Boundary Control Physics

Technical Subgroup of the Magnetic Fusion Concepts Working Group

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Contributing Participants:

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4. MFE Boundary control

4.1 Boundary control summary and introduction

The MFE boundary control group heard presentations on, and discussed the issues facing, a number of MFE concepts: tokamak, stellarator, ST, Reversed Field Pinch, FRC, spheromak, electric tokamak, and Levitated Dipoles. Several thoughts emerged from these meetings: (1) Boundary plasma control is an essential element of improving every concept; (2) The physics of open field line transport (perpendicular and parallel to the magnetic field) of impurities and D ions is common to all these concepts; (3) Boundary control techniques and physics understanding developed on one MFE device are usually applicable to another. This commonality is a strength of MFE in that the diversity of magnetic and physical geometries allows for a strong cross-concept exchange of ideas and techniques. The diversity gives us an important opportunity to test our physics understanding and modeling.

The progress of boundary control in MFE devices is very good. The tokamak has been the primary vehicle for this work, consistent with most of the MFE effort on this concept. We have a reasonable scientific basis for a steady-state tokamak divertor solution at high density (collisional edge, detached). Such low Te recombining divertor plasmas lead to low heat and particle fluxes at the first-wall, as well as adequate ash control, compatible with ELMing H-mode confinement. We have concerns about simultaneously handling disruptions/ELMs and tritium inventory which shorten divertor lifetime. The uncertainties regarding predictions of transport perpendicular to the magnetic field need to be reduced.

Wall conditioning, choice of wall materials and the development of the divertor magnetic configuration, all of which have been developed on tokamaks, have led to enhancements of core energy confinement and cleanliness and have been applied to a number of other concepts. We look forward to further communication of ideas among concepts.

We found that four particular issues or challenges for the next 5-10 years coalesced in our discussions. We outline those issues below as well as the opportunities to address them across concepts.

Issue A - Extend boundary control techniques to lower-collisionality plasmas and other magnetic geometries

Opportunities

- Poloidal Divertor at low density for current drive (AT, ST, Spheromak)
- Non-axisymmetric magnetic geometries (Stellarator, RFP toroidal divertor)
- Radiative Mantle (all)
- Kinetic effects, drifts (all); large mirror ratio (ST, FRC and LDX)

Issue B - Develop control of impurity sources & transport to maximize boundary radiation and core cleanliness.

Opportunities

- Flow control techniques, e.g. induced SOL ion flow, neutral flow valve (Tokamaks, ST)
- Better impurity source and transport characterization (all)
- Biasing, helicity injection, RF launchers (Tokamaks, ST, Spheromak, Mirrors)

Issue C - Develop reactor-relevant materials (e.g. low T retention and radiation effects) compatible with clean core plasmas.

Opportunities

- Solid Low-Z, e.g. Be (JET, move away from graphite)
- Solid High-Z (Mo in C-Mod)
- Liquid Surfaces, e.g. Li, FLIBE, etc. (no MFE devices yet, proposed for CDX)
- Disruption mitigation, e.g. He puff, killer pellets (several)

Issue D - Develop physics understanding of edge turbulence & core-edge coupling.

Opportunities

- Diagnosis and modeling of perp. transport in presence of open field lines, particularly J, no, and flow in edge (RFP, Tokamaks, Stellarator, ST)
- Deep core fueling, wall conditioning, materials effect on core (particularly important in emerging concepts and steady-state devices)
- Control heat and particle flux transients, e.g. ELMs (all)

4.2 Boundary control issues/opportunities for specific MFE concepts

4.2.1 Tokamaks

Contributions from A. Mahdavi, GA and S. Pitcher, MIT

The challenge of boundary physics in tokamaks and all toroidal magnetic confinement devices is to find an "integrated" solution to the coupled problems of heat exhaust, helium ash removal, tritium inventory and wall lifetime, in a manner that is consistent with good core performance. These problems in many situations impose conflicting design criteria that must be resolved by innovative techniques. An example of such a conflict is that which exists between maximizing power dispersal in the divertor (maximize impurity radiation) and minimizing radiative power losses in the core (minimizing impurity density there).

