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VALIDATED THEORY AND PREDICTIVE MODELING

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GENERAL ATOMICS

1. VALIDATED THEORY AND PREDICTIVE MODELING (V.S. CHAN, A.M. GAROFALO, R.D. STAMBAUGH)

Issue

Scientific progress from existing fusion experiments is providing the basis for conceptual design of an advanced tokamak (AT) DEMO. An example of an economically attractive compact fusion power plant is ARIES-AT [1], which is based on more aggressive AT physics extrapolating from on-going research. Establishing a predictive modeling capability for designing a fusion power plant will require development of theoretical models that extend to relevant reactor parameters. Equally important is a comprehensive validation program to ensure the fidelity of these models in reproducing the physics of experiments that advance toward reactor conditions.

Existing experiments operating in a low neutron environment provide ease of diagnostic access for measuring theoretically predicted quantities. They have flexibility in changing magnetic geometry and are tolerant of off-normal events. However, operating parameters of existing experiments are typically different from those of an AT DEMO, e.g. they operate in a fast ion mode instead of an electron-ion equilibrated high density regime; their pulse durations are orders of magnitude shorter; their current profiles are quite different from the broad negative shear profile expected in ARIES-AT with 90% bootstrap current that peaks near the plasma edge.

The new superconducting tokamaks in Asia will extend the AT operating space to much longer durations, and develop start-up and control methodologies that will be valuable for an AT DEMO. However, since they do not have a nuclear science mission imposing operation at high absolute plasma pressure, that particular aspect of scientific information will have to be obtained elsewhere.

ITER, being a superconducting tokamak with plans for a steady-state AT burning plasma campaign, comes closest to reactor conditions. Even here, because of the size difference, ITER would not replicate the ARIES-AT conditions. ARIES-AT operates at 80% higher β_N and requires much stronger shaping. The six times larger neutron flux requirement for ARIES-AT also means a higher absolute pressure. Last but not least ARIES-AT will have a higher edge pedestal and a broad negative shear profile that impact stability, transport and steady-state operation.

A new experimental facility that operates with parameters between ITER and ARIES-AT and produces a compact high performance burning plasma is highly desirable. *The Fusion Development Facility* (FDF) designed based on conservative expressions of advanced physics with capability for further enhancement in performance [2] will complement ITER by approaching the ARIES-AT physics performance. It offers valuable opportunities to fully diagnose and control unexplored burning plasma regimes.

The four groups of experiments will provide complementary information for validating theoretical models used to resolve critical issues facing the design of fusion reactors.

Technical Requirements

A predictive modeling code for designing a fusion reactor will include the following capabilities fully validated and integrated.

Equilibrium and Stability Optimization

MHD stability limits can be increased by optimizing the equilibrium configuration (cross-sectional shape and radial profiles) and by the combined use of conducting walls surrounding the plasma and feedback stabilization of modes.

<u>Resistive Wall Mode (RWM) Stabilization</u>. Small error fields can resonantly excite marginally stable RWMs. The result is an effective "amplification" by the plasma of the error field, which is resonant

with the marginally stable RWM. Uncorrected error fields and the associated plasma response can induce locked modes, which tend to limit the performance or cause a disruption.

The increase in the resonant plasma response to the intrinsic error field as β_N increases (yet to be fully understood) can be exploited by a feedback-based method to determine the optimal error field correction. DEMO operation at the beta values of an ARIES-AT plasma will require very accurate correction of n=1 error fields that could resonate with the marginally stable RWM.

Edge Localized Mode (ELM) Control. Future tokamak fusion power reactors, including DEMO, will almost certainly operate in high confinement H-mode. The defining feature of H-mode is its edge transport barrier that reduces particle and heat loss from the plasma. As plasma edge pressure increases, it drives repetitive ELM instabilities. Large Type I ELM event expels plasma thermal energy that impulsively heats divertor plasma facing component (PFC) surfaces. In reactors the ELM power pulses are predicted to melt or erode divertor surfaces and prohibitively limit their service life.

Suggested solutions that need to be validated include: (1) Develop a high-confinement, ELM-free operating mode, such as QH-mode, to a fusion relevant level. (2) Reduce ELM sizes, typically by triggering frequent small ELMs by means such as pellet injection or time-varying magnetic fields. (3) Reduce the plasma pressure and electric current free energy sources that drive Type I ELMs, notably by non-axisymmetric resonant magnetic perturbations (RMPs) to increase edge transport in a controlled way. (4) Use liquid plasma facing divertor surfaces.

