, \kappa_e = 1.8, B = 5.3, \) \( < n > = 1.1, \) and \( < T > = 0.89. \) Assuming broad profiles so \( n_0 = 1.18 \) and \( T_0 = 1.1 \) gives \( D = 0.42. \) Similarly, the European Union design for a pulsed demonstration (DEMO) tokamak reactor [2] has \( T_0 = 25 \) keV and \( n_0 = 1.5 \times 10^{20}/\text{m}^3, B = 5.9 \) T, \( R = 9 \) m, \( a_t = 2.9 \) m, and \( P_T = 2,014 \) MW, assuming central values are twice their volume averages. The stellarator equivalent minor radius is \( a = \sqrt{\kappa_e a_t} = 3.67. \) m. This requires \( D = 0.11. \) The power loading is \( p_w = P_T/(2\pi^2 \kappa_e Ra) = 1.2 \) MW/m², where \( \kappa_e = 1.6 \) is the elongation.

When the confinement factor \( D, \) the magnetic field strength \( B, \) and the wall loading \( p_w \) are held constant, the factor that determines the total power output \( P_T \) scales as \( a^2 \propto T_0^{3/5}. \) The higher plasma temperature required even in a pulsed tokamak reactor offsets its lower aspect ratio in comparison to a steady state stellarator reactor. For example, using stellarator definitions, the European Union DEMO has an aspect ratio of 2.45 but a central temperature is 2.5 times greater than would probably be chosen for a stellarator reactor, \( 2.45 \times (2.5)^{6/5} = 7.364, \) which is a reasonable aspect ratio for a stellarator reactor.

5. Non-turbulent transport

The characteristic diffusion coefficient for neoclassical transport in which the particle drift trajectories make small excursions from the magnetic surfaces is

\[
D_{nc} = \alpha_{nc} \rho_i^2 \nu_i, \quad \text{and} \quad \text{(B67)}
\]

\[
D_{nc} = \frac{D_{nc}}{D_{gb}} \quad \text{(B68)}
\]

\[
\approx \alpha_{nc} \frac{\lambda_i}{a}, \quad \text{(B69)}
\]

where \( \alpha_{nc} \) is a dimensionless coefficient, which can be of order \( 10^2 \) and \( \nu_i \) is the ion collision frequency.

The mean free path \( \lambda_i \equiv C_s/\nu_i \approx 10.05 \times 10^3 T^2/n, \) so \( \lambda_i/a \sim 5 \times 10^3 \) in a fusion reactor.

Non-turbulent transport scales differently when the drift of some of the particles away from the magnetic surfaces is limited only by collisions. The characteristic transport coefficient for this type of transport is

\[
D_{1/\nu} = \alpha_{1/\nu} \left( \frac{\rho_i}{a} \frac{C_s}{\nu_i} \right)^2 \nu_i, \quad \text{and} \quad \text{(B70)}
\]

\[
D_{1/\nu} = \frac{D_{1/\nu}}{D_{gb}} \quad \text{(B71)}
\]

\[
= \alpha_{1/\nu} \frac{a}{\lambda_i}. \quad \text{(B72)}
\]

That is \( D_{1/\nu} \) can be orders of magnitude greater than \( D_{gb}. \)

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