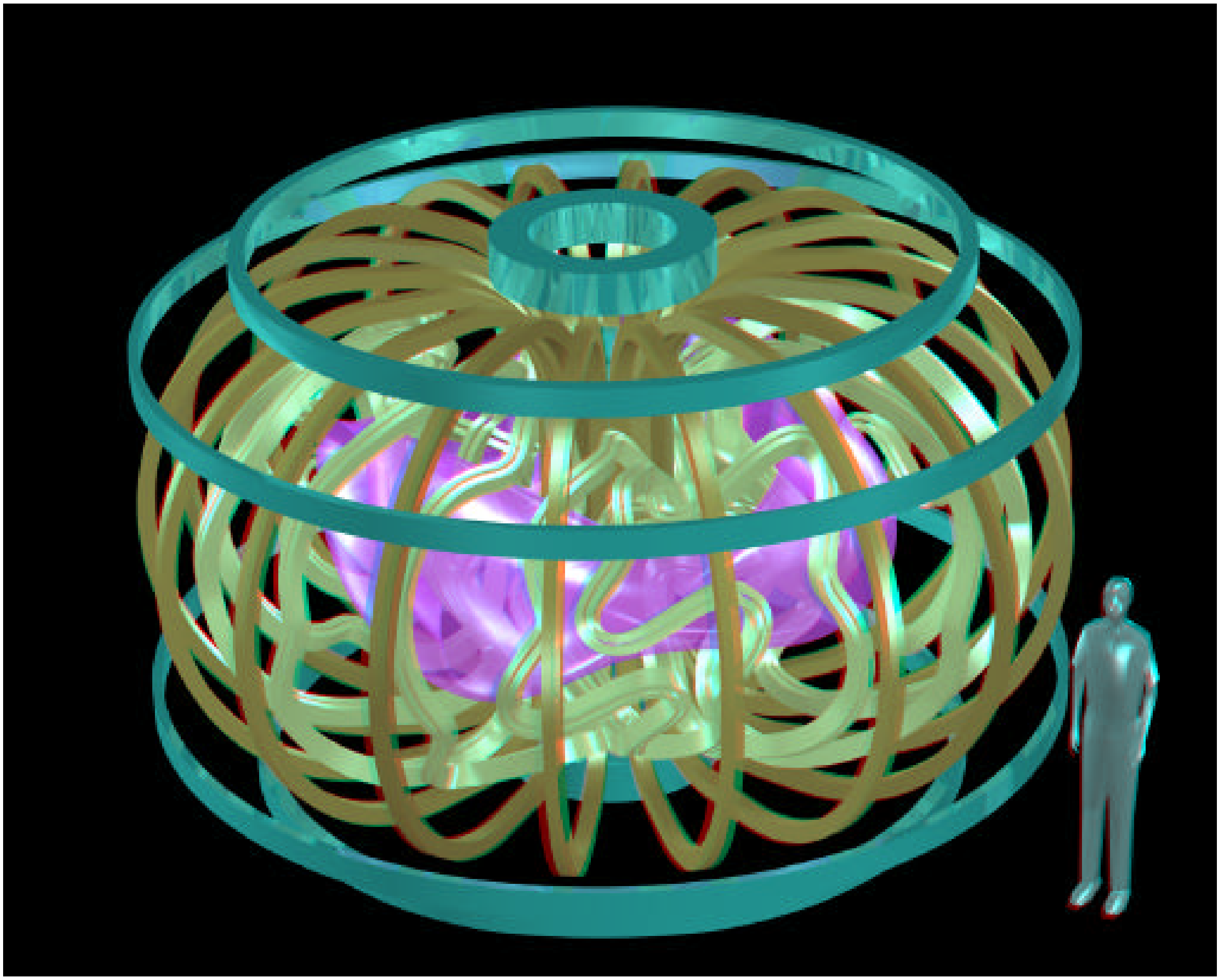
Workshop on Future Directions in Theory of 3D Magnetic Confinement Systems
Summary Report



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Oak Ridge, Tennessee

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Executive Summary

In recent years, there have been major advances in stellarator theory and in the ability to make detailed projections of stellarator performance based on this theory. In the U.S., these advances have been driven to a large extent by the interest in exploitation of symmetry or quasi-symmetry to improve stellarator performance and by the optimization and design studies to realize such improvements in compact experimental devices. The intense push on compact stellarator design that has preoccupied much of the U.S. stellarator theory effort for the past few years is now shifting to an engineering focus. Thus, it was apparent that now would be an appropriate time to assess the future needs and opportunities for stellarator theory and to examine its relationship to theory for other confinement concepts. Therefore, a workshop on Future Directions in Theory of 3-D Magnetic Confinement Systems was held in Oak Ridge on January 7-9, 2002. Broad objectives of the workshop were to:

- Have a scientific exchange on issues and opportunities for theory of 3-D confinement systems and to identify crucial issues not presently being adequately addressed.
- Provide a forum for the experimental programs to communicate needs and priorities to the theory community.
- Draw in researchers not traditionally associated with stellarators, but with interests and expertise that could contribute to, and benefit from, such a discussion—for example, the presence of 3-D magnetic structures in tokamaks.
- Stimulate the formation of collaborations and possible new initiatives.

The emphasis was on the discussion of scientific issues, opportunities for advances, and program priorities, not on presentation of traditional scientific papers. This document was prepared as a written summary of the workshop.

The workshop was organized around seven topical sessions, each with two session organizers: Input from experimental programs (D. Anderson), MHD (A. Reiman and C. Hegna), Confinement (A. Boozer and H. Mynick), Modeling and diagnostics (W. Houlberg and J. Callen), Wave propagation and RF applications (D. Batchelor and H. Weitzner), Optimization (A. Ware and N. Pomphrey), and Edge physics modeling (J. Hogan and T. Rognlien). The workshop attracted a wide cross section of U.S. fusion researchers totaling 38 attendees from 4 laboratories and 7 universities.

The body of this report contains summaries for each of the topical areas. The reader will immediately see the vitality of the field. In each area, there are interesting scientific questions whose importance reaches far beyond just the needs of stellarators but touches essentially all magnetic fusion concepts and extends into non-fusion fields. The reader will also perceive the richness of ideas presented and opportunities for scientific advance. Although the document represents a very compressed summary of a very active field, we have endeavored to extract a short list of key issues of two sorts—those of some urgency where progress is needed in the near term, and important areas for which development will be necessary over an extended period.

MHD

MHD equilibrium underpins essentially all theoretical activities for magnetic confinement—stability, transport, optimization and design, and interpretation of experiments. An important near term task is to better understand the role and the physics of the singular parallel currents that appear in equilibrium models assuming flux surfaces and to improve such models to better accommodate and provide physically meaningful interpretation of these currents. It is also important to computationally accelerate the models that do include island formation in the equilibrium. It will be important to make comparisons of the three types of codes presently available (codes assuming flux surfaces such as VMEC, equilibrium codes allowing for islands such as PIES, and non-linear initial-value codes such as M3D) among themselves and with experiment. On the longer time scale, it will be essential to apply the equilibrium models to the calculation of plasma evolution on resistive and transport time scales, including the non-ideal physics effects (resistivity, bootstrap current, flow…) which affect island formation, and to incorporate these non-ideal effects into the 3-D stability codes. Many of the present codes do not include any form of dissipation. In addition, computationally efficient methods to estimate low-n stability are needed.

Confinement

There are a number of related neoclassical effects that play critical roles in the newer classes of stellarators and which are difficult to calculate for the intermediate to low collisionalities of future interest. These include: physics of ambipolar electric potential and its variation within a flux surface, the accurate inclusion of momentum conservation, the determination of the in-surface flows that that can affect turbulence and island formation, and calculations of the bootstrap current and the confinement properties of impurity species. Methods for improved calculation of these effects are under development by Monte Carlo techniques or direct solution of the drift kinetic equation. Accelerated progress is needed in this development. In the intermediate term it is important to extend fluid based models (such as NCLASS) of plasma rotation, flow, and transport from 2-D, where they have been key to understanding advanced tokamak physics, to 3-D. Such fluid models will depend on the availability of the previously mentioned advanced kinetic codes to benchmark the analytic transport coefficients. In the longer term, it is essential to understand microstability and turbulent transport to the level that optimization with respect to anomalous transport can be included in the stellarator design process.

Modeling and Diagnostics

There is presently no actively used predictive or interpretive transport code in the U.S. for 3-D systems incorporating 3-D MHD equilibria or other physics associated with non-axisymmetry. The stellarator community needs to identify how to fill this void. Because of the magnitude of the work involved, it may be most effectively accomplished through collaborative development. However, many of the computational tools and models used in recently developed optimization codes already provide the appropriate level of depth and computational efficiency to be incorporated in time-dependent simulations. The capability for equilibrium reconstruction from experimental data, such as the EFIT code, has proved to be an essential tool for tokamak experimentation. However, a similar capability for non-axisymmetric configurations is just now beginning to be developed. As beta rises in stellarators and for compact (low aspect ratio) hybrid configurations with net plasma current, such as the NCSX, QPS and CTH experiments, this capability becomes increasingly important for accurate equilibrium analysis. There is some urgency in completing this development because it will be needed for design of the NCSX magnetic diagnostics.

Wave Propagation and RF Applications

Although most of the complications in the theory of wave propagation and absorption are already present in 2-D, the theory is less developed in every area for 3-D systems, the codes are computationally much more demanding, and there is much less experience in verifying the models with experiments. There are a number of open issues of a rather basic nature in 3-D plasmas largely having to do with approximations to the conductivity operator which are necessary to make computational progress. One of the near term needs is to understand the validity of these approximations for relevant 3-D systems and to find computationally feasible ways to improve on them where necessary. Another urgent near-term need is to find ways to accelerate the wave solver codes, by both advanced computer techniques and reformulation of the solution methods, so that comparisons can proceed with experiment and other codes. In the long time-scale, one of the most important and difficult problems is to determine the plasma evolution from the Fokker-Planck equation. It will be important to include not just collisional equilibration within a flux surface but also radial particle transport and the presence of rapid loss zones in phase space. Both Monte Carlo methods and direct solution of the drift kinetic equation need to be explored.

Optimization

Furthering the significant recent advances in stellarator optimization and design requires enhancing the speed of the existing codes and improving our understanding of the relevant physics issues. A near term goal in stellarator optimization is to complete the design of the NCSX and QPS stellarator experiments, including the implementation of the combined coil and configuration optimization processes. A longer-term goal is to include a good measure of surface quality in the optimization process which would require enhancing the numerical efficiency of codes that directly test surface quality. In addition, the optimization of a burning plasma experiment and coming up with an attractive compact stellarator reactor configuration are important mid-term goals.