Tearing Mode Avoidance and Stabilization. The penultimate limit to plasma pressure is due to tearing modes that break or tear magnetic surfaces. When excited, a classical tearing mode can convert to and be sustained at large amplitude as a neoclassical tearing mode (NTM). The NTM is maintained by the helical perturbation to the pressure-gradient driven bootstrap current and is therefore an issue in the AT. Local narrow ECCD at rational surfaces can stabilize or even preempt the existence of an NTM by both replacing the missing bootstrap current and increasing the classical tearing stability.

The issues for a DEMO are the "soft landing" of the q-profile during the startup as $q_{\min}$ comes down and beta is built up. The eventual steady state profile must be maintained between the rails of the desirable q values and tearing modes of any significance be avoided or quickly suppressed. A combination of broad ECCD to sustain and maintain the current profile and local ECCD at key rational surfaces for NTM stabilization must be in the mix. This is in need of determination experimentally as a basis for modeling and design.

Start-up and Ramp-up

In order to determine the optimal ramp-up scenarios and verify access to burning plasmas during the flattop phase, detailed modeling is necessary. At present transport codes with free-boundary equilibrium are being used for these ramp-up simulations. Extensive development will be needed to enable simulation of the control system, including models of power supplies and coil systems, an essential requirement for ramp-up optimization. Some of the issues that need to be addressed include:

- A viable plasma initiation sequence taking into account the large induced currents in the vessel wall and blanket assembly.
- Time evolving transport model to evaluate plasma characteristics such as current profile evolution, beam penetration and electron cyclotron (EC) access.
- Application of neutral beam and EC heating in ramp-up scenarios.
- Assessment of vertical stability using modeling to provide further input into the robustness of the vertical control system, and for evaluating different control schemes.

Heating and Current Drive

A fusion reactor or FDF requires external sources of heating to raise the plasma temperature to 10 keV or above so that effective D-T fusion can take place, with the bulk of the subsequent heating provided by the 3.5 MeV a-particles from the fusion reactions. This heating raises the plasma pressure to the point where, for ARIES-AT, 90% of the plasma current is maintained by the neoclassical bootstrap current. The bootstrap current has to be well-aligned with the desired current profiles for stability. The remaining 10% of the current, 1.4 MA, must be supported through the application of external sources of current drive.

Modeling shows that ECCD is adequate for central heating and current drive and for current profile control, although high power is required. FWCD can supply central current drive and ICRF can provide central heating, but the antenna coupling problem requires progress and only a little control of the profiles is possible. LHCD can provide high current drive efficiency, but its localization near the edge and difficulties in antenna coupling remain major issues for further research. Both FWCD and LHCD may also suffer losses in efficiency from parasitic power losses to the a-particles in the plasma.

Confinement and Transport

Analyses of experimental data together with extensive theoretical modeling have provided scientific insights into key mechanisms for confinement improvement. ExB shear is considered an effective tool in creating edge or internal ion transport barrier. Experiments also suggest that magnetic geometry may influence confinement quality. Strong positive magnetic and negative magnetic shear are thought to reduce turbulent growth rates and improve confinement. According to theory, strong Shafranov shift in high β equilibrium can suppress turbulence (α stabilization). Experiments in the next few years should provide a stronger confirmation. *The role of magnetic geometry on confinement is an active area of research that promises a unified means to improve both stability and transport, especially electron heat transport.*

The advent of drift-wave based transport models with increasing fidelity has enabled 1-D transport simulation of plasma discharges from start-up to steady-state. These physics-based transport models do not use any fitting parameters to experiments but are only fitted to more exact gyrokinetic turbulence simulations. Reproducing the temperature profiles and stored energy has been accomplished with accuracy statistically better than from using empirical formulae. Reproducing the density profile has been a challenge due to incomplete understanding of particle transport such as the existence of a particle pinch. Applying 1-D transport modeling for predicting new experiments introduces two difficulties. *The first is determining the boundary conditions and the second is justifying the density profile.* Both have to be based on sound physics. Since rotation plays an important role in confinement, *an additional difficulty is the modeling of momentum transport*. With more powerful high performance computers, gyrokinetic transport simulation over a discharge duration may be feasible.

Alpha Particle Physics

Understanding the basic physics of plasmas dominated by strong self-heating is the key goal of ITER, FDF, and other proposed burning plasma experiments and represents a necessary step in the realization of fusion as an attractive energy source. In D-T plasmas, characteristic of these devices, self-heating is provided by the slowing down of 3.5 MeV alphas generated through D-T fusion reactions. With respect to alpha particle physics, there are several outstanding issues; at the forefront is the susceptibility of future next-step devices to fast ion driven instabilities such as Alfvén eigenmodes. These instabilities may impact the success of ITER and other burning plasma experiments since they can resonate with fast ions and be driven unstable, possibly causing enhanced transport of the energetic particles necessary for heating. Present tokamak experiments with Alfvénic instabilities show flattening of neutral beam fast ion profiles and loss of injected beam ions during periods of strong Alfvénic activity. This type of redistribution or loss of fusion born alpha particles in a burning plasma experiment could reduce the performance of these devices and potentially damage the first wall.

