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Toward Improved Stellarators Future Directions for U.S. Research

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TOWARD IMPROVED STELLARATORS: FUTURE DIRECTIONS FOR U.S. RESEARCH

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Steady state high-performance, transients, and efficient current-drive are major challenges for magnetic fusion energy (MFE). Stellarators offer promising solutions, and indeed there is a world program, including two large experiments developing stellarators, Japan's Large Helical Device (LHD) and Germany's Wendelstein 7-X (W7 X), both designed around 1990. Advances in stellarator physics and engineering in the years since these experiments were designed have the potential to substantially improve future stellarator-based fusion systems. These advances, which define the priorities for stellarator research in the U.S., are in the following areas: quasi-symmetric configurations, turbulent transport optimization, divertors and plasma-material interactions, impurity control, energetic particles, and engineering optimization. Currently, U.S. stellarator experimental research is carried out through a major collaboration with Germany's Wendelstein 7-X program and on smaller domestic experiments. A theory and design activity is proposed as a next step toward developing and evaluating new designs that can become the basis for new experimental facilities that would then be constructed. It is argued that, with timely action, new U.S. experiments could begin to come on line in the 2020s and could impact the direction of fusion development in the ITER era and decisions on next steps beyond ITER.

Keywords: stellarators; strategy; Wendelstein 7-X; innovation

I. INTRODUCTION: THE NEED

Steady state high-performance, transients, and efficient current-drive are major challenges for magnetic fusion energy (MFE). Stellarators offer promising solutions, and indeed there is a world program, including two large experiments, Japan's Large Helical Device (LHD) [1] and Germany's Wendelstein 7-X (W7-X) [2, 3], developing stellarators. With the recent startup of W7-X operations, experimental investigations on the world's most advanced stellarator have begun. Moreover, advances in understanding and design capability since LHD and W7-X were designed (ca. 1990) offer opportunities to make quantum improvements in the design of future stellarator reactors. A U.S. initiative that takes advantage of experimental opportunities afforded by W7-X and of theoretical advances since W7-X was designed can move fusion forward.

II. ADVANCES THAT CAN IMPROVE STELLARATORS

II.A. Quasi-aymmetry

The design of stellarators was revolutionized in the late 1980's and early 1990's by the development of device designs in which the energetic particle drift trajectories stay sufficiently close to magnetic surfaces as to confine energetic alphas. The W7-X is an example of such a "drift-optimized" design (Figure 1). In W7-X, self-driven currents such as the bootstrap current and Pfirsch-Schlüter currents have been designed to be small. This has the virtue that, for a given beta, the changes in the equilibrium field produced by pressure driven currents are small relative to those in other configurations, thus reducing deleterious effects that are associated with increased beta: instability and breakup of magnetic surfaces. However, the W7-X design features a relatively high plasma aspect ratio (11) and a large number of non-planar coils (50).

Quasi-symmetry is an underlying symmetry property of a 3D magnetic configuration that leads to drift trajectories that approximate those of their truly symmetric counterparts. In a quasi-helical (QH) configuration, of which the HSX device at the University of Wisconsin [4] is an example, the drift trajectories approximate those of a straight helical stellarator in the limit of large aspect ratio. In a quasi-axisymmetric (QA) device, the drifts approximate those of a tokamak. Both QH and QA designs rely primarily on externally-generated 3D magnetic fields, and hence do not require external current drive, to form an inherently steady-state magnetic configuration. They do, however, have non-zero self-generated currents: the bootstrap current in QA stellarators is comparable to that in a tokamak, and it is in a direction which adds to the externally generated rotational transform while the bootstrap current in a QH stellarator opposes it. The NCSX machine design [5] was based on a quasi-symmetric configuration. In both cases, the net plasma current that can be restricted to whatever level is required to avoid major runaway-electron risks and, as externally control rather than self-plasma configurations, both have robust positional stability, which prevents tokamak-like disruptions. Quasi-axisymmetric stellarators can accommodate sheared toroidal flows, which are advantageous for turbulence stabilization, and can have aspect ratios approaching those of tokamaks. Examples of QH and QA experiment designs are shown in Figure 2.

II.B. Turbulent transport optimization

While drift-optimization strategies like quasi-symmetry have made it possible to design stellarators for reduced energetic particle and neoclassical transport losses, it remains the case that energy losses are dominated by turbulence-driven transport. For this reason, understanding and if possible reducing turbulent transport is an issue of considerable importance.

The gyrokinetic (GK) codes GENE[6,7], GKV[8,9], and GTC[10] can simulate the microinstabilities that drive transport in the 3D toroidal equilibria of stellarators. To date, GENE has been the principal workhorse for performing nonlinear simulations in 3D geometries. One fruitful line of application with these has been to investigate the sensitivity of plasma turbulence to the shape of the device, identifying key geometric parameters (such as the radial curvature or local shear) that affect the transport levels [11, 12, 13].

The GK codes in conjunction with the stellarator optimization code STELLOPT [14] provide the capability to design stellarators with substantially reduced levels of turbulent transport, often without degrading the neoclassical transport, resulting in "turbulence-optimized" designs [15, 16, 17]. The heat flux Q_{gk} from nonlinear GK runs has been reduced by factors of 2-3 based on these optimizations, an amount comparable to the improvement in going from L-mode to H-mode in tokamaks. Improvements in the theoretical basis and methodology are planned and may lead to further improvements. In addition, the changes in turbulence with plasma shape, which is the basis for turbulence optimization, needs to be tested experimentally.