Edge Physics Modeling

Because of the existence of open field lines in the scrape-off layer region, edge problems are inherently multi-dimensional, even in nominally axisymmetric systems such as the tokamak. Thus, any significant improvement in our capability to treat 3-D problems will have great benefit across the whole range of edge physics problems in fusion. The dramatic changes in confinement properties and scaling recently observed in the W7-AS stellarator, which were due to divertor optimization, point to an urgent need in the near term both for self-consistent 3-D neutrals calculations that adopt fixed plasma background properties, and for vigorous implementation of 3-D self-consistent coupled plasma-neutrals codes, such as BORIS. Success in dealing with both of these problems will require more accurate MHD equilibrium resolution in the edge region. In the longer term, the development of coupled plasma turbulence and edge plasma models will be needed to create credible and extrapolable models for stellarator behavior. The understanding of pellet ablation dynamics is an inherently time-dependent 3-D problem, and development of an extrapolable model for this process was also identified as an important and feasible long-range goal.

Throughout the report, the reader will observe that most of the important issues are computationally very challenging. Much of the recent progress in the theory of 3-D configurations results from the availability of powerful new computers. Still, there are great opportunities to increase accuracy and level of physics detail achievable by application of advanced computing techniques. Perhaps now is the time to mount a major initiative of extending and integrating 3-D models to be ready for the new stellarators that will be coming on line and to address the outstanding 3-D issues in other confinement concepts.

<u>Workshop Organizers</u>

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1. MHD Equilibrium and Stability

3D Equilibrium and Magnetic Island Physics

Accurate calculations of MHD equilibria are required for interpreting experimental data and are necessary for theoretical assessments of the stability and transport of any toroidal confinement device. Unlike the equilibrium theory of axisymmetric magnetic confinement systems, which is essentially complete, a number of crucial questions remain in the theory of three-dimensional MHD. The calculation of three-dimensional equilibria is complicated by the fact that if nested surfaces are assumed, 3-D MHD equilibria are singular at rational surfaces. The general solutions to the MHD force balance relations describe parallel plasma currents that contain resonant Pfirsch-Schlüter currents and delta-function current sheets at rational values of the rotational transform. These singular currents are resolved by allowing magnetic islands to form. An understanding of the physical processes involved in the behavior near rational surfaces is critical for accurate predictions of 3-D MHD equilibria and impacts our understanding of stability, transport, and the ability to reconstruct experimental equilibria. Specifically, we need to understand: the presence of singular currents in 3-D equilibria when flux surfaces are assumed, the resolution of current singularities via magnetic island formation, and the relevant physics involved in describing the magnetic island structure including effects outside of the magnetostatic equilibrium assumption such as rotation, neoclassical, two-fluid, and kinetic effects. Progress toward understanding 3-D equilibria can be made by improving computational capabilities, including physics beyond ideal MHD in the models, and comparing equilibrium calculations against other theories and experiment. In addition to stellarator applications, 3-D equilibria codes that can handle islands could also provide useful insights for understanding the physics of tokamaks with slowly growing 3-D instabilities. Furthermore, the problem of current singularity formation in 3-D space and astrophysical plasmas is a related problem of physical interest, with significant implications for the physics of solar and stellar coronal heating.

A number of computation tools to describe 3-D equilibria have been developed that are extensively used throughout the stellarator community. Several of the pioneering stellarator equilibrium codes were developed in the United States. These include BETA [1], its descendent NSTAB [2], VMEC [3], and PIES [4]. These codes can be distinguished by their treatment of the magnetic topology of the equilibria. The codes BETA, NSTAB and VMEC use a flux surface representation for the magnetic field that defines a set of nested flux surfaces. Equilibrium codes that allow for the presence of islands include PIES, a code developed in the U.S., and HINT, a code developed in Japan. Island physics in stellarator geometry can be addressed with the initial value resistive code M3D (described in the stability section) [5]. Additionally, analytic theory techniques can be used for detailed investigation of the island physics [6].

Comparison calculations between these different techniques would be useful. In particular, a range of different equilibria might be investigated ranging from vacuum cases both with and without magnetic islands (which could be checked against Biot-Savart solutions) up to finite-beta equilibria cases with and without net current. It would be of interest to compare the PIES, HINT and M3D predictions for island widths in a variety of cases. Equilibrium codes, such as VMEC, that assume the existence of the magnetic surfaces will necessarily be inaccurate near rational unless the Hamada condition is exactly satisfied (which usually does not occur with the numerical accuracy presently achievable in these codes). This manifests itself in calculations of the parallel component of the current profile, as deduced from Ampere's law, that are inconsistent with the singular currents found from solutions of the quasineutrality equation. The resolution of the singular current issue is an important challenge that will require strong analytical and computational efforts. Insights into the physical significance of the current singularities may be obtained by doing comparison calculations with codes that resolve the island physics, such as PIES. The NSTAB code has a diagnostic that has been developed to indicate islands (although no island topology is allowed in the calculation) by examining the helical distortion of the magnetic surfaces. A comparison of these distortions with predictions of actual islands would be interesting. A possible outcome would be a tool for diagnosing the presence of islands in optimization studies.

Analytic theory of magnetic island formation in three-dimensional equilibria using resistive and neoclassical magnetohydrodynamic models is highly developed and yields detailed predictions of how various physical processes influence the island current profile and subsequent effect on the magnetic island widths [6]. These studies have led to important insights as to how to improve the quality of magnetic surfaces in stellarators. A comparison between analytic theory and computational predictions of magnetic islands would illuminate the critical physics and give confidence that reliable predictions of particular configurations can be made.

All of the 3-D MHD equilibrium codes have seen major improvement in recent years in their speed and in their capability for handling increased spatial resolution. While the assumption of nested surfaces is never exactly satisfied, it is often a useful approximation when the configuration of interest has small regions of magnetic islands and stochasticity. One possible avenue for further investigation would be look at methods for calculating 3-D equilibria without using inverse magnetic coordinate techniques.

The PIES code is presently the only code capable of calculating free-boundary equilibria with islands (i.e. equilibria in which the coils are specified and boundary conditions are applied at infinity). Speed remains an issue for the PIES code. A free-boundary run can take two or three days to run on a DEC Alpha workstation. It is believed that there are straightforward opportunities for speeding the subroutines that are bottlenecks in the PIES code, and there are also ideas for speeding the code by improving the algorithm. Parallelizing the code will also be desirable.

The inclusion of additional physics in the PIES code would also be useful. In particular, using a time-dependent Ohm's law would introduce the slow resistive time scale associated with the nonlinear evolution of magnetic islands. These calculations can be compared with resistive MHD calculations from the M3D code. The incorporation of plasma physics outside of the resistive MHD model (such as flows, neoclassical effects, two-fluid effects, plasma transport, kinetic ion effects) in both of these codes would be useful. Additionally, a free boundary capability for the M3D code and the ability to examine more realistic plasma parameters would be welcomed.

An assessment of non-ideal MHD effects on islands in 3-D equilibria needs to be made. To a large extent, the important physical processes that affect magnetic islands in stellarators also influence the nonlinear dynamics of tearing mode induced magnetic islands in tokamaks. The connections between resistive MHD tearing mode theory and island formation in 3-D equilibria are highly developed. However, important physical effects outside the resistive MHD model also impact nonlinear tokamak tearing modes such as neoclassical and two-fluid effects. A firmer connection to these calculations for stellarator applications is called for. One of the important issues in magnetic confinement physics is the role of plasma flows in preventing the penetration of field error perturbations. With the increased emphasis on quasi-symmetric stellarators (where sustained plasma flows should be allowed), models that include the interaction of self-consistently generated plasma flows, radial electric field formation, magnetic field errors, and magnetic island formation needs to be developed.

VMEC will continue to be applied extensively to the interpretation of stellarator experimental data. It is expected that in the future VMEC will also be incorporated into time-dependent modeling, and into new diagnostic software. The calculation of PIES equilibria with islands will also be of interest for the interpretation of experimental data, particularly for the high beta compact stellarator experiment, because of the flux surface issues that arise in that context. In the near term it will be of interest to apply the PIES code to edge modeling, where it will be of importance for handling the transition from nested to open field lines for a three-dimensional magnetic field. Three-dimensional equilibrium codes that handle magnetic islands will also potentially provide a useful tool for the interpretation of tokamak experimental data. In the presence of nonlinear tearing modes and resistive wall modes, the tokamak plasma evolves through a sequence of three-dimensional equilibria on long resistive timescales. Accurate modeling of these processes remains an open question in the tokamak community. As such, 3-D equilibrium codes with islands can play important roles outside of stellarator applications.

There is a need for a new code capability to do reconstruction of MHD equilibria from experimental data. This code will interleave regression on experimental data self-consistently with convergence to the equilibrium condition, $\nabla p = j \times B$. Such a code is a necessary precursor to accurate comparisons of experiment with stability theory. Development is beginning this year on the code V3FIT. This code will also be used to optimize the location of magnetic diagnostics external to the plasma, which will make it an indispensable tool for real-time plasma shape control (see the section on Modeling and Diagnostics).

Linear and Nonlinear Ideal MHD Instabilities, and Kinetic Effects

One of the principal goals of the stellarator program is to understand the instability processes that limit plasma operation. Low mode number MHD instabilities and ballooning instabilities are believed to lead to disruptions in tokamaks and to limit their operating space. (Whether Mercier modes pose a danger in tokamaks remains controversial.) The situation in stellarators is less clear. Prior to LHD, access to potentially unstable regimes in stellarator experiments has been limited by the fact that stellarators have operated with little or no current and by the relatively low beta's (< 2%) that have prevailed experimentally. LHD now appears to be operating successfully in some regimes that are predicted to be unstable. This observation is not well understood and highlights a need for improving our comprehension of the stability properties of 3-D systems. Important stellarator related stability tasks include: an assessment of 3-D geometry stabilization mechanisms on ideal MHD instabilities, examinations of the viability of local criterion in predicting beta-limits, a formulation for calculating asymptotic matching data in resistive MHD instability theory, and an elucidation of the role magnetic islands play on ideal MHD. The role of these calculations is to yield guidance for interpreting and predicting experimental operation and to improve our ability to design stellarators.

An important aspect of stability calculations in toroidal confinement devices is the role of shaping. One of the major achievements of tokamak physics is a deep understanding of how axisymmetric shaping impacts operational limits. This understanding has greatly benefited the designs of tokamaks and the interpretation of experiments. In three-dimensional configurations, an additional dimension of shaping parameters is available to affect key geometric features that influence MHD stability properties. An approach for assessing the role of 3-D shaping effects on ideal MHD stability properties is called for.

In theoretical assessments of particular stellarator configurations, criteria for the onset of ideal MHD ballooning and Mercier modes are often used to predict beta limits. However, there is little experimental evidence to support the use of local stability criteria as a method to accurately predict the operational limits. One of the open questions concerns the viability of using local criteria in 3-D systems. Ballooning theory is predicated on a WKB-like formalism that is rigorously valid in symmetric systems. Whether this approach is valid in three-dimensional systems is still not certain. Questions on the application of ballooning formalism will also impact micro-instability studies that use this approach. The most unstable eigenvalues of the local ideal MHD stability theory typically correspond to eigenfunctions that are highly localized to distinct regions of each surface since the local eigenvalue is field-line dependent. Additional physics such as ω^* and Armor radius effects will affect the stability properties of these localized modes. A quantitative measure of this needs to be developed. These studies might indicate that the use of local criterion is overly pessimistic.