Geometry: In tokamaks the current leading geometry for boundary plasma control is the pumped single null poloidal divertor, although the double null configuration has not been ruled out. This configuration is favored over limiters because of its amenability to radiative heat dispersal, more effective impurity control and ease of particle exhaust. In this configuration, the typical experimentally observed e-folding width for the parallel heat flux in the SOL is about 1 cm at the outside midplane of the plasma. Although narrower SOL widths have been observed in present devices (e.g. C-MOD), in the absence of a good model the 1-cm nominal value is normally used for divertor heat flux estimates.

Heat removal: In absence of radiative heat dispersal or a technique for broadening of the SOL, the divertor surface heat flux in a conventional tokamak reactor will be several tens of MW/m^2. It is a simple exercise to show that once the requirements of erosion lifetime are taken into consideration (i.e. plate thickness), this level of heat flux can not be exhausted with solid target technologies. Fortunately, extensive experiments in a number of existing tokamaks, notably DIII-D, C-MOD, JET, JT60-U and ASDEX-U, have shown that sufficient radiative heat dispersal is achievable in a standard tokamak reactor (e.g. ITER or ITER-RC), operating in the high recycling or detached regimes, i.e. with collisional scrape-off-layers. In these experiments radiative heat dispersal was achieved in high confinement H-mode plasmas, in most cases accompanied by ELMs.

Impurity and density control: Impurity and density control has been achieved in H-mode plasmas in both normal high recycling and detached plasmas. In most machines this is limited to discharges with ELMs, although EDA discharges in C-MOD similarly have good particle control. Particle exhaust rates sufficient for helium ash removal in a reactor have been also demonstrated in various plasma conditions, including partially detached plasmas. In the future, further work is needed to understand impurity transport in core plasma modes other than the ELMing H-mode, since in many high confinement modes such as VH-mode, despite a very low impurity source term, core impurity accumulation has been observed.

Divertor target lifetime: The divertor target lifetime is primarily determined by erosion due to normal particle flux and off-normal events (such as plasma disruptions). The normal erosion rate can be reduced to acceptable levels if the temperature of the divertor plasma in contact with the target materials is reduced below a certain threshold. In the case of tungsten, this is readily achieved in present devices; however, tungsten does not survive the high heat pulse of an unmitigated disruption. Graphite on the other hand can withstand a moderate disruption, however it has a low threshold temperature for physical and chemical sputtering. Although recent experiments on DIII-D indicate that the divertor can be operated at temperatures below the graphite erosion threshold, it is not clear that this can be achieved in a next-step tokamak reactor. Carbon also suffers from co-deposition, which will probably result in unacceptable levels of tritium inventory in co-deposited layers.

Future Directions: From the point of view of boundary physics we are ready to proceed with a next-step tritium burning experiment that includes a detached divertor. With it we can verify the validity of our radiative divertor models and carry out development and testing of plasma facing materials in a reactor relevant environment. While there are some concerns with respect to simultaneously satisfying the requirement of long divertor plate lifetime and minimal tritium retention, present designs for future devices appear to allow flexibility to deal with uncertainties. In the meantime we can effectively utilize present devices for the purpose of refining our physics models with the goal of achieving basic understanding of the boundary plasma and achieving predictive computational capability. Furthermore, we can apply our resources towards further development of plasma control concepts suitable for the more demanding conditions of lower-density AT plasmas.

Unresolved aspects of boundary plasmas physics can be addressed with modest upgrades of existing facilities and codes. These include, the relationship between edge impurity sources, impurity transport and core impurity levels, the thermal stability limits of edge and divertor plasmas, the impact of ELMs and off-normal events on the lifetime of divertor components and the eventual mitigation of any associated erosion, the fundamental understanding of perpendicular transport in the SOL (and its dependence on core confinement) and the optimum choice of divertor target materials. The measurements that will allow progress in modelling and physics understanding include the neutral density profile, neutral velocity distribution, plasma flow velocity and two dimensional density and temperature profiles throughout the SOL.

Conceptual development and testing of several advanced boundary-control schemes can be carried out in existing tokamaks with modest modifications. Advanced divertor concepts currently include divertor impurity enrichment (“puff and pump”) and SOL broadening by formation of an ergodic layer. Both of these are aimed at extending the radiative divertor solution to lower plasma densities. Finally, liquid divertor targets might even permit handling the full divertor heat load without radiative dispersal.