II.C. Divertors and Plasma-Material Interactions

II.C.1 Stellarator Divertors

The Wendelstein 7-X experiment will provide the opportunity to study its island divertor configuration. The island divertor relies on a chain of islands at the edge of the plasma and a system of divertor targets and baffles arranged to intersect the islands. Because the islands form where the rotational transform profile passes through a rational value (in the case of W-7, unity), an island divertor is a resonant configuration, which makes it sensitive to small departures, such as bootstrap currents and error fields, from the as-designed stellarator configuration (Figure 3). The inferface between the divertor islands and the structures has some similarities with the tokamak poloidal divertor, however the boundary between closed and open field lines in a 3D system is not sharp; in general, all configurations have a stochastic layer at the plasma edge. Island divertor systems feature significantly longer connection lengths than poloidal divertors in tokamaks [18], allowing for increased parallel temperature gradients and a larger perpendicular to parallel transport ratio. This beneficial feature may help to reduce the heat flux peaking on the divertor targets and to retain impurities in the edge islands, reducing core contamination.

Other divertor configurations are possible in 3D systems. In particular, quasi-symmetric configurations are more likely to have a non-resonant divertor that occurs naturally in regions of high flux expansion. The configuration of target and baffle structures, including the lack of toroidal continuity, can be similar to the island divertor.

II.C.2 Plasma-Material Interactions (PMI) in Stellarators

Stellarators face most of the same PMI issues as tokamaks, but key differences necessarily affect the approach to these problems. In three dimensions, new phenomena occur, like the formation of hot spots at the surface and the modification of the wetted profile as a function of the plasma current conditions. As a consequence of the three dimensional magnetic topology, the parallel and perpendicular transport has more complex patterns dominated by the different connectivity of adjacent magnetic flux tubes. In tokamaks there is a direct relation between the material and plasma performance; for example, the use of lithium in NSTX has shown that transient behavior, such as ELMs, can be affected. However, the influence of materials on the edge turbulence and transport in stellarators has not been extensively studied.

II.C.3 Stellarator Divertor and PMI Research Tasks

Existing facilities can be used to explore basic divertor concepts, edge transport physics, and innovative materials within their parameter range. Many of the key aspects of a reactor-relevant divertor design will be tested on W7-X, e.g., integrated core-edge solutions, divertor heat flux width, ELMs and H-mode, access to the high heat flux regime and detached operation. However, new facilities will be needed to investigate non-resonant divertors and make comparisons with the island divertor.

The leading code designed to simulate stellarator edge physics issues at this time, EMC3-EIRENE [19] will receive an important validation test via comparisons with W7-X measurements. Many aspects of stellarator divertor physics have only been explored for limited parameters and configurations and require additional research. These include access to the high recycling regime, divertor flux widths, and detachment. Development of a new code incorporating additional physics would play an important role in stellarator modeling and optimization. In order to include a divertor design into an a stellarator optimization code such as STELLOPT, cost functions related to divertor and edge physics must be developed.

With respect to PMI, there are many open questions. How are the edge plasma flow, transport, and scrape off layer (SOL) profiles affected by the interactions with material surfaces? Are impurity screening and plasma detachment profiles important? Also, there are unique opportunities to investigate issues important to both tokamaks and stellarators, by taking advantage of their capability for long-pulse operation (e.g., up to 30 min. in W7-X). While the high power machines W7-X and LHD currently use carbon-based materials, chosen in part because it is well characterized through tokamak studies, replacement with a more reactor-relevant material is already under consideration by W7-X. In conjunction with linear machines and tokamaks, long-pulse stellarators provide a means to investigate issues of plasma-facing material migration and property changes in fusion devices.

II.D. Impurity Control

Impurity control is an important concern for all magnetic fusion concepts. In the core, impurities radiate energy and would dilute the fusion fuel in a reactor. A particularly important impurity in a reactor will be the helium ash, which must be extracted somehow. On the other hand, impurities confined to the plasma edge can be beneficial, as impurity radiation reduces the peak heat fluxes on the divertor.

A serious impurity control issue for stellarators is the existence of a robust inward neoclassical impurity pinch. The reason is that, due to the lack of inherent ambipolarity in 3D systems, stellarators typically operate in an ion-root regime in which the radial electric field points inward , implying impurity accumulation. One solution may be to operate in an electron-root regime but access to this regime typically requires strong electron heating and/or very low collisionality, so it is not clear this approach is relevant to a reactor where the ions must be hot and high density is desired. Since stellarators tend to operate at higher densities than comparable tokamaks, and therefore lower temperatures, impurities are more likely to have bound electrons that efficiently radiate.

The situation may fundamentally different in the case of perfect quasisymmetry. In true axisymmetry the plasma is able to rotate; the inertial forces are negligible in the rotating frame, the electric field is transformed away, and fluxes cannot depend on the radial electric field. Thus, in symmetric geometry the main impurity pinch mechanism is absent. While this favorable property applies to *perfect* quasisymmetry, it is unclear if it can be achieved in realistic designs, which are *imperfectly* quasisymmetric.

Encouragingly, two low-impurity regimes have been observed experimentally, though neither is well understood theoretically. One regime is the high-density H-mode (HDH) observed in W7-AS [20]. This regime is accessed with neutral beam heating and rapid initial gas puffing such that n_e exceeds a threshold around 1-2 × 10^{20} m^{-3}. Radial profiles of impurity radiation are hollow, steady in time, and generally much lower than for non-HDH discharges, as shown in figure 3 of [20]. In HDH mode, impurity confinement times measured by laser blow-off are strongly reduced compared to the expected scaling with density in "normal" discharges, as seen in figure XX.a above.

The other noteworthy regime is the "impurity hole" in LHD [21, 22, 23], which in contrast to HDH mode is found at low rather than high electron density. The regime is associated with neutral beam heating and a peaked ion temperature profile. Core impurity radiation is seen to decrease in time, so there is evidently outward impurity convection in the core, despite measurements of a negative core radial electric field and predictions of inward turbulent impurity transport [24]. Multiple impurity species are seen to have hollow profiles, with the hollowness increasing with atomic number Z.

In terms of research needs, the transition from a stellarator-like impurity pinch to tokamak-like impurity screening with quasisymmetry needs to be investigated theoretically. Studies in quasi-symmetric experiments can illuminate how close to quasisymmetry one should strive to be. While not quasi-symmetric, the W7-X experiment offers opportunities to study impurity transport in a large 3D system and, with appropriate modeling support, improve understanding and impurity control methods.