The theory of resistive MHD instabilities in 3-D systems needs further development. With the inclusion of substantial equilibrium currents in stellarator configurations, tearing mode instabilities may play an important role. Calculations of tearing mode stability use a boundary layer theory where an inner singular layer solution describes a thin region encompassing the rational surface where magnetic reconnection and magnetic island physics is accounted for. Solutions in the inner region are matched asymptotically to an outer region that is governed by the marginal ideal MHD. Calculating the outer region asymptotic matching data for 3-D systems is more complicated than calculating it in 2-D systems since helical harmonics with different toroidal and poloidal mode number are coupled. A method for calculating the asymptotic matching data for 3-D resistive MHD instabilities should be developed.

While tools for studying ideal MHD stability of 3-D equilibria with good flux surfaces are fairly well developed, there is a need to develop tools to describe the stability properties of 3-D equilibria with magnetic islands. In performing Mercier or global stability calculations using an equilibrium produced by a code that assumes good flux surfaces, it is standard practice to recalculate the current from force-balance. An issue that needs to be addressed is the reconciliation of the (non self-consistent) current calculated in this way with the current calculated by the equilibrium code. Singular currents can have a large effect on local stability. An open question is what effect a resolved island structure would have on the ideal MHD stability properties.

Low mode number linear ideal instabilities are presently studied using the European codes Terpsichore [7] and CAS3D [8]. The NSTAB code, whose equilibrium capabilities have been described in the previous section, also provides a capability for doing ideal stability calculations that includes some nonlinear information. The M3D code, developed in the U.S., provides a capability for doing fully nonlinear, time-dependent MHD stability calculations. It will be of interest to benchmark these codes against each other and against experiment. Unlike stability calculations of axisymmetric equilibria where perturbations with different toroidal mode numbers are decoupled, linear eigenmode structures of 3-D configurations contain a large number of Fourier harmonics. Due to this fact, calculations of linear MHD mode structures are computationally challenging. An improved formulation of the stability properties of 3-D systems would be welcome.

M3D is a 3-D time dependent extended MHD code intended for ideal and resistive MHD that can also include two-fluid effects, neoclassical MHD, and a hot particle population [5]. The code can run on massively parallel computers (SP2 and T3E), shared memory parallel machines (Sun and Origin2000), or single processor computers. M3D should provide insight on the role of nonlinear, resistive MHD, two-fluid, and neoclassical MHD effects in stellarator operation. Progress on the inclusion of a particle simulation capability will be welcomed.

Kinetic effects may also play a role in stabilizing MHD instabilities. The fully kinetic HINST code solves a system of kinetic equations for medium to high-n modes including kinetic ballooning, drift and Alfvén modes. It presently works only in axisymmetric geometry, but is being converted to stellarator geometry. HINST is a linear code and utilizes a Fourier ballooning representation for perturbed quantities. Such instabilities as kinetic ballooning modes may be more stable in compact stellarators than in tokamaks due to the larger number of trapped electrons.

Energetic Particle Driven Modes

Energetic particle destabilized MHD instabilities have been extensively observed on the W7-AS, CHS, and LHD stellarators. Interest in these modes is motivated by their: (a) ability to rapidly eject energetic particles, leading to lowered heating efficiencies and wall damage, (b) potential diagnostic use as a measure of the MHD spectrum and rotational transform profile, (c) complex nonlinear physics, and (d) possible role in channeling fast ion energy directly into heating core ions.

Energetic particle driven Alfvén modes are in many ways similar to those in tokamaks. However, 3-D effects introduce significant new physics such as toroidal mode coupling, non-conservation of toroidal angular momentum and new particle orbits. A self-consistent analysis requires a full treatment of 3-D geometry for both linear stability and nonlinear dynamics. Low aspect ratio stellarator hybrids will include significantly higher levels of plasma current and mode coupling than existing stellarator experiments. These features can fundamentally change the form of such instabilities and will require the development of new theoretical tools for their analysis. The computational challenges posed by energetic particle driven instabilities in compact 3-D systems are substantial and will likely evolve through a number of reduced models and require access to cutting-edge computing resources.

Work at ORNL on energetic particle destabilized MHD instabilities in stellarators is focusing initially on the accurate calculation of the continuum spectrum and Alfvén eigenmode structure for low aspect ratio systems, taking into account the full extent of toroidal, helical and poloidal couplings that can be present in such devices. Tools will next be developed for obtaining linear stability thresholds in the presence of various fast ion populations. This work will then finally be extended to models to address the nonlinear evolution of these modes, taking into account both MHD and wave-particle resonance and trapping effects.

There are plans to use the M3D hybrid code at PPPL to simulate energetic particle driven modes in stellarators. The hybrid model describes bulk plasma as a single fluid or two fluids and the energetic component as gyrokinetic particles. The model is fully self-consistent including both fluid and particle nonlinearity. The M3D code has been used to simulate the internal kink mode, fishbone, and TAE modes in axisymmetric tokamaks [5]. Work is in progress to extend the code to stellarator geometry. The new code will be used to study both linear stability and nonlinear evolution of energetic particle driven modes in stellarators.

Additionally, there are also plans to convert the HINST and NOVA codes to stellarator geometry in the future. The HINST code solves a system of kinetic equations for medium to high-n modes including kinetic ballooning, drift and Alfvén modes. The NOVA code solves for global TAE modes.

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2. Confinement

Areas of High Priority Research

Confinement issues are generally divided into two classes: neoclassical and turbulent. The issues in neoclassical confinement in which studies are most critically needed involve the ambipolar electric field required for quasineutrality, the bootstrap current and its effect on island formation, flow damping rates and the effects of symmetry, and the efficient evaluation of the losses of high energy particles, either from external heating or fusion alphas. Also needed is improved understanding of numerical results from the various codes through benchmarking, and comparison with analytic theory and experiment. The stellarator-specific issues in turbulent transport that are most critically required are the effect of stellarator design on turbulence drive mechanisms and shear flow stabilization, and how turbulence affects neoclassical transport and the ambipolar electric field. Unlike a tokamak, where the ambipolar potential does not affect the neoclassical fluxes to lowest order, in a non-axisymmetric device neoclassical rates are greatly affected by the potential. Hence, a very different coupling is expected between neoclassical and turbulent flux contributions to the potential and to the behavior of transport barriers.

The ambipolar electric field strongly affects and is affected by neoclassical transport, it is closely related to the rate of rotation damping, and the radial gradient of this field can reduce turbulent transport. In theoretical models of neoclassical transport, the ambipolar field can have either sign at low collisionality. However, at high collisionality only the ion root, which confines ions, is generally present. Even in low collisionality experiments, the ion root also appears to be more easily accessed than the electron root, which generally has better ion as well as electron confinement. The sign of the ambipolar field also arises in the important issue of impurity and ash accumulation; again, the electron root is preferable. The standard assumption is that the electric potential is constant within a magnetic surface. It has been pointed out that this assumption may not be consistent with quasi-neutrality, and that variations in the electric potential on the magnetic surfaces can cause large enhancements in neoclassical transport [1]-[3]. Even when a constant potential is consistent with quasineutrality, this may not be the unique or stable solution. Given the magnetic field associated with a stellarator equilibrium, the losses of high-energy particles can be determined numerically. Unfortunately, numerical evaluations are prohibitively long for use in optimizers. Improved methods for estimating these losses would have a large impact on stellarator optimization, particularly for reactor grade plasma devices for which alpha confinement is an issue.

Turbulent transport is an area of active study for tokamaks as well as stellarators, which share many of the same issues. However, stellarator specific studies are needed to determine how stellarator design can affect turbulent transport and how turbulent transport affects neoclassical transport and the ambipolar electric field. The local geometric properties of the magnetic field, local shear and curvature, affect plasma turbulence and both can be modified by stellarator design. However, such considerations have never been used in the designs of optimized stellarators. The radial shear in the rate of plasma rotation can greatly reduce turbulent transport. However, the implications on the desirable level of rotation damping in stellarators have not been estimated. In particular, it is not understood how strongly viscosities depend on the degree of quasi-symmetry, and what the possibilities are for wave-driven, or turbulence self-driven flow shear.

In tokamaks neoclassical and turbulent transport are studied as independent processes, and turbulent transport is usually thought to affect plasma rotation only through the cross-field viscosity. The independence of neoclassical and turbulent transport and the absence of a direct effect of plasma turbulence on the in-surface, or parallel, viscosity are related assumptions. If untrue, there could be unique turbulent transport effects in the stellarator. In the least favorable regime for neoclassical transport, the "$1/\nu$" regime, the effect of plasma turbulence may be to reduce the neoclassical transport by breaking the longitudinal adiabatic invariant.

These critical research needs in the theory of confinement should be considered as part of a larger research program, which is outlined below.

Drift Orbits and Neoclassical Transport

Adequate confinement in a stellarator requires more careful design than in axisymmetric configurations. Several different principles have been developed to ensure adequate confinement. These include: 1) quasi-symmetry [4], [5] (QS), with quasi-axisymmetric (QA), quasi-poloidal symmetry (QP), and quasi-helical (QH) variants in which the magnetic field strength in Boozer flux coordinates is well approximated by a function that depends on only a radial flux coordinate, Ψ, and a single angular coordinate, 2) quasi-omnigenity [6], [7] (QO), in which the longitudinal action of the particles is approximated by a function of Ψ alone, implying vanishing bounce-averaged radial drifts, 3) a strong magnetic well, which can be produced by the diamagnetic effect of the plasma. Experimental devices have been built or are planned which can test these principles. Less explored are more recent concepts, providing a broader space of opportunities, 4) isometry or approximate omnigeneity [8], [9], properly containing the QS class while still having vanishing bounce-averaged radial drifts, and 5) pseudo-symmetry [10], in which only ripple wells are eliminated, removing the typically dominant contribution to stellarator transport from ripple-trapped particles.*)*

Analytic theory exists for both particle orbits and transport coefficients in idealized stellarator fields of the standard, QS, and QO types, but numerical means are necessary for accurate confinement assessments of configurations departing from this highly constrained subset. First principles calculations of particle and heat transport are made by directly solving the kinetic equation, which is done in the DKES code [1] using a variational principle, or by way of various Monte Carlo (MC) techniques. The MC transport codes are of three types: (a) full-f [12]-[14], (b) delta-f [15], [16], and (c) delta-f plus fluid pressure [17], [18]. The lower the collisionality, the more demanding are the calculations. Moving toward the analytic side of the spectrum of numerical tools, Nemov and Kernblicher [19] have developed a formula for an equivalent ripple that is often used as a measure of the confinement quality of stellarator orbits based on the $1/\nu$ neoclassical transport regime, and 1-D transport codes can make use of this in computing self-consistent profiles. However, careful comparisons between the detailed output of first principles codes at differing collisionalities versus the effective ripple and other analytic approximations need to be made.