4.2.2 Spherical Tori

Contributions from Rajesh Maingi, ORNL, and Anthony Leonard, GA

Spherical Tori (ST) are, by design, compact high power density, low toroidal field (B_ϕ) devices, and many of the issues related to boundary physics are generic to high power density systems. In addition, ST have several edge magnetic topological features (high mirror ratio, short connection lengths) which can affect the SOL width, impacting heat removal and impurity control requirements. It is noteworthy that helium transport/exhaust issues applicable to tokamaks should apply to ST as well, and readers are referred to the tokamak discussion in that area.

Heat removal: High power density systems face a more severe challenge in removal of input and generated power than low power density systems, due to a smaller heat deposition area. One technique to reduce peak heat flux in present ST experiments is to operate in inner-wall limited (IWL) configuration, which has magnetic flux expansion factors (~ 10) just outside the last-closed flux and which can disperse power over a wider footprint than in the diverted configuration. The extrapolability of this configuration to next step devices is debatable, however, due to impurity control and helium exhaust issues. Because ST are closely related to tokamaks, many boundary physics issues are common and some effort should be made to connect ST edge research to prior, extensive tokamak studies.

In addition, however, ST edge magnetic topology features differentiate them from other magnetic confinement devices, providing an opportunity to study the fundamental boundary transport processes and complementing studies from other MFE concepts. These topological differences include an absolute magnetic well and short SOL open field line connection length, leading to enhanced kinetic effects (e.g. strong magnetic mirror) and E X B drift flows.

1. Mirror physics– the total magnetic field in ST scrape-off layers (SOL) varies by a factor of 4 in inner-wall limited (IWL) configurations and by a factor of 2 in diverted configurations. This high mirror ratio ($R_M \equiv |B|_{max} / |B|_{min}$) leads to a trapped ion fraction between 75% and 90% in the SOL. Thus, a substantial flux of ions is reflected in the SOL prior to striking the limiter or divertor plates, effectively increasing the parallel connection length in a collisionless edge plasma. Because the SOL width is set by the ratio of parallel and perpendicular transport fluxes, the longer effective connection length could lead to a larger SOL width in ST. The picture is complicated by the effect of collisions, which tend to mitigate the effect of the mirror reflection. Note that kinetic/mirror effects in the SOL are present in other MFE concepts as well, but the magnitude is accentuated in ST.
2. High Field line pitch – ST configurations result in long field line length in the good curvature region and short field line length in the bad curvature region (e.g. outboard side). At the outer midplane, the ratio of the poloidal field (B_θ) to B_ϕ is ~ 1 in present systems, and the field line pitch at the outer midplane can be 45-60^0. This leads to a short connection length from the

midplane to the target plates, particularly in double-null divertor configuration. Note that this effect counteracts the longer effective connection length obtained from mirror reflection effects discussed above in (1).

3. SOL drifts – The fact that B_θ and B_ϕ are comparable and relatively small in the outer SOL suggests the $\mathbf{E} \times \mathbf{B} / B^2$ drift terms may lead to unique SOL flows and drift patterns.

Research Needs: 1) Measure SOL parameters, particularly e-folding widths, and compare with theoretical models of cross-field transport. 2) Look for evidence of mirror physics and SOL drifts. 3) Study heat flux mitigation scenarios in ST, including detached divertors, radiative mantles, and possibly even liquid walls. 4) Compare properties of IWL configurations with diverted configurations. 5) Compare boundary characteristics with tokamaks, which share many common issues.