Currently, a key focus of the worldwide energetic particle physics program is developing a set of validated models that are capable of predicting fast ion driven instabilities as well as their nonlinear consequences.

Plasma Rotation

An outstanding issue for a nuclear fusion reactor based on the tokamak confinement concept is that of imparting sufficient toroidal rotation to a high-density plasma. Toroidal rotation has been shown to be beneficial for a number of reasons, including the effect on energy confinement from toroidal rotation shear stabilization of turbulence, and the yet not well-understood effect on the β threshold for stability of neoclassical tearing modes. Some minimum values of the toroidal rotation are even considered essential to avoid the problems of error field penetration, and at very high β_N (such as in an advanced tokamak reactor) to stabilize resistive wall modes. All of these rotation effects tend to improve the plasma performance or reduce the risk of disruptions.

It has been observed that plasmas exhibit a non-zero toroidal rotation even in the absence of external momentum injection. Although this rotation may not be negligible in a fusion reactor, theories do not exist that can predict its characteristics. Toroidal rotation induced by RF has been observed in two frequency regimes, Mode Conversion Flow Drive (MCFD) in the ICRF frequency regime, and due to Lower Hybrid Current Drive (LHCD) in the LH frequency regime. Finally, recent experiments have shown that the application of non-resonant magnetic fields to plasmas with near zero toroidal rotation leads to an acceleration toward a “neoclassical offset” rotation value in the counter-I_P direction.

Pedestal Optimization

High performance operation in tokamaks is achieved via the spontaneous formation of a transport barrier, or “pedestal”, near the edge of the confined plasma. The formation of the pedestal (“H-mode operation”) is strongly beneficial to fusion performance for several reasons: (a) The pressure at the top of the pedestal serves as a boundary condition for confinement in the core. Heat and particle transport are driven primarily by gradient driven microinstabilities, and hence core confinement improves with pedestal height. Global confinement and fusion Q improve due both to this core improvement and to the pedestal itself. (b) The presence of the pedestal broadens the pressure profiles, generally resulting in higher global beta limits, which allow both higher global pressure and higher bootstrap fraction. (c) A high pedestal allows fusion relevant temperatures throughout most of the plasma volume, further optimizing fusion performance.

All of these improvements are proportional to pedestal height. Hence predicting and optimizing the pedestal height is critical for reliably achieving very high fusion performance. However, the free energy in the edge barrier can drive instabilities called edge localized modes (ELMs), and large ELMs can constrain the lifetimes of plasma facing materials. Hence the optimization process is complex, as pedestal height must be maximized, while ELMs are minimized or eliminated.

Heat Flux Handling and Divertor Physics

An AT DEMO produces significant fusion power via optimization in the plasma energy confinement time (τ_E) and the plasma beta (β_T). Higher values of τ_E and β_T are more readily obtained as the plasma shape becomes increasingly "triangular," and this is achieved more naturally in a balanced double-null (DN) magnetic configuration rather than in a single-null (SN). To design a divertor that can handle the heat flux in a DN divertor requires quantitative knowledge of the heat flux exhausting through the two legs of the divertor. This information depends critically on the cross-field transport from the closed field line region to the open field line region, the parallel heat and particle transport in the open field line, the magnetic field and plasma facing component geometry together with atomic and radiation physics.

Several empirical formulas are available based on fitting of selected experimental data. They give quite different predictions when applied to different experiments. A major source of uncertainty is the exponential heat flux scrape-off width λ_p at the midplane for conditions expected for high triangularity, highly powered tokamaks. Experimental measurements of this scrape-off width have large uncertainties. 2-D edge modeling codes that claim to calculate the scrape-off width actually substitute one uncertainty with another, namely the cross-field diffusion at the boundary of the closed and open field lines. To have confidence in designing a divertor for DEMO, a much better job of measuring the scrap-off width and a theoretical model that self-consistently treats the open and closed field line region will be required.