In stellarators, unlike in axisymmetric configurations, the transport coefficients for thermal particles generally depend sensitively on the ambipolar electric field E_{ambi}, which must be determined as part of a self-consistent calculation. Although simpler approximations exist, the ambipolar field is most accurately calculated using delta-f codes [18]. At low collisionality there are usually two stable values for E_{ambi}, [20], [21] the ion and the electron roots, depending on which species the electric field confines. Only the ion root is generally present at high collisionality. The electron root, when it exists, is preferable, giving lower transport and tending to expel impurities. However, historically it has been difficult to achieve an electron root using purely thermal plasmas [22]. It was achieved in CHS and W7-AS with poorly confined, non-thermal electron populations [23], [24], and only recently with thermal electrons, on LHD [25].

The confinement of energetic particles is insensitive to the ambipolar field E_{ambi}. Such particles may be lost through prompt loss, when the particle's collisionless banana or superbanana orbit intersects the wall, or through stochastic loss for toroidally trapped [26], passing [27], [28], and transitioning [29] particles. Analytic theory exists for transport of energetic particles as well, but again, first-principles calculations with full-f MC codes are necessary for reliable assessment of their loss rate in all but very special configurations, and the relation of these loss rates to that of the thermal bulk or with the $1/\nu$ effective ripple is not simple, nor even always monotonic.

Methods for calculating a number of neoclassical effects are still under development and have had only limited application to the study of configurations. These include: 1) the accurate inclusion of momentum conservation [18], which is important for quasi-symmetric systems as it is in tokamaks, 2) the determination of the in-surface flows [30], [31] that result from a torque, 3) calculations of the bootstrap current, 4) the confinement properties of impurity species, and 5) the determination of the variation of the electric potential within the magnetic surfaces that is required for quasineutrality [32]. This variation would cause drifts across the magnetic surfaces and modify the transport. An approach to these issues by extending the DKES code to higher dimensionality (4-D) is presently under study. Such an extension will especially be needed to deal with low collisionality stellarator regimes where radial drift related superbananas are important.

Fluid Model for Confinement Analysis

Plasma flows and rotation have become an important part of fusion confinement research, in part because of the theoretical understanding of the enhanced confinement modes in tokamaks and stellarators [33]-[35]. For tokamaks, the NCLASS code solves the force balance equations for every species in conjunction with the transport equations used to model observed plasma rotations, flows, and transport phenomena [36]. Although a 3-D system is different from a tokamak, the force balance equations are practically the same except for plasma viscosity in the symmetry direction. However, most of the needed information about plasma viscosity in a system is either known or can be approximated by analytic calculations and benchmarked against the DKES or a Monte Carlo code. It would be beneficial to extend the NCLASS type of code to model rotation, flows, and transport phenomena in realistic 3-D confinement experiments.

Microstability and Turbulent Transport

The empirical scaling of transport in stellarators is similar to that in tokamaks [37], so it is generally assumed that the turbulent transport mechanisms are the same in the two configurations. However, features of the configuration that are thought to affect microstability and transport can be controlled in stellarator designs. These include the sign of the global shear, the variation in the local shear, the relative locations of the regions of bad curvature and the trapped particles, and the profile of the ambipolar electric field. No stellarator has been designed to minimize turbulent transport, but such an exciting prospect could become feasible with tools now being developed and applied to better understand the experimental and theoretical situation.

Recent experimental and theoretical work on the reduction of turbulent transport in tokamaks has focused on the spontaneous development of transport barriers, which are regions of high ExB flow shear. It is not yet understood whether the development of such barriers places a strong constraint on flow damping and, therefore, on stellarator design. The radial electric field is more robustly determined by ambipolarity in stellarators than in tokamaks. This electric field can show strong gradients, which would be particularly strong in a transition between an electron and an ion root, and there is experimental evidence indicating [38], [39] that this is effective in suppressing plasma turbulence. Again we need to understand how viscosities depend on the degree of quasi-symmetry and what are the possibilities for wave-driven, or turbulence self-driven flow shear. Further experimental and theoretical work is needed to determine the circumstances under which such transport barriers can be produced. In particular, diagnostics for E_r, B, and fluctuation measurements are essential to enhance the level of theory-experiment comparisons to the level achieved by major tokamak teams.

In tokamak simulations of turbulent transport, neoclassical transport is generally assumed to be an independent effect. However, turbulent transport breaks the longitudinal adiabatic invariant, and it is the conservation of this invariant that causes the most virulent for of neoclassical transport in stellarators-transport that scales inversely with the collision frequency.

Alfvénic eigenmodes and fishbones are expected in stellarators as well as in tokamaks, and the self-consistent interaction of these being driven by energetic particles, and of the energetic particles being transported due to these modes, should be of concern in stellarators as well.

The gyrokinetic (GK), gyrofluid, and linear microstability codes that have been developed for studying the microstability and anomalous transport in tokamaks need to be adapted to stellarators. This has already been done for the FULL microstability code,[40] the GS2 flux tube GK code [41], [42], and the EUTERPE global GK code [43], and these are beginning to yield results. The global GK code, GTC [44], should soon have this capability as well.

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3. Modeling and Diagnostics

Modeling and diagnostics covers a broad spectrum of activities. Modeling is addressed here in terms of predictive and interpretive capabilities for energy, particle, momentum, and magnetic flux evolution on the 'transport' timescale. Within the predictive and interpretive capabilities, there are several levels of complexity that can be included in the physical models, with choices depending on the detail required in the analysis and computational resources. In the following discussion, diagnostics are only discussed in terms of interpretive modeling tools.

Transport simulation, whether in axisymmetric tokamaks or non-axisymmetric stellarators, involves the coupling of MHD equilibria in higher real space dimensions with the evolution of one-dimensional time-dependent quantities that are constant on a flux surface. Therefore, the extension of transport analysis from tokamaks to stellarators primarily involves generalizing the mapping between real space and flux surfaces from 2-D to 3-D—a non-trivial effort. Progress was made in this effort in the U.S. on the ATF project, and some additional progress has been made in both the U.S. and world stellarator efforts since then. But, progress in theoretical models, physics understanding from experiments, computational models, and computing abilities all indicate that substantially greater improvements can be made.

Areas of High Priority Research

There presently is no actively used predictive or interpretive transport code in the U.S. for stellarators—one that incorporates 3-D MHD equilibria into its analysis as well as other physics governed by the loss of axisymmetry. The stellarator community needs to identify how to fill this void, which may be most effectively accomplished through collaborative development because of the magnitude of the work involved. It must be noted, however, that much of the work discussed in the other sections of this report would lead to a more credible computational physics basis for the development of a stellarator transport code. In addition, many of the computational tools and models used in optimization codes provide the appropriate level of depth and computational efficiency to be incorporated in time-dependent simulations. Therefore, a time-dependent modeling effort based on advanced theoretical and design tools could easily be built on the present and planned theory, design, and optimization tools in the program.

Coupled Transport Equations

The PROCTR (predictive and interpretive) [1] and WHIST (predictive) [2] transport codes were used for ATF and other stellarator analyses in U.S. collaborations with Germany and Japan. These codes relied on using fixed 3-D MHD VMEC equilibria for the geometry rather than coupling to active MHD equilibrium calculations, because applications were for low beta plasmas with minimal internal currents. PROCTR is no longer in use in the U.S. and WHIST has been mothballed, so there is no presently active 3-D transport simulation code in the US. For NCSX and QPS applications, evolution of the poloidal magnetic flux would be a necessary addition to the models for particle and energy transport. Generalization of the poloidal flux evolution models for non-axisymmetric plasmas to include bootstrap current, inductively driven current, and externally driven currents in addition to the externally supplied transform has recently been completed for general magnetic flux coordinate representations (including non-straight field line systems) in support of the NCSX and QPS studies [3], but with the assumption of simply nested surfaces without magnetic islands as is usually done in tokamak transport analyses. The accommodation of this generalized magnetic flux evolution model in any code that uses a Grad-Hogan scheme is relatively straightforward (requiring computation of the susceptance matrix for coupling toroidal and poloidal fluxes with poloidal and toroidal currents from the MHD equilibrium).

The PROCTR and WHIST codes also included fairly advanced particle and energy transport models, including the strong neoclassical transport matrix effects of heat flux driven density gradients and particle flux driven by temperature gradients. The neoclassical model was limited to a single helicity, and used in the WHIST code to calculate the evolution of the radial electric field (in terms of flux surface quantities, the electric potential gradient) [2]. Improvements would require generalization of the neoclassical models to cover a broader range of geometries (e.g., low aspect ratio effects), the effects of energetic particles (large orbits and direct losses) and turbulence effects in the ambipolarity constraint, and inclusion of the momentum equations for poloidal and toroidal rotation. The models for anomalous transport were empirical, and could be improved by making extensions to the turbulence models used in present-day tokamak simulations.

The NBI and neutral source terms in PROCTR essentially used axisymmetric approximations to the plasma to adapt tokamak routines. In the present computational environment, fully 3-D neutral, NBI and RF codes are possible, but require development. The geometry routines for extracting the 3-D mappings between local real space coordinates and flux surface variables were developed as part of the ATF analyses and upgraded recently as part of the NCSX and QPS studies. These include the AJAX module that provides the mapping between Cartesian, cylindrical and magnetic flux (VMEC) coordinates and calculation of flux surface averages, and the TRACK routine that provides the magnetic coordinates along a line-of-sight for inversion of diagnostic data. Extensions to other magnetic flux representations (e.g., Boozer coordinates) would require additional effort.

MHD Equilibria

The VMEC representation was chosen for the representation in ATF analyses because of the identity relationship between the 3-D toroidal flux coordinate and the toroidal cylindrical coordinate that reduces the coordinate transformations to 2-D and greater computational efficiency. There are a wider variety of 3-D MHD equilibria presently available that need to be reviewed along with the information that is desired from transport analysis. The extent that the analysis will be coupled with MHD stability and whether magnetic islands are to be included make a major impact on the choice of MHD equilibrium representation and its interface with a transport code.

Magnetic Flux Reconstruction and Control

NCSX is an example of a current-carrying, high beta stellarator. Both of these features imply that the plasma boundary shape will be a function of internal profiles, as well as the external field. Discharge evolution studies have shown that the key to a stable trajectory with adequate quasi-symmetry is the maintenance of the plasma shape. Because we need to control both stability and quasi-symmetry it will be necessary to control the shape in more than a single toroidal plane by varying the currents in both the helical and poloidal fields. In order to maintain the boundary shape, it is necessary to find a set of (magnetic) signals which have adequate sensitivity and control algorithms that relate these signals to the flux at the coils. To accomplish this task we need to postulate a set of magnetic diagnostics, and compute their response functions for a large set of free-boundary equilibria sufficiently "near" the desired solution. By sufficiently near, we mean that they are close enough so that the same control algorithms will work, but sufficiently far from the desired solution so that they encompass solutions that do not preserve stability or quasi-symmetry.