II.E. Energetic particle confinement

Energetic particle (EP) confinement is an important issue for stellarators and remains a significant driver for 3D optimization strategies. Energetic particle populations in stellarators can arise from neutral beam and ICRF heating techniques and, eventually, from D-T fusion reactions in the form of alpha particles. Lost energetic particles can impact the first wall in concentrated areas resulting in localized high heat-flux “hot spots,” and EP populations can drive instabilities through various resonant wave-particle interactions. In order to predict stellarator performance and choose optimization strategies, it will be important to understand EP transport both through direct classical orbit loss and from EP-driven instabilities. Since existing stellarators have not achieved the levels of EP confinement optimization required in a reactor, addressing this issue will be critical to further development of the stellarator concept

Improvement of EP classical orbit confinement in 3D configurations has been addressed theoretically using a variety of optimization approaches [25, 26, 27]. Losses of fusion-born alpha particles in reactor-sized stellarators can be reduced from 10-40% levels in un-optimized systems to a few percent in well optimized systems. Fast ion transport driven by EP instabilities can lead to large of beam ion losses in tokamak experiments, but the issue has not been studied extensively for stellarators. Deuterium operation and neutral beam injection in both LHD and W7-X will provide new opportunities to study energetic particle physics experimentally.

II.F. Engineering Optimization

While stellarator research traditionally focuses primarily on physics issues, engineering considerations will necessarily require increased attention in the design of future stellarators. Earlier concerns about fabrication feasibility have been largely resolved through the successful construction and operation of two large superconducting stellarators– LHD with its continuous helical coil and W7-X with its array of 50 non-planar and 20 planar coils. By their very nature, stellarators start with important engineering benefits: the capability to maintain a high performance plasmas in steady state, to avoid disruptions and other damaging transients, and to minimize recirculating power. However, plasma exhaust, tritium breeding, shielding of superconducting magnets, rapid removal and replacement of limited-life in-vessel systems, and minimization of overall components count are issues that require further efforts to address.

Nowhere is the tension between stellarator physics and engineering attractiveness more pronounced than in the design of coils. As discussed in Section II.B, there have been great advances in the tools to optimize stellarator plasma configurations for attractive physics properties such as reductions in energetic particle orbit losses, in neoclassical transport, and now even in turbulence-driven transport. The successful manufacture of coils for NCSX and the completed construction of W7-X demonstrate that coils for physics-optimized designs can be realized and they validate the coil optimization tools used in the design of those machines.

Those tools relied on a Fourier representation of the coil winding center trajectory on a prescribed winding surface. Further improvements in coil engineering require that the designer have more freedom and more control over the coil trajectory, without compromising plasma properties. In order to provide these capabilities a new code, COILOPT++, was developed with the following features 1) a spline-based representation of the coils, 2) the ability to target coil penalties and freeze coil geometry for individual coils, 3) freedom to straighten the outer legs of non-planar ("modular") coils over asymmetric distances above and below the outboard mid-plane, 4) capability to include multiple coil topologies, including TF-like, PF-like, and saddles, and finally, 5) the ability to import coil winding surface geometry from a CAD model. Like its Fourier-based predecessors, the COILOPT++ code finds coil geometries and currents that satisfy the magnetic boundary conditions, i.e. B-normal = 0, for a prescribed, physics-optimized plasma configuration.

The new tool has been applied to the problem of improving maintenance access to in-vessel systems. The issue was highlighted as an outstanding R&D need in the ARIES-CS stellarator power plant study [28]. The stellarator configuration for that study was based on the physics-optimized NCSX design, in which the out-of-plane toroidal excursions of the modular coils on the outboard side left only small openings through the coil "cage," and thereby placed a limit on the size of in-vessel components needing to be removed and installed in scheduled maintenance operations. However, availability considerations favor transport of a small number of large in-vessel sectors through widely-spaced coils. Therefore, the design of a modified ARIES-CS machine model based on a sector-maintenance philosophy (a significant perturbation) was undertaken. Using the new coil code and working closely with the designer, a new coil design compatible with large-sector maintenance was developed. The resultant machine design is shown in Figure 4. In the process, the ARIES-CS aspect ratio (A) = 4.5 plasma configuration was replaced by a A = 6 configuration. To the extent that the increased aspect ratio would lead to a larger machine, it can be thought of, qualitatively, as an investment in plant equipment in order to realize higher availability and lower operating costs. Quantifying this tradeoff is beyond the scope of this initial study.

The ideas described in this section are the first attempt to include constraints on the physical location of the coils for an optimized stellarator to enable better engineering and maintenance features of the final device. Additional physical constraints, for example associated with coil forces, peak magnetic fields, and divertors, can be added. While the A = 6 plasma configuration is a stability-optimized quasi-axisymmetric design, more thorough of analysis of the free-boundary equilibria provided by this coil design is needed. Moreover, the possibility of reducing the aspect ratio needs to be studied. Eventually the goal of this coil optimization using geometric constraints is to include it in a complete optimization activity, which includes a comprehensive set of physics and engineering attractiveness metrics.

III. U.S. STELLARATOR RESEARCH ACTIVITIES

Stellarator research in the U.S. currently has three main components: experimental research on Germany's Wendelstein 7-X stellarator in collaboration with the Max Planck Institute for Plasma Physics (IPP), small experiments at the University of Wisconsin and Auburn University, and theory and computation. We discuss the first two in this section, and theory and computation in the next. The current activities and future directions for these programs are defined by the stellarator concept advancement opportunities described in Section II. In addition, these opportunities together with the needs for improvement in the vision of future magnetic fusion energy systems make a strong case for a major U.S. stellarator initiative including new experiments.