Impurity and density control: compact systems have a greater need for impurity control because the plasma and first wall are necessarily close, which can lead to high impurity production rates. Also, the short connection length in certain ST configurations makes maintenance of a temperature gradient from the outer midplane to the target more difficult in high power systems. Thus the compatibility of H-mode confinement discharges, which require high plasma temperature at the outer midplane, and low impurity production, which requires low plasma temperature at the target, could become an issue. Finally the ST plasma has an additional requirement of non-inductive startup, in present and future concepts, due to the limited space available for an ohmic heating central solenoid. This requirement is shared with other MFE concepts, such as spheromaks. One mechanism to supply that current drive is coaxial helicity injection (CHI), via strong (~ several kV) divertor and vessel biasing. Thus, the issues related to boundary control may have the additional complication of a biased outboard divertor, with respect to the inner divertor in the case of single-null. Finally, the limited space for an inner wall makes pellet fueling from the high field side more problematic also, but fueling from the low field side may have sufficient penetration and efficiency due to the reduced B_ϕ in ST. **Research Needs:** 1) Study the compatibility of basic edge parameters required for efficient current drive in the startup phase with requirements for heat flux mitigation scenarios. 2) Study impurity production rates in H-mode discharges. 3) Test pellet penetration and fueling efficiency from the low field side to compare with other MFE concepts.

4.2.3 Stellarators

Contributions from P. Mioduszewski, ORNL

Stellarator boundary layers, in general, don't have an ordered magnetic field-line structure outside the last closed flux surface (LCFS) like axisymmetric tokamaks. Stellarator boundary layers are more complex than tokamak boundary layers and can be combinations of stochastic or ergodic layers, single islands or island chains, and laminar layers near the coils. Due to their helical symmetry, stellarator boundary layers require <u>3-D models for magnetic fields and plasma transport</u> (neutrals transport needs to be 3-D in any magnetic configuration). Since stellarator boundary configurations can vary widely with the coil design, usually optimized for the core plasma, they need to be taken into account in the design of the vacuum vessel and the plasma-facing components (PFCs). Specific issues are discussed below.

<u>Heat removal</u>: Average power fluxes in the plasma boundary of present stellarators are smaller than those in tokamaks, because the aspect ratios are typically 2-3 times as high as in tokamaks. Compact stellarators, on the other hand, will have aspect ratios similar to standard tokamaks and their power loads will be similar to that of tokamaks. The details of the power flux densities, peaking factors, etc. depend on the details of the magnetic boundary configuration and on the design of the plasma-facing components. Challenges will be <u>highly localized islands</u> intercepting the walls and <u>toroidally discontinuous divertor modules</u>, which can both lead to high local power flux densities. Stellarators don't depend on current drive for long-pulse operation and can be operated at high densities.

<u>Impurity and density control</u>: In stellarators that have been operating over the past decade, density and impurity control have been a major challenge, especially with NBI. To overcome the uncontrolled density rise during neutral beam injection, these stellarators originally applied titanium (Ti) gettering. Ti gettering is still applied in LHD, whereas in most other stellarators, boronization has become the preferred method of impurity control. Oxygen reductions by factors of ten can be achieved.

<u>Particle confinement</u> in stellarators has been observed to be high enough for possible <u>impurity accumulation</u>. Therefore, impurity sources need to be minimized and scrape-off layer shielding maximized. Because of the possible island structure, it is important to carefully tailor the plasma-facing components to the boundary configuration. Since impurity concentrations tend to decrease with plasma density, high-density operation in stellarators is favorable for impurity control. Radiative boundary operation is being studied in W7-AS. Since there is presently no continuous impurity removal, such as neon pumping as in some tokamaks, the preferred impurity is nitrogen, which is sufficiently pumped by the wall to allow injection with feedback control. Particle control during plasma start-up, especially with neutral beam injection, has been

an issue for essentially all stellarators. Except for the experiment with a local island divertor in CHS, particle control in stellarators is mainly performed through wall conditioning. The problem of limiting the density rise has been solved with titanium gettering, helium glow discharge cleaning between plasma discharges, and boronization. Fueling of stellarators is performed with gas puffing, pellet injection, and neutral beam injection. Wide ergodic layers or island chains could make gas fueling very inefficient and load the boundary with neutrals, so that pellet fueling might lead to better plasma performance.

Helium pumping: Helium pumping has not been a priority in stellarator research up to now. Since it needs some divertor or pump limiter configuration with continuous pumping, most existing stellarators are not equipped for helium exhaust. The two large stellarators LHD and W7-X, however, will be able to investigate helium pumping and exhaust since they will have full divertors with continuous pumping.