A first task is to develop the response function formalism for a non-axisymmetric system. This involves a numerically efficient calculation of Green's functions, G, relating the currents to the *k*th magnetic diagnostic, S_k,

$$S_k = \sum_i G_{ki}^{ext} I_i + \sum_m G_{km}^{\phi} J_{\phi_m} + \sum_n G_{kn}^{\theta} J_{\theta_m}$$

Here, I_i are the external coil currents, $J_{\phi\, m}$ and $J_{\theta\, n}$ are the toroidal and poloidal plasma current densities discretized on the computational grid, and the G's are the Green's functions relating the currents to the *k*th magnetic diagnostic. This will allow U.S. to install a generic diagnostic, e.g., a flux loop specified by its location and orientation, and calculate a matrix of the coupling of the magnetic diagnostic to the coil currents and the internal plasma current. These response functions will be needed for flux loops (both components), local magnetic probes, and saddle coils.

Having obtained the responses for this large set of equilibria, it is necessary to relate the responses to control algorithms specifying aspects of the shape. A control algorithm will relate a specific feature, e.g., the outer midplane radius in the ϕ=39° plane, to some subset of the signals. This might be done by linear regression, but a neural net treatment is more likely to be successful. To the extent that successful relationships are not found, it will be unclear whether the deficiency is the proposed control points or the sensors are insufficient. Iteration of the procedure is likely.

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4. Wave-Particle Interactions

To achieve high performance in stellarators will of course require heating but will also require all the tricks of control by RF needed for Advanced Tokamaks, but in 3-D. The present view of confinement in stellarators is that it is basically anomalous. Reactor devices projected from the ISS95 scaling tend to be very large indeed, so production and control of transport barriers to improve confinement will therefore be needed, perhaps connected with the electron root of the neoclassical transport. Whereas we have normally thought of stellarators as zero current devices, the new compact stellarator concepts being considered for the U.S. program (quasi-axisymmetric stellarator and quasi-poloidal stellarator) do have net current. Allowing for a current profile lends added flexibility to satisfy the requirements for shear, needed for stability, magnetic configuration, and needed for improved neoclassical confinement at low aspect ratio. Although it seems to be possible to make the net current largely consistent with the bootstrap in these devices, the bootstrap profile again depends on pressure and iota so that profile control will certainly be necessary. Another consideration for near term experiments is that energetic particles are difficult to confine in non-axisymmetric devices, particularly small ones with modest magnetic field. Thus developing a technique for bulk ion heating such as with fundamental ICRH or perhaps IBW would be very beneficial.

A complete theory of RF interaction with plasma is conveniently separated into pieces, which interact but to a large extent can be studied independently: 1) analysis of antennas, 2) wave propagation and absorption, 3) calculation of plasma response—modification of the distribution function, power deposition, driven current, driven flow, and 4) coupling of RF models with equilibrium, stability and transport models. In comparison with 2-D systems, in every area the theory is less developed, the codes are computational much more demanding, and there is much less experience verifying the models with experiments.

To fulfill programmatic goals, the theory and computational models must satisfy certain requirements:

<u>Antenna Analysis</u> – There are two kinds of theoretical needs: 1) it must be possible to reliably predict the spectrum of launched waves, and 2) we need to predict the intense near fields of the antenna, and the interaction of these fields with the edge plasma to maximize antenna survival and minimize impurity generation. This necessitates that the 3-D launching structure be modeled with sufficient realism and that the coupling of this structure to the plasma be included. Existing codes do allow for 3-D structures in the antenna, but usually approximating the geometry as straight, rectilinear segments. The present state of coupling such antenna models to the plasma assumes a 1-D, plane stratified plasma. However, even in tokamaks, the plasma boundary only approximately conforms to the shape of the antenna so this is an important needed generalization. Since neither toroidal nor poloidal modes are normal modes of the system, it is not clear how to even characterize the launched spectrum in a stellarator or, when using geometrical optics, how to initialize the rays to properly represent the launched power.

Wave Propagation and Absorption –The wave equation must be solved in the plasma to predict the field structure and the ultimate absorption of the waves by the plasma particles. For situations in which geometrical optics is valid, the extension from 2-D to 3-D is trivial provided the propagation is adequately described by the uniform plasma conductivity tensor. However, for frequencies much below the ECRF, or in any circumstance for which mode conversion is important, the wave equation must be solved. The most obvious challenge is the enormous computational load of solving a 3-D rather than 2-D differential equation. Codes have recently been developed which solve the warm plasma wave equation in 3-D but the size and running times are so large that only the fast wave can be resolved and only a few cases can be run on the largest MPP computers. Critical needs are both to find ways to accelerate these codes and to reduce the required number of unknowns, for example, by adaptive griding or the use of alternate basis functions.

Another issue is the validity of present approximations used in constructing the wave plasma current. The plasma current $\mathbf{J}_P$ induced by the wave fields $\mathbf{E}(\mathbf{x},t)$ is a non-local integral conductivity operator on the field

$$\mathbf{J}_P(\mathbf{x}) = \sigma \circ \mathbf{E} = \int dx' dt' \mathbf{K}(\mathbf{x},t,\mathbf{x}',t') \cdot \mathbf{E}(\mathbf{x}',t').$$

This is typically approximated by the uniform plasma conductivity tensor, or a first order generalization thereto, which effectively assumes that the parallel velocity of particles is constant along unperturbed orbits and that the wave field has a well-defined constant, $k_{\|}$. The scale length for parallel variations in a stellarator is much shorter ($\sim 2\pi R / N_f$, where N_f is the number of field periods) than in a tokamak ($\sim 2\pi qR$) so that non-local effects may be much more important than in tokamaks.

Plasma Response – The long time-scale response of the plasma distribution function, $f_j(\mathbf{x},\mathbf{v},t)$, is obtained as a solution of the Fokker-Planck equation. Waves also induce macroscopic local sources to the plasma of: power, $W_{RF}(\mathbf{x})$, driven current, $j_{RF}(\mathbf{x})$, and plasma flow, $\mathbf{V}_{RF}(\mathbf{x})$. All of these are quadratic operators on the wave field and must be calculated using the full detail of the wave field solution and the necessary structure of the underlying plasma equilibrium. It is certainly the case that all of the models for plasma response need substantial generalization to adequately treat 3-D systems. The long time-scale response of the plasma distribution function is most reasonably expressed in constant of motion space. However, in most stellarators such space is very difficult to characterize because of the large number of classes of trapped and transitional orbits and because of the lack of enough constants of motion. In addition, radial drifts away from flux surfaces are likely to be more of an issue than for tokamaks. Perhaps the most general approach is through Monte Carlo and orbit averaging techniques although this results in a very high dimensionality problem. Similar difficulties exist in calculating driven current and driven plasma flow. In the linear approximation, the quantities needed to carry out the adjoint calculation of driven current are not available for 3-D systems—the appropriate Green's function and the quasilinear flux. Also needed is a reliable calculation of the flow-damping rate for wave driven flow drive.

Integration with Transport and Stability Theories – The discussion of transport models is in another section. However, it is generally true that because of the computational load of just calculating the wave heating and current, only the most simplified of wave source modules are incorporated in the transport codes. In addition, the wave propagation and absorption modes have not been made at all self-consistent with the evolving distribution function or plasma profiles.

A number of these issues are being addressed within the SciDAC project on wave-plasma interactions, particularly regarding issues of wave solvers and incorporation of specified non-Maxwellian distributions. However, it is not within the scope of SciDAC to develop the complete RF modeling capability in 3D, especially in the area of solution of the Fokker-Planck equation. Development of a higher dimensional drift-kinetic equation solver that includes both cross flux surface transport and energy scattering will be essential in the long term.

5. Optimization of 3-D Magnetic Confinement Systems

Stellarator optimization has progressed in the past few years to the point where the design capabilities we now have and anticipate rivals that which is realized in, for example, the aerospace industry design of futuristic, low drag aircraft. The future of stellarator optimization will certainly be impacted by the direction of development that new computer platforms take, the development and application of global optimization techniques, and a new focus towards the optimization of burning plasmas and reactors. In the following, we outline the issues and opportunities in stellarator configuration optimization, coil design, configuration flexibility and control, and burning plasma and reactor optimization.

Plasma Configuration Optimization

Plasma configuration development has been at the forefront of stellarator optimization research for a number of years. Advances in this area will likely result from a transition from a "reverse engineering" approach to a "forward engineering" approach, the identification of a minimum set of control parameters, and the development of global optimization techniques.

Outstanding progress has been made in the design of stellarator experiments using the "reverse engineering" method pioneered by Nuhrenberg, et al., in Germany [1]. This method decouples the plasma configuration design from the coil design. First, an attractive physics configuration with assumed profiles is sought by adjusting Fourier coefficients describing the plasma shape. A cost function measures the suitability of the configuration based on a combination of physics objectives such as β-limit value (for kink and ballooning modes), weighted average of effective helical ripple over several flux surfaces (measuring the level of neoclassical transport), etc. Finally, an optimization algorithm such as the Levenberg-Marquardt scheme [2] is used to minimize the cost function.

Recent substantial improvements in the speed of the VMEC free-boundary code [3] have allowed a return to a "forward engineering" approach to designing stellarators such as NCSX and QPS. In the forward engineering approach, the plasma and coil systems are developed together. For free-boundary equilibrium optimizations, Fourier components of current potential for a sheet current on some winding surface may be targeted. This may be a preferred approach for reactor design (see below) since coil feasibility measures are more easily incorporated.

A primary outstanding issue in configuration optimization is the disparity between the typically large number of Fourier coefficients varied during the optimization and the relatively small number of physics properties that define an attractive configuration (kink and ballooning stability, quasi-symmetry measure, neoclassical transport). One would expect, therefore, that only a few plasma shape parameters are primary determinants of the desired properties. At present, for stellarators, the shape parameters that determine

MHD stability are unknown. While initial attempts to define the relevant control parameters have begun [4], more follow-up work is needed.

Most stellarator optimizers used to date have employed "local" search algorithms, such as the steepest-descent and Levenberg-Marquardt methods which, while efficient in suitable cases, often become trapped in local minima of the cost function, requiring human involvement to dislodge the system from such minima. For this reason, an optimizer with a global "differential evolution" (DE) search algorithm [5] have been developed. The DE algorithm is similar to a Genetic Algorithm (as applied, for example, to some early NCSX saddle coil designs) but suited to exploration of continuous spaces. The DE follows a large population of candidate configurations, naturally providing a map of the underlying stellarator landscape. Such "global" search algorithms require more function evaluations for a run, but are much more robust to being locally trapped. In addition, they provide an understanding of stellarator parameter space that local search algorithms do not. Further experimentation with and development of such global search strategies should be carried out to exploit the virtues of these methods.

Coil Design

A crucial element of stellarator design is obtaining a realizable coil set. The challenges in coil optimization include the large number of possible coil sets for a given plasma equilibrium, the coil complexity, the definition of an appropriate set of coil parameters, and the high computational cost of coil optimization.