III.A. Collaboration on W7-X

The Wendelstein 7-X (W7-X) stellarator, which began operating in late 2015, is a large, modern facility based on an optimized magnetic configuration design and superconducting magnet technology. It has a mission to validate its drift-optimized physics design strategy for reducing energetic particle losses and neoclassical transport. The device is optimized in such a way that as beta increases, bootstrap and Pfirsch-Schlüter currents are not generated and there is no Shafranov shift, making the configuration robust to beta changes. Currently the facility is configured for pulsed operation, but it will be reconfigured 4-5 years from now with an actively-cooled divertor system to support its mission to demonstrate high-performance plasma operation for pulse lengths up to 30 minutes.

The U.S. has a significant national team collaboration with W7-X, already involving seven U.S. institutions and significant investments in equipment and analysis capabilities. The U.S. team played key roles in the first W7-X operating campaign, known as OP1.1, which ended in March 2016, leading important physics experiments and producing key data and results. U.S. equipment contributions to the project to date include: 1) a complete set of field error control coils (“trim coils”) and their power supplies; 2) a diagnostic for measuring plasma ion temperature and velocity profiles; 3) diagnostics for measuring infrared and visible radiation from plasma-facing surfaces; and 4) diagnostics for measuring light emission from partially ionized atoms in the plasma edge region. Key U.S. research contributions during OP1.1 included studies of: 1) error fields and the scaling of island width with externally imposed $n = 1$ field perturbation [29], 2) control of neutral particle fueling and exhaust [30], and 3) measurements of ion temperature profiles, radial electric field, and transport [31].

The next campaign, OP1.2, is scheduled to begin in mid-2017. In preparation, IPP will complete the installation of the divertor system and additional heating and diagnostic systems. Additional U.S. research equipment items now in preparation include an instrumented divertor “scraper” that will be used to test our models of edge transport in diverted plasmas. U.S. imaging equipment is being reconfigured to measure heat fluxes to the scraper and the main divertor targets, information needed to evaluate performance of the scraper as a protective component. Finally, U.S. diagnostics to measure plasma turbulence will be installed for OP1.2. The divertor and scraper components will rely on inertial cooling between pulses and carbon-based plasma facing armor materials in OP1.2. These choices provide a robust environment in which to optimize diverted high-performance plasma operating scenarios for this new device, prior to a transition to actively-cooled systems after OP1.2. A future conversion from carbon to more reactor relevant plasma-facing materials such as tungsten is in the discussion stage.

The U.S. team carries out its W7-X research in collaboration with IPP and other W7-X partners in keeping with the project’s “one team” philosophy. Its research agenda for OP1.2 has been planned to as to advance U.S. concept improvement goals and centers around five major themes.

1. Divertor characterization and control. The W7-X island divertor configuration offers the best near-term opportunity to advance the physics of 3D divertors. U.S. and IPP scientists are collaborating in the application of state-of-the-art edge transport modeling tools, e.g. the EMC3-EIRENE code, to design experiments and make predictions by simulating heat loading of plasma facing components and impurity transport in the core plasma during the first W7-X operating campaign. Research in this area in OP1.2 will enable evaluation of the divertor flux widths and the potential advantages of long connection lengths that characterize the W7-X island divertor. More generally, we can improve our understanding and models of edge plasma physics and impurity transport in diverted stellarator plasmas. In the longer term, we will collaborate in the extension of these studies to steady-state conditions.
2. Core turbulence and transport. New U.S. turbulence diagnostics will be installed on W7-X for OP1.2. A phase contrast imaging (PCI) diagnostic will measure turbulent density fluctuations in the core of W7-X spanning wavenumbers of 1.0 to 20 cm^{-1}, and frequencies from 10 kHz to ~2 MHz, a expected to include important transport-driving turbulent modes. A fast (~1 ms) 2D imaging camera diagnostic will measure plasma edge turbulence. Gyrokinetic codes such as GENE have already been used to simulate the structure of key turbulent fluctuations [16]. By validating the key simulation codes used in turbulence optimization of 3D plasmas, these W7-X studies can be an important step in the overall improvement of stellarators.

3. Error fields and island physics. Magnetic islands are an important distinguishing feature of 3D equilibria. Island formation is believed to play a role in setting stellarator beta limits and, in the case of W7-X, the divertor relies on the existence of a robust static island chain at the plasma boundary. However, the divertor islands can shift during start-up transients and affect the heat load distributions on divertor the divertor targets. A goal for OP1.2 is to determine whether a system of divertor scrapers will be useful to protect the weakly cooled edges of the divertor targets in steady state. Finally, the U.S. trim coils provide a means to compensate for intrinsic error fields and to correct for heat-load non-uniformities among the ten W7-X divertor chambers. Islands have already been investigated in vacuum with electron beam and field-line following codes. With plasma, 3D MHD equilibrium tools that allow for magnetic islands, such as the HINT, PIES, SIESTA or SPEC codes, are available to support the experimental research on W7-X.
4. Equilibrium reconstruction. The ability to analytically reconstruct the plasma equilibrium from experimental data is a basic need. Many analysis tasks, such as inversion of chordal diagnostic signals, power balance analysis, stability analysis, and edge plasma analysis rely on an accurate numerical model of the MHD equilibrium state of the plasma. Mature reconstruction tools for tokamaks have existed for some time; for stellarators the STELLOPT and V3FIT codes have been developed. These codes use the VMEC code to solve for the equilibrium that provides the best fit to available data, such as coil currents, magnetic diagnostic signals, and profile diagnostics. Both will be used on W7-X.
5. Energetic particle confinement. Neutral beam injection heating on W7-X will provide the opportunity to investigate the important issue of energetic ion confinement. Several simulation codes, in various stages of development, will enable modeling and experiment

design to go forward in OP1.2. Fast ion loss detection techniques used on TFTR and JET will be adapted for W7-X.