Active Real-time Control

DEMO and a commercial fusion reactor require high reliability control performance in many areas in order to operate steady state for sustained durations of many months close to various stability limits. Fusion power plants are expected to be among the most complex systems ever to operate for such long periods without shutdown. Many of the operational control issues are similar to those encountered in present-day fusion experiments and process-controlled plants. However, the number of regulated subsystems, the range of timescales, the complexity of the highly-coupled nonlinear multivariable control problems, the complexity of the technologies required, the sustained-duration operation required, the robustness and reliability needed, and the nuclear safety issues involved make it unlike any other large-scale process control problem, as well as a significant extension beyond present devices and ITER. Many specific control solutions needed by DEMO will remain to be developed and demonstrated following ITER, and a level of robustness and reliability far in excess of ITER requirements must be demonstrated and certified for assured performance and licensing, as well as for economic evaluation. Assuming DEMO operates in an advanced tokamak (AT) regime, it will require control solutions and reliable performance operating much closer to stability limits than required by ITER, over periods far in excess of the long pulse "steady state" ITER target of several thousand seconds.

Disruptions

Burning plasma devices including ITER and DEMO cannot tolerate frequent disruptions. Each disruption will contribute significantly to the erosion of plasma-facing surfaces. In addition, a disruption will lead to loss of operating time, in order to assess the cause and consequences of the disruption and then to regain satisfactory operating conditions. Even rapid shutdowns generated by a disruption mitigation system must be minimized, owing to their potential for wall erosion or melting and subsequent downtime.

A DEMO will require systems to avoid disruptions, and backup systems for disruption mitigation. Both types of protection systems must function with high reliability. Disruption avoidance includes several layers of plasma control. The plasma shape, pressure profile, and current density profile will be controlled to avoid stability limits; for greatest accuracy the stability limits should be calculated in real time with measured plasma parameters. Some instabilities (e.g. neoclassical tearing modes) are amenable to active suppression. Conditions potentially leading to a disruption may still result from unanticipated events such as a sudden influx of impurities or failure of a control actuator. If a disruption threatens, the control system will attempt a controlled shutdown, or "soft landing". As a last resort, when a disruption is inevitable, the mitigation system will create a rapid shutdown by gas injection or other means.

Research Thrust

Figure 1 shows schematically the structure of a fully integrated fusion simulation capability. Some bilateral couplings of enabling elements have been initiated under the Fusion Simulation Pilot Projects (Stability and Wave Heating, Core and Edge Transport) but much remains to be done. It is a grand challenge that pushes the limit of fusion theory, computational and computer science. A comprehensive validation effort is essential, which requires full utilization of existing and future experiments.

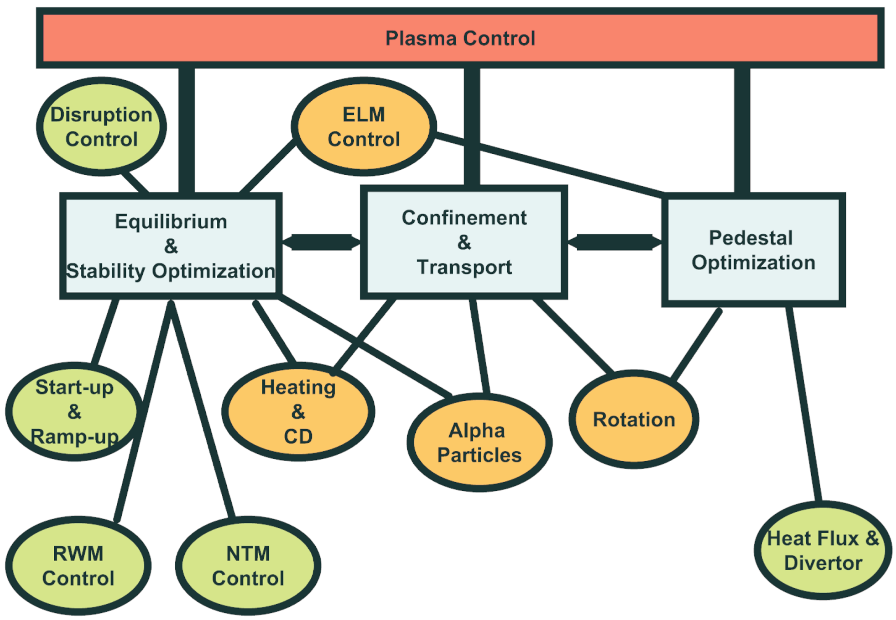

Fig. 1. Integrated predictive simulation capability.

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[1] Farrrokh Najmabadi, the ARIES Team, Fusion Engin. Design **80** (2006) 3-23.

[2] R.D. Stambaugh, et al., “Research Thrusts Made Possible by a Fusion Development Facility,” General Atomics White Paper WP 09-01, 2009.