In the reverse engineering approach, after an attractive 'reference' plasma configuration is identified, a coil set must be determined. A coil 'winding surface' is defined at some appropriate separation distance from the plasma and a current sheet is determined on the winding surface which provides a magnetic field that matches B-normal from the plasma, to some suitable precision, evaluated at the plasma-vacuum interface. The current sheet is discretized (for example using a Genetic Algorithm [6]) to provide a filamentary description of the coils, and the ability of the discrete coils to reproduce the reference plasma configuration is tested by calculating a free-boundary equilibrium with optimized coil currents. Errors in equilibrium reconstruction are correlated with errors in B-normal matching. Matching errors increase with increasing winding surface-to-plasma separation distance, d_{c-p}, as does current sheet and coil 'complexity'. The increase in coil complexity is due to ill conditioning and is non-physical. SVD methods have had partial success in solving the ill-conditioning problem. However, additional filtering methods must be developed if this coil design procedure is to be successfully applied to reactor design (where d_{c-p} is large). It has been suggested that high mode number components in the plasma contribution to B-normal should be eliminated when establishing the baseline configuration [7]. The elimination of such components, in combination with SVD methods, may provide the necessary filter to produce feasible coils.

A second coil design procedure is realized in the COILOPT code [8]. Here, a coil winding surface is parameterized by a Fourier representation in poloidal and toroidal angle variables u,v, respectively. Coils on the winding surface are represented by a

Fourier winding law v(u). COILOPT varies the coil geometry and coil currents, seeking to minimize the average and/or maximum mismatch between the B-normal from the coils and plasma at the fixed plasma boundary. Engineering constraints such as minimum coil bend radii and coil-to-coil separations are included in the COILOPT cost function.

Present approaches use many Fourier coefficients in R and Z to represent the plasma boundary. This has a potential drawback of over-specifying the plasma boundary shape in order to satisfy a limited set of physics requirements. If the plasma shape is too precisely defined for the needs of the physics mission of the machine, we run the risk of obtaining coils that are unnecessarily complicated (too twisted for example). This may complicate the coil engineering and add to the cost. An attempt has been made to place some of these ideas on an algorithmic footing but further work is needed.

Once a coil set has been found which can adequately reconstruct the reference configuration, flux surface quality is checked by performing free-boundary PIES calculations [9]. If surface quality is not adequate, a 'surface healing' algorithm is applied which modifies the geometry to improve surface quality. The healing algorithm is not presently required to preserve kink/ballooning stability and transport physics. There is, therefore, no guarantee that the healed coil set will satisfy all desired physics criteria. Since the present surface healing method is computationally expensive, presently comes on the heals of an already expensive COILOPT design, and is not guaranteed to preserve the physics of the COILOPT design, it makes sense to explore the potential for making PIES the core equilibrium solver within COILOPT and to target good flux surfaces within the optimizer.

It has recently been shown that compact (low aspect ratio) plasma configurations can be produced with simple, planar coils. An example is the plasma configuration being considered for the Columbia Non-neutral Torus experiment [10]. A small number of planar circular coils (some of them tilted and interlocking) can be shown to produce excellent magnetic surface quality and significant rotational transform. An investigation of stable high beta compact stellarator configurations with good transport properties formed by simple planar coils should be encouraged. If attractive configurations can be formed with such simple coils, it would greatly improve the prospects for a stellarator-based path to a fusion power reactor.

Finally, whether one uses the LM or DE approach to coil design it is clearly advantageous to increase the speed of a typical optimization calculation. The number of independent variables, N_{var}, is the main determinant. For plasma configuration design N_{var} ~50 typically (e.g., number of Fourier coefficients describing the plasma boundary), and for coil design N_{var} can be as many as 200-300 (e.g., number of harmonics defining the coil winding surface and coil winding law). Improvements in the efficiency of the representation used to describe the shape of the plasma or coils will translate into a smaller N_{var} and a consequent decrease in computational turn-around time, and to a more efficient search of the multidimensional search space. A possible alternate approach that does not require a winding surface is to describe each coil as a perturbation of a tilted circular coil (issues of coil overlap, of course, have to be resolved). Even with a winding surface, we should eliminate any null space associated with the non-uniqueness of the angle variable (as was done by Hirshman/Breslau for the plasma flux surface description). Without the time pressure of project deadlines, surely we can do better than what we have at present. Methods for accelerating the convergence of the DE method to a global minimum by incorporating a simplex hybrid approach have met with success in airfoil design in aerodynamics (where like the stellarator application one is confronted with computationally expensive cost function evaluations). We need to test a similar approach in the stellarator applications.

Configuration Flexibility and Control

It is fair to say that we do not yet have a detailed understanding of what design parameters lead to good flexibility for a compact stellarator experiment. For this we will need to know which components of the poloidal and toroidal fields are responsible for controlling <u>separately</u> the kink and ballooning stability and the transport. Equivalently, we need to know what are the 3-D shape parameters that control these physics properties. A desirable goal is the same level of understanding we have in tokamaks where elongation and triangularity are well-known to be major determinants of the MHD stability.

The NCSX coil system (comprising three independently-controlled modular coil types per toroidal period, an auxiliary 1/R TF coil system, and five independently-controlled axisymmetric PF coils) has been shown to have considerable flexibility. The coil currents in NCSX can be changed in such a way that β-limits can be varied over a substantial range while maintaining the same level of neoclassical transport; the effective helical ripple can similarly be varied while maintaining a fixed β–limit; the iota profile can be varied by 50% at fixed plasma current and average toroidal magnetic field. For NCSX, changing the shape on the "bullet" cross section (v = 1) of the plasma is known to have a strong effect on kink stability—squareness, and outboard indentation at v=1 appears to benefit stability—however, we do not yet understand the global dependence of 3-D shape on stability. Similarly, the dependence of 3-D shape on quasi-symmetry is not fully understood. Some preliminary understanding has been achieved using a perturbative Control Matrix method [10], but more work is needed to complete the study. The potential payoff for such detailed understanding is great, however, because it can be used to design efficient coil sets for future stellarator designs as well as for the design of retrofit coils for existing stellarators.

Reactor Optimization

Two new challenges facing the stellarator theory community are the optimization of a burning plasma experiment and coming up with an attractive compact stellarator reactor configuration. The optimization targets, criteria, and strategies will be different from those used to design present day experiments. Optimization of a burning plasma experiment will lead to the further development of optimization targets such as those related to energetic particle confinement (beams, ICRF tails and alphas) and optimization for fast ion-destabilized Alfvén instabilities. For the optimization of a minimum-cost reactor, the plasma-coil spacing and alpha confinement would be more important metrics, while plasma beta and aspect ratio would be free parameters. On the engineering side, more attention would have to be paid to reliability and maintainability considerations, which could prompt a re-examination of simpler concepts, such as planar-coil ideas. An approach to reactor optimization should build on the existing NCSX/QPS tool set but new or improved tools will be needed. Experience also teaches that some physics research, i.e., new theories or models, will likely need to be done to solve this problem.

Summary

In summary, the driving motivation in optimization research is the goal of including more of the relevant physics directly in the optimization process. This requires enhancing the speed of the existing codes, reducing the number of control parameters, and improving our understanding of the relevant physics issues. A near term goal in stellarator optimization is to complete the design of the NCSX and QPS stellarator experiments which would include completing the merger of the coil and configuration optimization processes. A long term goal is to include a good measure of surface quality in the optimization process (which may require a new code, enhancing existing codes, or embedding the optimizer within PIES).

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6. Edge Physics Issues in Complex 3-D Systems

Introduction

Because of the existence of open field lines in the scrape-off layer region, edge problems are inherently multi-dimensional, even in nominally axisymmetric systems such as the tokamak. Thus an improvement in our capability to treat 3-D problems will have great benefits for edge physics. While the preponderance of 3-D problems are found in stellarator research, there are also areas of tokamak work where improved 3-D theoretical capability would have a significant impact.

Plasma boundary control and divertor development in stellarators

The plasma edge in stellarators does not have the ordered magnetic field structure that axisymmetric systems have. Whereas the region inside the separatrix of stellarators has nested surfaces, the outside can typically have islands, remnants of islands, or regions of stochastic field lines. Edge control with divertors is realized in different ways, depending on the type of stellarator. In LHD and CHS intrinsic helical field line diversion is used to create the Helical Divertor, or additional coils are used to create the Local Island Divertor. In W7-AS and W7-X, as well as in the compact stellarators NCSX (National Compact Stellarator Experiment) and QPS (Quasi-Poloidal Stellarator), intrinsic islands are used to create Island Divertors or Ergodic Divertors

The important benefits to be gained from a full 3-D treatment of the plasma boundary and divertor issues become clear from the evolution of the divertor configuration in the W7-AS stellarator. In the final stage of its divertor development, leading to a fully 3-D configuration, the plasma performance improved considerably. The trend of impurity confinement was even reversed from an unfavorable increase with density in previous stages of the plasma-facing components, to a favorable decrease with density with the final (3D) divertor configuration.

Owing to the complexity of the island and stochastic field topology in the SOL of stellarators, recycling, heat removal, and impurity control all have strong toroidal non-uniformities. The systematic divertor development on W7-AS has shown that careful control of plasma-wall interactions leads to performance improvement in stellarators as well as in tokamaks.

3-D issues for tokamaks

Many edge physics in tokamak research are especially sensitive to 3-D effects. The sensitivity of the separatrix location to currents flowing in the so-called "vacuum" region has been well established, and there is a strong sensitivity of exhaust efficiency to 3-D effects.

For both the stellarator and the tokamak areas, however, the underlying physics mechanisms are complicated and not understood and, therefore, the development of 3-D transport models is a prerequisite for reliable prediction of plasma behavior.

Major topic areas

The major challenges in edge physics for complex systems, encompassing stellarators and 3-D tokamak topics, lie in the areas of magnetic equilibrium, turbulence, power handling and particle control, direct velocity-space loss, impurity control, and coupling to core physics. These topics are addressed in turn, in greater detail, below.

Magnetic Equilibrium

There is a major need to develop an accurate and extrapolable model for detailed scrape-off layer (SOL) magnetics in complex systems. This is a topic that has heightened sensitivity for edge physics: it involves the fundamental magnetic control of the particle and power effluxes and determines whether steady state approaches are feasible. The issue is important both for near-vacuum and finite-beta magnetic fields in complex 3-D tokamak and stellarator systems.

For stellarators, the magnetic geometry is the fundamental basis for all edge transport models; indeed, the identification of the last-closed-magnetic-surface (LCMS), if one exists, defines the edge region. A standard procedure at present for determining **B** in the edge region is to use magnetic data from an harmonic analysis code (VMEC or similar), which assumes the existence of nested magnetic surfaces inside the LCMS as input to a code that calculates the magnetic field beyond the LCMS (e.g., MFBE, [1]). A successful match is then determined by a somewhat subjective determination as to whether field-line tracing of the external magnetic field gives the same LCMS as the core harmonic code. A better solution to this problem would be to make a unified calculation of the magnetic field in the core and the SOL with a code that does not assume the existence of flux surfaces (e.g., PIES).