III.B. Domestic Experiments

Currently there are four experimental stellarator facilities in the United States. All are in the so-called "concept exploration" (CE) class– machines that are smaller in scale and more focused in research scope when compared to the large integrated LHD and W7-X research projects. All are investigating topics of importance for the optimization of the stellarator concept. The Helically Symmetric Experiment (HSX) at the University of Wisconsin is the world's only operating quasi-symmetric stellarator. Its quasi-helical configuration has been demonstrated to reduce neoclassical losses and improve confinement. The Compact Toroidal Hybrid (CTH) focuses on the physics of current-carrying stellarators, including the effects of 3D magnetic fields and externally-generated rotational transform on instabilities and disruptions. The CNT stellarator at Columbia University investigates high beta equilibrium and stability, relying on its small-volume, low aspect ratio design to make high beta conditions accessible with low power heating. The HIDRA facility at the University of Illinois uses a classical stellarator device (the former WEGA stellarator, recently acquired from Max Planck Institute for Plasma Physics) to create a confined steady-state plasma for plasma-material interaction research and plasma-facing component development.

Experiments at the CE scale, including the above, will continue to be useful for addressing fundamental physics and technology issues of importance to W7-X and future U.S. experiments. Divertors, novel configurations, and material and technology innovations are leading examples of research areas where CE experiments could make important advances in stellarators.

IV. AN INITIATIVE TO IMPROVE STELLARATORS

There exist both the need and the opportunity for a research initiative aimed at making a quantum advance in the vision of future fusion energy systems. The ITER project is on a course to demonstrate control of a burning plasma for a few minutes, but a clear path to a practical steady-state fusion system does not currently exist. Disruptions, ELMs, and the recirculating power requirements for current drive are risks to the success of MFE which compel increases in stellarator research. Today's leading stellarator programs are addressing these risks, but are not designed to take advantage of the advances in physics and concept design, such as those described in Section II, of the past twenty-five years. These advances provide the opportunity for major improvements in the concept through an initiative to build and operate new machines designed to test and evaluate their potential.

IV.A. Theory, Computation, and Design

Nearly all plasma physics properties in stellarators are influenced by the properties of the three-dimensional magnetic fields; there is a vast design space available in which to search for optimum solutions. Because stellarators do not require large plasma currents, they are less susceptible to the complex self-organization physics that permeates other magnetic confinement concepts, so stellarator design lends itself to an optimization strategy solves for coil geometries and currents that provide a best fit, subject to engineering constraints, to targeted plasma properties. The optimization tools for stellarator design comprise a set of codes that calculate MHD equilibrium and the physics properties of that equilibrium (such as transport, stability, and magnetic surface quality), and then navigate the parameter space defining the magnetic configuration to optimize the calculated properties towards the desired targets.

The physics analysis codes at the heart of this process embody our understanding of stellarator physics and its dependence on the magnetic configuration. Design improvements generally result from the development of new tools that, either directly or by informing reduced models, can be added to the optimization suite to target new physics properties and further improve on possible stellarator designs. In addition, engineering properties are now being incorporated to a greater extent than in the past (see Section II.F). So, the first step toward a new experimental stellarator program is an initiative in theory and design, the aim of which is to develop new designs that exploit recent advances in physics understanding, particularly in the areas of 1) turbulent transport, 2) edge physics and divertors, 3) impurity transport, 4) energetic particle confinement, and 5) engineering feasibility. Research in these areas has progressed to the point where the necessary tool developments can now be carried out in a project-structured activity focused on developing and evaluating new designs that can become the basis for new experimental facilities. This activity could be organized and launched immediately.

IV.B. New experiments

The potential for scientific advances to result in marked concept improvements requires that they be tested in new, purpose-built experimental facilities. In the current era, scientific decisions on the direction of fusion development are taken internationally, so a U.S. program to develop new stellarator concepts must be planned so as to influence international thinking on steps beyond ITER. A coherent national program including multiple domestic experiments and continued strong collaboration on overseas stellarators, supported by a vigorous theory and simulation program, is necessary. Convincing results require a program on a scale comparable to other impactful programs which, in the coming decades, will certainly include W7-X, now operating, and JT60-SA, which is to start operating in 2019. Timely action on the theory and design initiative described in Section IV.B would maintain the possibility of bringing new U.S. stellarators on-line in the 2020s and having an impact on fusion directions in the 2030s.

Though specific facility plans and designs have not yet been developed, the minimum scope of the needed stellarator portfolio can be foreseen on the basis of experience with tokamak development. It would include a new mid-scale experiment, a new flagship experiment, and continuation of research in existing or new experiments at the concept exploration scale.

1. Mid-scale experiment. The next step in quasi-symmetric stellarator development would extend the research on HSX into a more fusion relevant configuration and plasma regime. The configuration would necessarily include a divertor while the core configuration could be QA or QH. While HSX has demonstrated the benefits of quasi-symmetry in low-density, hot-electron plasmas, the next step will require hot thermal ions and high density. Research aims would include the investigation of a "non-resonant" divertor configuration and its control requirements, for comparison with the W7-X island divertor. Plasma-material interactions and impurity control in such a configuration would be studied. Transport in reactor-relevant ion collisionality regimes would be studied, including the characterization and role of turbulence in plasma transport. Studies of high β and long-pulse issues, which drive costly facility requirements, would not be a focus of the mid-scale experiment.
2. Flagship experiment. The term "flagship" refers to an experiment that would likely compare in scale and scope to machines like W7-X, LHD, and JT-60. Such a device is needed to provide convincing integrated tests of concept advances under reactor-relevant conditions. Critical scientific questions that motivate such an experiment include:

- Quasi-symmetry at high β: How much externally generated transform is needed to prevent disruptions? How much internally generated transform can be tolerated?
- Energetic particles: Can we acquire a predictive understanding of energetic particle losses in QS adequate for designing reactor first wall solutions?
- Long-pulse / steady-state impurity control: Can impurity sources and transport be understood and controlled well enough to maintain high core performance? Is quasi-symmetry beneficial?
- Turbulence / transport: Can we experimentally confirm the predicted turbulence characteristics and test transport reduction through shaping?
- Divertors and edge: Can a QS-compatible (non-resonant) divertor be compatible with both material limits and high core plasma performance.