A general goal for the stellarator magnetic calculations is to find equilibria with no, or minimal, magnetic islands in the core region. The consequence of this strategy on the form of the magnetic field in the SOL region needs to be better understood, i.e., when are SOL islands replaced by a fully stochastic region, and what is the consequence for edge transport, including plasma density and temperature at the LCMS? Coupled with these questions is that of the accuracy and convergence properties of the SOL magnetic field, since such properties may deteriorate as one moves outside the LCMS. In addition, the sensitivity of the SOL magnetic field to the expected small variations in coil shape and currents must be determined

The 3-D character of the magnetic field in tokamaks, especially in the edge region, is receiving considerable attention. Possible field errors have been identified in DIII-D which may affect rotation, wall stabilization, and particle exhaust efficiency. The iron core in JET can reduce exhaust efficiency for high triangularity configurations and in the JET-EP design. TEXTOR-94 is beginning experiments with a time-modulated ergodic divertor (Dynamic Ergodic Divertor) on the inner (high-field) side of the torus and the Tore Supra CIEL long-pulse experiments are conducted with high toroidal-field ripple, and other examples exist. To illustrate some of the tokamak issues, the present status of this topic is described:

3D non-axisymmetric effects in tokamak boundaries

The boundary region of a diverted tokamak is inherently sensitive to breakup of flux surfaces near the separatrix by resonant radial magnetic perturbations. This occurs because of the high edge magnetic shear that provides a large number of closely spaced low order resonant surfaces there. Perturbations as small as $\delta B_r/|\mathbf{B}| \sim 10^{-4}$ are found to be sufficient to destroy the primary separatrix, according to vacuum (non self-consistent) magnetic field line tracing calculations. When the separatrix is destroyed, the tokamak edge pedestal region takes on a 3-D character, with closed, nested flux surfaces replaced by island remnants embedded in a stochastic layer.

These effects have been studied for inner wall limited and diverted (upper and lower single null; double null) discharges in the DIII-D tokamak using the TRIP3D magnetic field line tracing code, for which magnetic fields for the axisymmetric equilibrium are provided by the EFIT equilibrium reconstruction solver. In tokamaks the external coils used to control performance-limiting locked and resistive wall modes constitute an external source of resonant radial field perturbations and also affects the boundary. For the example of DIII-D, TRIP3D results indicate that, on the basis of vacuum calculations, 5-20% of the poloidal flux surfaces inside the unperturbed separatrix can be lost depending upon C-coil current and plasma shape. This is corroborated in some cases by edge n_e and T_e profiles (measured by Thomson scattering) which show localized flattening across the region that is predicted to be stochastic. Because this stochastic layer forms just inside the separatrix in a region that becomes the H-mode pedestal/transport barrier zone, these predictions indicate that either: 1) the self-consistent plasma response heals the stochastic layer, or that 2) the existence of high performance plasmas with edge transport barriers is compatible with a stochastic boundary.

To resolve such questions, theoretical and computational tools must be developed to treat the plasma response to magnetic topology changes self-consistently, including such effects as: field line connection length versus collisionality, momentum input/plasma rotation, particle drifts, neutrals, and impurity screening. The formation of a stochastic layer in the region which becomes the H-mode pedestal/transport barrier zone suggests that understanding this physics could have a major impact on confinement concept viability, because of its impact on L-H transition physics and H-mode accessibility, edge pedestal height (and the closely related core performance), edge stability and ELM behavior; particle and heat exhaust, and plasma-wall interactions.

More general 3-D non-axisymmetric effects

Of course, the potential importance of 3-D magnetic structures is not limited to tokamak boundaries. It is essential to understand the various issues for stellarators in order to ensure good particle and heat flux handling, helium exhaust, and anomalous transport control via E x B shear suppression. Also, for the tokamak, resolution of these issues could serve as an opportunity for more active *control* of the boundary not merely a passive response to fix error. Thus, collaboration between the stellarator and tokamak researchers on these issues can enhance the capabilities of both. Stellarator research provides theoretical understanding of 3-D topological effects in toroidal systems, the use of state-of-the-art computational tools for addressing 3-D topological effects in toroidal systems, and innovative approaches to heat exhaust, particle and impurity control ("natural" divertor action). The tokamak research contributes existing flexibility for power and momentum input, shaping control, integrated equilibrium/stability/ transport diagnostics and an extensive database.

Existing tools: MFBE (with VMEC input), PIES, HINT, GOURDON, TRIP3D

<u>Instabilities and characterization of anomalous transport</u>

The region near the LCMS and into the SOL gives rise to various plasma instabilities that dominate cross-field transport. Substantial work has been done for axisymmetric tokamaks, but the impact of equilibria with full 3-D variations needs to be understood. Some major issues are the 3-D effects on: changes in instability characteristics, effect of field-line stochasticity, saturation mechanisms and transport barriers, non-diffusive transport, role of velocity-space loss cones, and ELM behavior.

The 3-D BOUT code is used as an example to illustrate the status and prospects in this area. BOUT has been used extensively to analyze turbulence in the edge region of tokamaks below the ion cyclotron frequency by solving the Braginskii fluid equations, including some extensions [2]. The extension of the BOUT code to stellarators, where the equilibrium itself is 3-D, is in progress, in collaboration with IPP-Greifswald. Short perpendicular wavelength perturbations are assumed, i.e., for wavenumbers $k_{\|} << k_{\perp}$. However, the variation of metric coefficients along the field line is retained, and includes the more rapid variation of **B** (compared to 2-D tokamaks) as one moves along a field line. Thus, besides the drift-wave turbulence familiar in tokamak plasmas, BOUT will also able to investigate the important 3-D coupling between magnetic geometry and turbulence on the meso-scale for 3-D (stellarator) plasmas. An extended version of BOUT will start simulations from magnetic geometry given by VMEC 3-D equilibrium solutions. Radial midplane plasma profiles are taken from fits to experimental data, e.g., probes or Thomson scattering, or from 3-D transport modeling. Since BOUT uses flux surfaces to define its coordinate system, a modified formulation must be devised to include a scrape-off layer with magnetic islands, remnants of islands, and regions of stochastic field lines.

The BOUT code is an example of what is needed to address the critical issues in 3-D boundary plasmas of stability, nonlinear saturation, and turbulent transport, namely: 1) formation, structure, and stability of edge barrier; 2) role of zonal flows and bursty transport; 3) physics of ELMs in the presence of sheared flow, diamagnetic flow, and bootstrap current, and 4), plasma-neutral interactions on turbulence, neoclassical effects, and physics of density limit.

There is substantial evidence from existing devices that edge turbulence transport does not follow the classical small-amplitude diffusive model. Thus, one of the roles of the turbulence simulations is to understand the character of the transport, and to determine if this character changes from 2-D to 3-D systems.

The issue of 3-D transport effects likely occur even in axisymmetric tokamaks, because various ELM types themselves are believed to be associated with a magnetic perturbation of moderately low toroidal mode number (~10 or less). Consequently, the associated plasma transport in the edge region has a substantially 3-D character.

Existing tools: BOUT, b2-Eirene, UEDGE-Eirene

Power handling and hydrogen particle control (exhaust and fueling)

The challenge in this area is to establish the needed control of the heat and particle fluxes, especially for steady-state operation, constrained by the need to exhaust helium ash efficiently in burning-plasma devices. Both field-line structure and cross-field transport must be understood to accomplish such control. Major issues include having acceptably low power fluxes on plasma-facing components, maintaining edge temperatures that lead to reduced core-edge turbulence for improved core confinement, and obtaining SOL plasma profiles that adequately shield the core from hydrogen, impurity, and recycling particles generated at material surfaces. Models to address heat flux issues are divided into the (interrelated) areas of field-line tracing and plasma transport and profiles. Particle control issues introduce the need to describe neutral transport and pellet injection. The prompt loss of energetic particles, due to charge exchange, is also related to these edge particle dynamics characteristics. The status of these areas is discussed in turn.

Heat flux issues: Field-line tracing

The simplest method to estimate heat-flux profiles, widely used at present, is the field-line tracing technique. Field-lines are traced from near the LCMS until they intersect a material wall. Since the charged particles mainly follow the field lines, such calculations give an estimate of the location where the escaping power will be deposited; this method has been used extensively in the W7-X design [3] and is being used for initial NCSX design. As discussed in Sec. 2, it is, thus, essential for use of this technique to have an accurate representation of the magnetic field.

In addition to giving the location of the power flux, field-line tracing provides the connection length from near the LCMS to the wall. At present, simple 'zero-dimensional' methods (‘two-point’ models based on the parallel electron heat conduction equation and plasma pressure balance) are often used to estimate the electron temperature near the LCMS and at the wall for a given escaping heat flux, and the spreading of the heat-flux profiles is modeled by positing a level of field line perpendicular diffusion to mimic anomalous heat diffusion arising from plasma turbulence.

Heat flux calculations: Plasma transport and profile calculations

The next higher level of detail employed in models for macroscopic plasma characteristics in the edge region is to solve a set of plasma fluid transport equations for density, parallel ion velocity, separate electron and ion temperatures, and electrostatic potential, as is done for tokamaks using the B2 [4] and UEDGE [5] codes. We first discuss cases in which some approximation to a flux-surface mesh can be assumed, and then discuss the case in which treatment of stochastic-regions is essential.

The plasma fluid equations can be solved by Monte Carlo or finite-volume techniques. For stellarators, the fluid Monte Carlo approach advances fluid elements of density, momentum, and energy as dictated by the fluid equations. As these equations are nonlinear, one must perform an iteration procedure by updating terms in each of the equations with information (e.g., density, temperature, etc) from previous fluid-element trajectories. This technique has been implemented in the EMC3 code [6] for modeling W7-AS at IPP Garching and the E3D code [7] at Juelich/Greifswald for modeling the TEXTOR ergodic divertor.

The finite-volume technique for solving the 3-D fluid transport equations has been chosen for the BORIS 3-D transport code project at IPP-Greifswald [8]. Substantial collaborative work is underway in the U.S., emphasizing linear-algebra solver technology, 2-D benchmarking, and a fluid neutrals model (discussed below). Presently, BORIS has been used to solve the 3-D energy equations in W7-X and NCSX geometries, and benchmarking with B2 and UEDGE is being done, including testing to understand the numerical error associated with the mesh not being perfectly aligned to the magnetic field, such that the very large anisotropy between the parallel transport and radial transport can be resolved. More work is needed to fully understand the limitation here.

The third method for solving the fluid equations helps address the high transport anisotropy issue by using a finite-difference approach. Here, the field-line (via tracing) can be used as one of the coordinates, but specific volumes do not need to be calculated. The perpendicular derivatives are calculated by finding a set of nearest neighbors. This could be used to determine the transport in a stochastic region where one or a few field-lines may be able to adequately cover a stochastic region in their many transits around the device. This method is being developed at IPP Greifswald, with U.S. consultation.