The flagship experiment design would certainly include an optimized, quasi-symmetric core and divertor configuration and engineering-optimized coils, and be capable of long pulse operation. Critical choices that require more rigorous design and analysis include the choice between QA and QH symmetry, the choice of materials for magnet conductors and plasma-facing components, and the mix of plasma heating technologies.

3. Concept exploration. A continuing role is foreseen for CE experiments targeting fundamentals of key emerging issues important to research programs on the larger domestic and overseas experiments. Examples of possible CE missions include:
   - Physics model validation for stellarator divertor designs, e.g. through comparison with scrape-off layer heat flux spreading in long connection length regimes.
   - Physics of current-carrying stellarators, including equilibrium diagnosis and effects of 3D fields on stability and disruptions.

- Plasma-material interactions, for example exploration of liquid-metal plasma-facing surfaces.

V. SUMMARY

Although there is as yet no clear path to a practical steady-state fusion system, stellarator research greatly improves the chances for success. Advances in stellarator physics and engineering in the years since the large stellarators LHD and W7-X were designed have the potential to make quantum improvements in the design of stellarators. The current aim of the U.S. stellarator program is to continue to advance the science and technology of stellarators through theory and experimental research on domestic experiments and, in a major collaboration with Max Planck Institute for Plasma Physics, on Wendelstein 7-X. The envisioned next step is a new initiative to develop improved stellarator concepts, taking advantage of recent progress in stellarator physics and engineering, on a scope and time scale to impact the direction of fusion development in the ITER era and decisions on next steps beyond ITER. The initiative would begin with a theory and concept optimization activity focused on developing and evaluating new designs that can become the basis for new experimental facilities. New experiments, which would be essential for convincing assessment of the potential of new concepts would be built and would begin to come on-line in the 2020s.

VI. ACKNOWLEDGMENTS

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[31] N. Pablant, *et al.,* " Core Radial Electric Field and Transport in Wendelstein 7-X Plasmas," invited paper to be presented at the 58th Annual Meeting, American Physical Society, Division of Plasma Physics, 31 October to 4 November 2016, San Jose, CA.

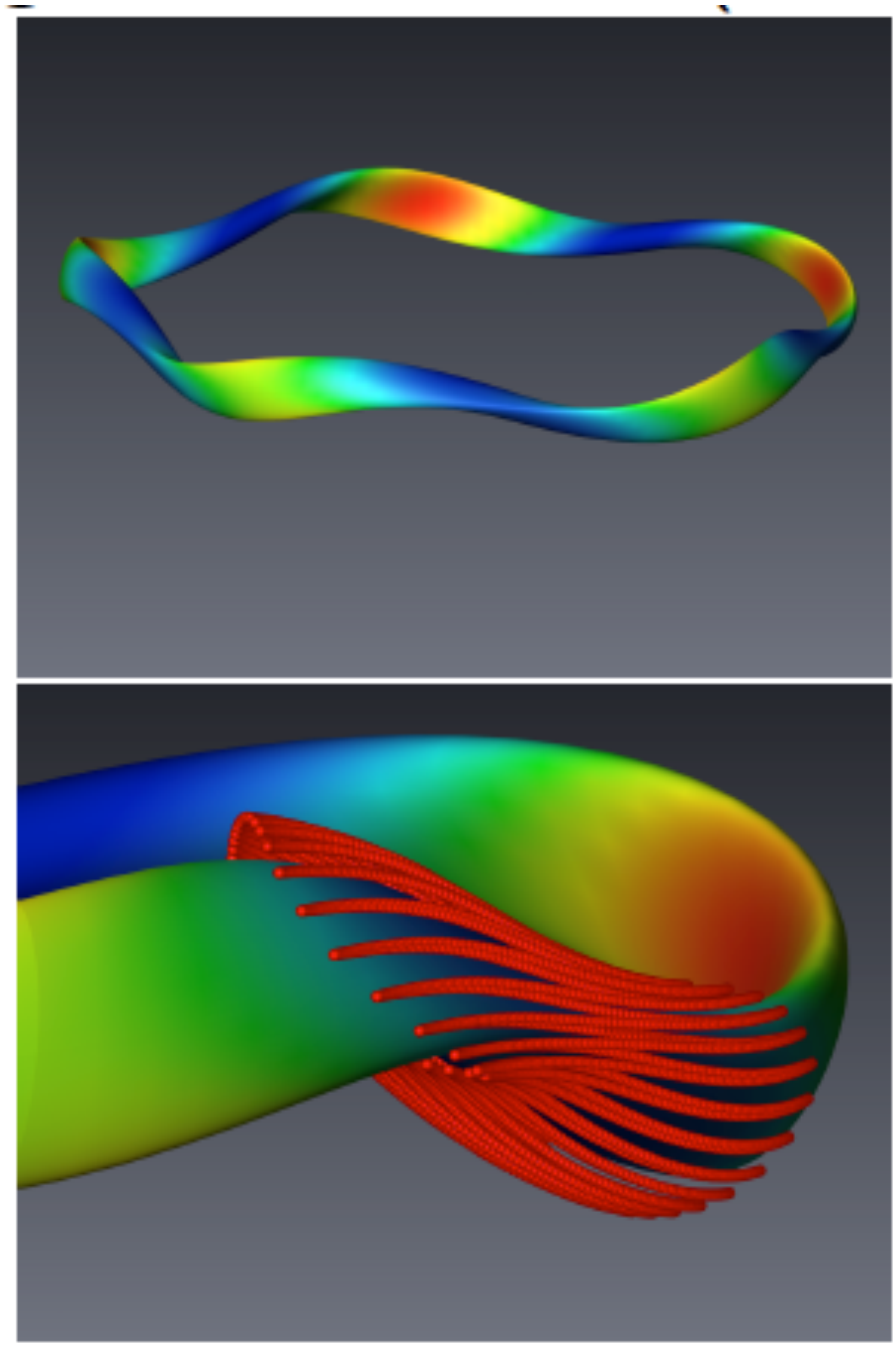

Figure 1. Wendelstein 7-X plasma configuration showing mod-B contours (top frame) and an example trapped particle trajectory. (Images courtesy of J. Proll, Max Planck Institute for Plasma Physics.

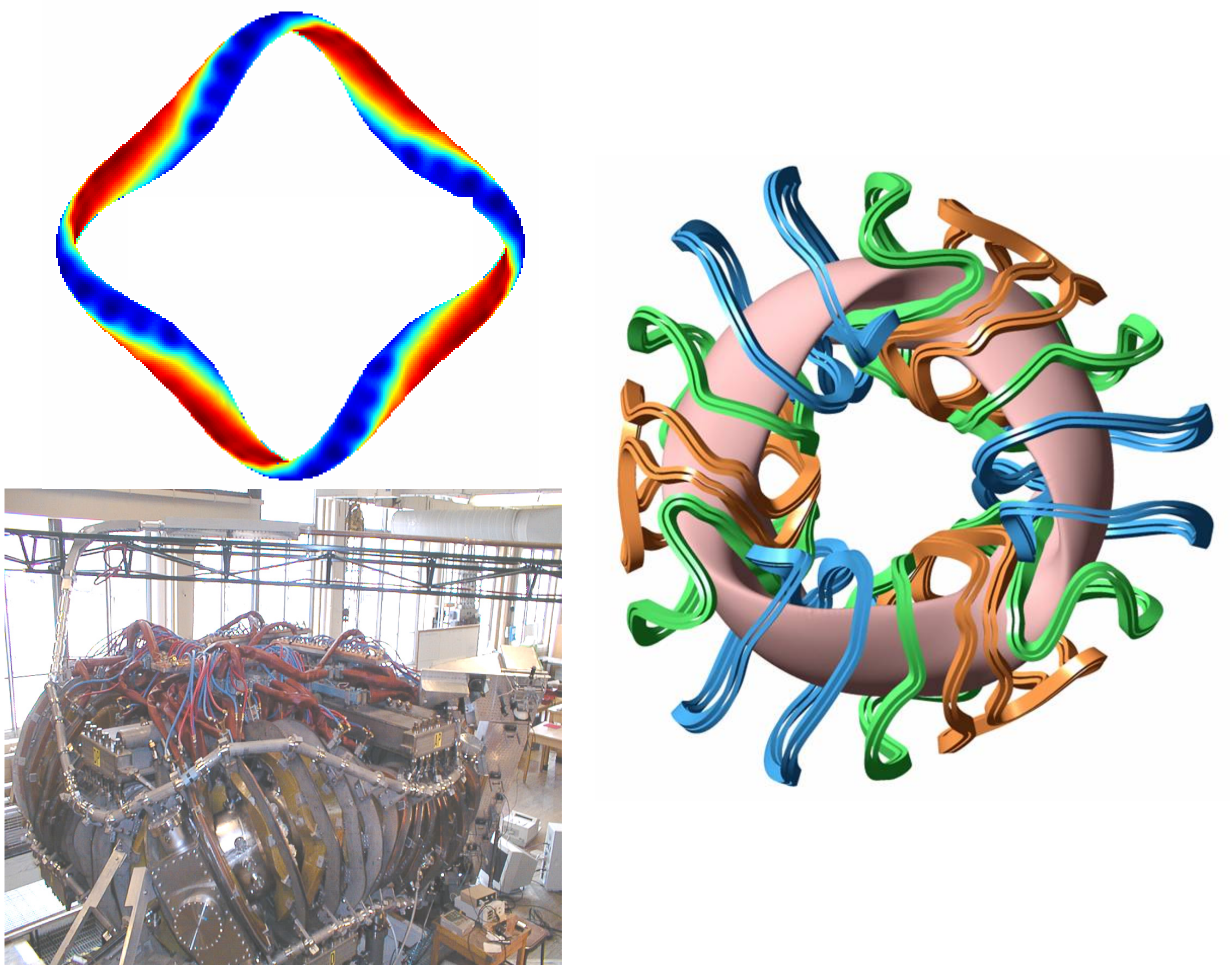

Figure 2. HSX quasi-helical plasma showing mod-B contours (upper left) and machine (lower left). NCSX quasi-axisymmetric plasma and non-planar modular coils design (right). NCSX planar coils not shown.