Neutral transport and profile calculations: fueling and charge-exchange power losses

The interaction between neutral gas and the plasma is a central concern in the edge region, and 3-D geometries present a number of challenges. These issues include: (i)-determining the distribution of non-axisymmetric recycling, gas puff and pellet fueling sources, (ii) determining the SOL plasma parameters in the region between the separatrix and the vacuum vessel wall, (iii) determining the spatial distribution of the separatrix-wall gap, (iv) calculating impurity production and transport mechanisms, and (v) describing 3-D structures such as antennas, diagnostic ports, and baffles for pumping and particle control, etc. Accurate assessment of the effects of each of these requires 3-D codes for simulation of the plasma and neutral particle transport. Approximate assessments of the effects of these factors on core fueling and charge-exchange power losses for NCSX are presently available only from neutral transport calculations in 1-D and 2-D geometry using mostly analytic plasma profiles. It is particularly important for stellarators to quantify the charge-exchange loss in the long-thin cross-section regions where the neutral gas may be able to penetrate more deeply into the hot core region.

As with the discussion of heat flux issues earlier, there is a hierarchy of approaches thatare used. Since the main drawback of coupled neutrals/plasma codes at present lies in the extensive computational effort required, it is essential in both 2-D and 3-D development projects to create simpler neutral fluid models to be used within the coupled code. While not as detailed as Monte Carlo neutral codes, such models 1) are relatively fast and provide the necessary physics behavior (‘predator-prey’ ion-neutral interaction) which is often needed for convergence and 2) can have the added convergence advantage (e.g., in the 3-D BORIS code described earlier) of being implemented as a set of fluid variable, thereby allowing direct nonlinear solution to the coupled plasma/neutral equations instead of using a relaxation method. A model has been benchmarked in 2-D with a similar model in UEDGE as part of U.S. participation in the BORIS project.

In the more detailed approach, existing 3-D Monte Carlo neutrals transport codes (DEGAS-2 [9], EIRENE [10]) require knowledge of background plasma parameters, a description of the plasma facing components (PFC), and models for the interaction of the plasma ions, impurities, and neutral particles with each other and with the PFC. Both the plasma and neutrals codes require grids that are typically based upon the magnetic field geometry. For example, for 2-D tokamak simulations, grids that span the entire computational domain can usually be constructed using flux surfaces generated by a data-constrained plasma equilibrium code such as EFIT. Grids based upon VMEC flux surfaces should suffice in the core plasma region inside the separatrix for NCSX simulations. However, since flux surfaces do not exist in the stochastic SOL region, an alternative prescription for grid generation must be devised for this region. This will require accurate 3-D equilibrium reconstruction in the edge and SOL–a problem that is being pursued but is not yet solved.

The considerable computational power and long turn-around times required for fully 3-D toroidal plasma simulations will limit the utility of these codes for doing detailed data-constrained discharge modeling of the type currently done for tokamaks [11]. For this reason, the parallel development of a 2-D toroidally averaged SOL transport code (or modification of an existing 2-D code) is needed. It would be necessary to extensively benchmark such a code against the corresponding 3-D code results and have a preprocessor that performs the required toroidal averages of the 3-D geometry. The primary utility of such a code is for the determination of anomalous transport coefficients, recycling coefficients, core particle and energy balances, etc. The 2-D results could then be “fine tuned” with 3-D simulations to fit the details of the diagnostic data.

Neutral transport and profile calculations: Pellet injection processes

Pellet injection to fuel toroidal magnetically confined plasmas has been studied for more than 25 years. It has been discovered that the interaction of the pellet with the plasma is more complicated than initially presumed, because the magnetic field curvature and gradients of the toroidal field give rise to polarization drifts in the ablated pellet material as it symmetrizes around the torus. This has been observed in tokamak experiments and is beginning to be be investigated in stellarators. A theory effort to understand the physics of the polarization drift has been started and applied to 1-D tokamak problems [12]. However, pellet deposition is inherently a 3-D effect, both in tokamaks and in stellarators, that is difficult to model self consistently at present, so a further extension of this work to full 3-D geometry is needed to be able to understand and predict the fueling deposition expected from pellet injection. This is especially important for burning plasma devices where the fueling obtained from pellets could be critical for the operation of the device. A strong effort to further enhance our understanding of the pellet interaction problem is needed in the next 3-5 years.

Prompt energy loss via unconfined ion orbits

Hot ions at the plasma edge may have complex orbits, which, for some regions of velocity space, result in a direct loss to the wall or limiter surface. Two processes that can produce such orbits are B-field ripple and radial electric fields. In 3-D systems, ripple effects can be strong, and likewise edge electric fields can be large because of the transition from open to close field lines. Since the loss-cone region of velocity space is depleted by escaping ions, the ultimate loss process can be limited by velocity-space scattering into this region. Because of the complexity of 3-D systems, these prompt loss processes need to be evaluated. Presently, guiding center codes such as ASCOT and field line codes such as MASTOC are used to perform such calculations.

Existing tools: EMC3, E3D, BORIS, DEGAS-2, EIRENE, GOURDON, b2-Eirene, UEDGE-Eirene, ASCOT, MASTOC

Impurity control

Because impurity production at surfaces is a nonlinear function of ion energy and surface temperature, localized deposition and heating that can occur in 3-D systems is especially dangerous. Furthermore, as we proceed to configurations with inherently high particle confinement (accumulation), such as stellarators and advanced tokamak modes, we must better understand how to minimize impurity production. The general issues are to improve the analysis of impurity production peaking factors due to non-axisymmetric plasma-facing structures and plasmas, to understand the role of ELMs for impurity control in tokamaks and to describe potential impurity sources from localized auxiliary heating components, such as antennas and couplers.

Impurity ion transport in 3-D involves characterization of the plasma conditions at the material surface, which is provided by the main-plasma transport theory discussed in Sec. 4, and the transport of the impurity (neutrals and ions) in the plasma. Because the impurities typically are a minority species, this transport is often approached as a pertubative problem, assuming fixed (or 'frozen' main-plasma profiles). Monte Carlo ion particle codes can provide important information when the fixed plasma profiles can be taken directly from measurements, thus, justifying the separation of main plasma and impurity calculations. However, extrapolation of results to next-step machines requires a coupled plasma-impurity calculation, along the lines of those discussed in Sec. 4.

Existing tools: BBQ, DIVIMP, EMC3, MCI, WBC

Coupling to core physics

Edge-plasma conditions are found to impact core confinement in tokamak and stellarator experiments, both through impurity control and also edge temperatures and densities. An important goal is to understand conditions and mechanism wherein topics 2-5 (above) facilitate high core confinement.

This area is very important, but not well developed even for 2-D systems. Major issues include formation and maintenance of edge transport barriers, processes that control the experimental density limits, and the effect of ELM dynamics on core confinement and profiles. For 2-D problems, the SOL code UEDGE and the core transport code CORSICA have been coupled, as have the JET codes EDGE-2D and JETTO, to form the COCONUT system. For 3-D systems, such coupling will be more challenging, probably requiring some reduced descriptions as a beginning.

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APPENDIX A

ANNOUNCEMENT

Dear Colleagues;

The workshop on Future Directions in Theory of 3D Magnetic Confinement Systems will be held in Oak Ridge on January 7-9, 2002 at the Garden Plaza Hotel in Oak Ridge. Registration will begin at 8:30 AM, Monday and the meeting will begin at 9:00 AM. A block of rooms has been reserved under the name 3D Plasma Workshop at a rate of $55.00 per night. This rate is available beginning Saturday, Jan 5th. A registration fee of $60 will be required. Attendees should make their own hotel reservations. The information for the hotel is:

Garden Plaza Hotel
215 S. Illinois Avenue
Oak Ridge, TN 37830
Phone (865) 481-2468
Fax (865) 481-2474

Remember, the emphasis will be on the discussion of scientific issues and opportunities for advances, not on presentation of research results. A projector and materials for traditional viewgraphs will be available, as well as a laptop computer and projector for PowerPoint or PDF format presentations. Or, you can use your own laptop.

I will send separately to those who have registered for the meeting a draft edition of the report document put together by the session organizers. If others not registered wish to receive the draft document they can contact me. I hope that attendees will have a chance to look through the document before the meeting. It should serve as a starting point for the discussion at the meeting and as a basis for the final document.

Sincerely

Don Batchelor

APPENDIX B

Workshop on Future Directions in Theory of 3D Magnetic Confinement Systems - Garden Plaza Hotel, Oak Ridge, TN

Monday morning 1/7
8:30 Registration, coffee and munchies
9:00 Welcome - Lee Riedinger , ORNL Deputy Lab Director for Science and Technology
9:15 Announcements - D. B. Batchelor
9:20 Input from experimental programs - D.T. Anderson
12:00 Lunch

Monday Afternoon
1:00 MHD - A. Reiman, C. C. Hegna
~5:30 Adjourn

Monday Evening
8:00 - 10:00 PM - wine and cheese, coffee and dessert at Don Batchelor's house

Tuesday morning 1/8
9:00 Confinement - A. H. Boozer, H. E. Mynick
12:00 Lunch

Tuesday afternoon
1:00 Modeling and diagnostics - W. A. Houlberg, J. D. Callen
2:00 Wave propagation and RF applications - D. B. Batchelor, H. Weitzner
2:30 Optimization - A. S. Ware, N. Pomphrey
4:00 Edge physics modeling - J. T. Hogan*, T. D. Rognlien
~5:30 Adjourn

Wednesday morning 1/9

9:00 Wrap-up session - D. B. Batchelor

Wednesday noon - Adjourn

APPENDIX C

LIST OF ATTENDEES

Donald Batchelor	ORNL
Larry Baylor	ORNL
Lee Berry	ORNL
Mark Carter	ORNL
Steve Hirshman	ORNL
John Hogan	ORNL
Wayne Houlberg	ORNL
Fred Jaeger	ORNL
Ed Lazarus	ORNL
J. F. Lyon	ORNL
Stan Milora	ORNL
Peter Mioduszewski	ORNL
Larry Owen	ORNL
Jim Rome	ORNL
Don Spong	ORNL
Dennis Strickler	ORNL
David Anderson	University of Wisconsin
A. Bhatttarcharjee	University of Iowa
Allen Boozer	Columbia University
Bas Braams	New York University
Jim Callen	University of Wisconsin
Steve Eckstrand	US DOE
Guoyong Fu	Princeton Plasma Physics Laboratory
Paul Garabedian	New York University
James Hanson	Auburn University
Chris Hegna	University of Wisconsin
Nick Hitchon	University of Wisconsin
Stuart Hudson	Princeton Plasma Physics Laboratory
Steve Knowlton	Auburn University
Arnold Kritz	US DOE/Lehigh University
Richard Moyer	General Atomics
Donald Monticello	Princeton Plasma Physics Laboratory
Harry Mynick	Princeton Plasma Physics Laboratory
Neil Pomphrey	Princeton Plasma Physics Laboratory
Xu Xue Qiao	Lawrence Livermore National Laboratory
Allan Reiman	Princeton Plasma Physics Laboratory
Kerchung Shaing	University of Wisconsin
Henry Strauss	New York University
Joseph Talmadge	University of Wisconsin
Andrew Ware	University of Montana
Harold Weitzner	New York University