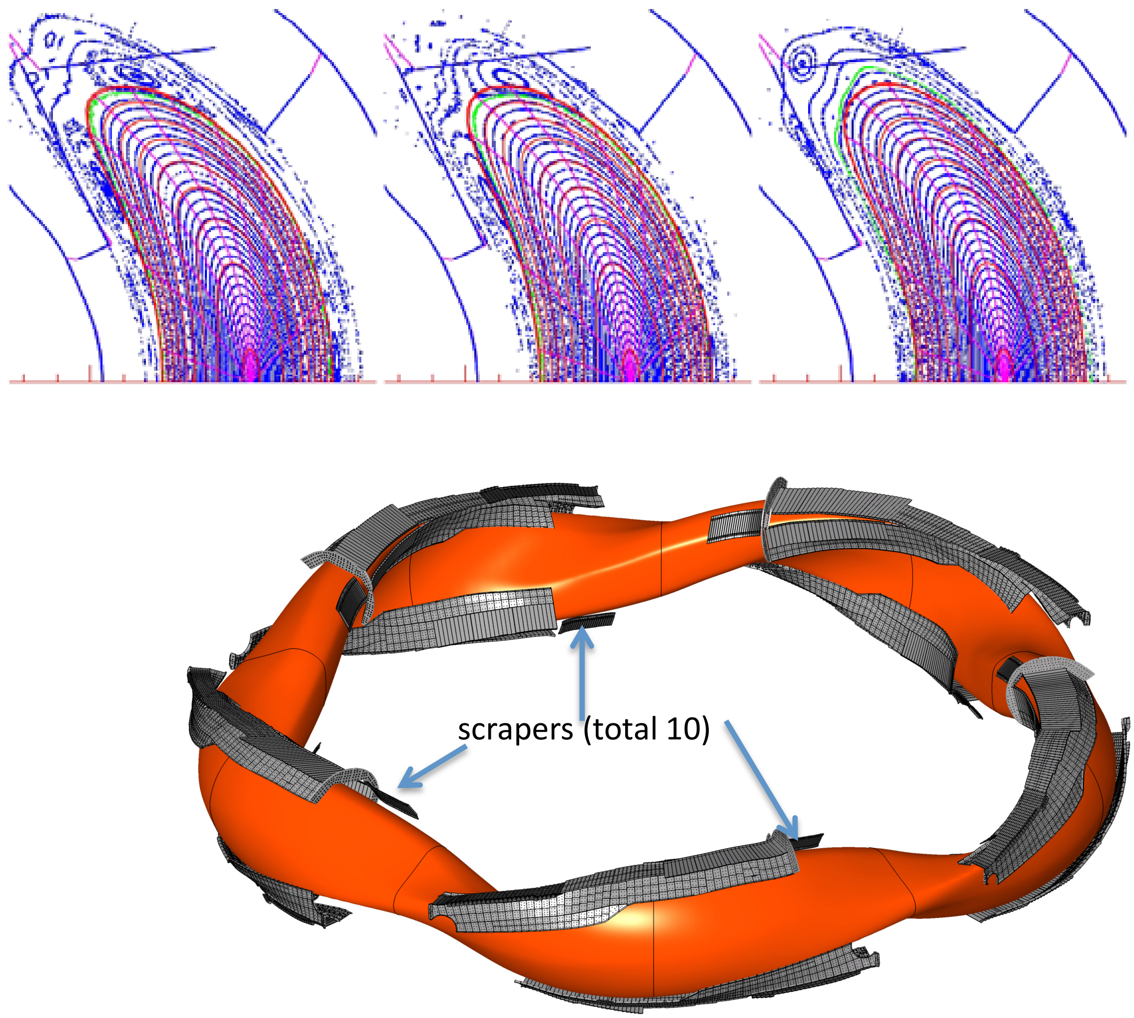

Figure 3. Wendelstein 7-X island divertor design. Top: Resonant islands at the plasma edge intersect divertor structures; bootstrap currents during startup transients can cause the islands to shift as shown. Bottom: There are ten discontinuous divertor chambers, each centered on the tips of the “bean” cross section. A “scraper” may be added to accommodate changes in loading patterns due to bootstrap currents.

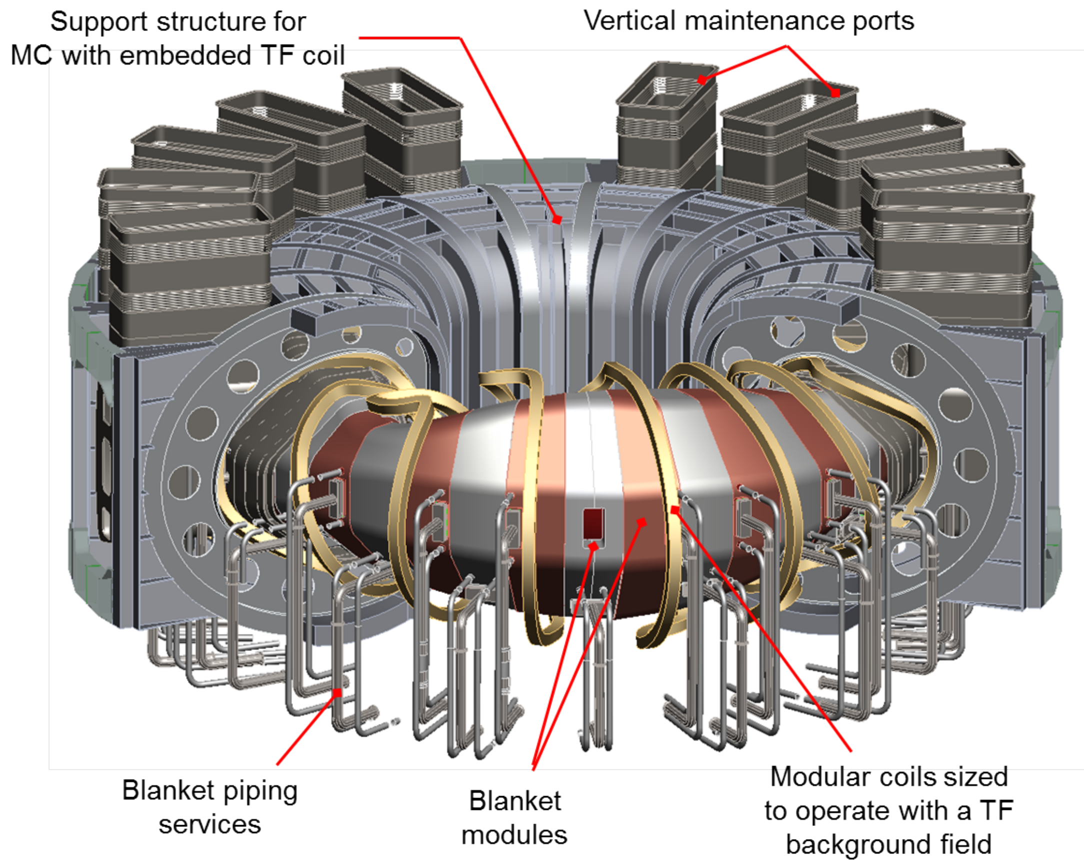

Figure 4. A quasi-symmetric stellarator power plant design, based on a modification of the ARIES-CS design including a new coil configuration compatible with large-sector maintenance of in-vessel systems.

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