First wall lifetime: Since stellarators don't experience disruptions like tokamaks, the lifetime-limiting process is likely to be 'normal' operation. While this is not a problem in present short-pulse devices, steady state operation can lead to intolerable erosion in a short time. To remedy this, our presently available options are materials choice and tailoring of the boundary plasma. Recent experiments have indicated that tungsten, which has much lower erosion rates than graphite, could be a viable choice for plasma-facing components if the plasma temperature can be kept low enough.

Edge/core transport coupling: With a symmetrized plasma boundary, the characteristic decay lengths in the SOL are exponential and in the order of 1-3 cm. In W7-AS H-mode operation, transport across the separatrix is diminished by a strong radial variation of the electric field and a corresponding velocity shear layer already observed in L-mode. At the L-H-mode transition the edge gradients increase: the density gradient increases by 40-60% and the temperature builds up a pedestal for $r/a>0.7$. Edge turbulence is strongly reduced during H-mode. During ELMy H-mode (1-2 kHz) global confinement is close to the L-state.

Overall, the transport from the core into the plasma boundary shows similar properties as in tokamaks. There is, however, some potential for differences in the boundary transport: (1) an ergodic layer can have strong radial components of the magnetic field, leading to enhanced radial plasma transport, and (2) plasma and impurity particles could be trapped in islands outside the core plasma and exhibit relatively long confinement times. Generation of transport barriers and investigation of their properties is presently subject of intense efforts in stellarator research.

4.2.4 Reversed Field Pinches

Contributions from Daniel Den Hartog, U. Wisconsin

Status:

Power and particle handling are important issues for any reactor concept, but especially so for the RFP as a potentially compact, high power density reactor core. However, present and past RFP devices have not had to deal with these issues in detail given the relatively short duration of the plasma pulse. Typically RFP devices use graphite, molybdenum, or other refractory materials to protect sensitive areas in contact with the plasma. Only the RFX and Extrap-T2 devices have full coverage graphite first walls.

Presumably much of the particle and power handling knowledge base currently being developed in the larger fusion research community (mostly the tokamak community) will directly transfer to the RFP. But there are several RFP specific features which need to be considered. For example, because the dominant magnetic field component at the plasma surface is poloidal, a poloidal divertor would require large divertor coil current. In the TITAN reactor study, a toroidal field divertor was chosen instead. However, most of the power was not deposited in the divertor, rather it was assumed to radiate uniformly on the first wall surface by deliberately doping the plasma with a small amount of xenon. The divertor functioned primarily to exhaust helium. Consequently the uniform first wall heat load (radiation) in TITAN was 4.6 MW/m^2 and the neutron load was 18 MW/m^2.

The RFP divertor (operating on either the toroidal or poloidal magnetic field) must be carefully designed to avoid decreasing MHD stability by moving the plasma far from the stabilizing shell. It is known that the stability of the RFP is decreased with a vacuum (current-free) interspace between the plasma surface and shell. Also, the harmonic structure of the divertor field might need to be chosen so as not to interact with unstable modes. Only one (smaller) RFP device operates with a divertor, TPE-2M in Japan. Both poloidal and toroidal magnetic field divertor configurations are produced, and the program emphasis is studying the divertor impact on MHD stability and impurity control.

Present day RFP experiments need to begin to develop a long-term strategy to handle plasma-boundary issues more proactively. This will become especially important as current experiments add auxiliary power, both radio-frequency and neutral beam. Such power sources will cause increased energetic particle outflux, but will require good particle control to maintain efficiency.

Key issues:

<u>**Heat removal**</u>:

The dispersal of the heat load by a deliberately formed highly radiating plasma is a key TITAN design feature likely to be incorporated in any high beta, compact reactor concept incorporating a close-fitting shell. It is critical to develop experimentally a radiative edge and/or core while not degrading core transport - a task which the RFP shares with the tokamak. A related issue that affects heat and particle handling is mode locking. The nonlinearly interacting band of tearing modes tends to form a spatially localized perturbation which concentrates the plasma-wall interaction. When these modes lock, the power flux to the first wall is not uniform. Learning how to prevent mode locking is already an area of intense RFP research. Finally, the fast electron population observed at the edge of most RFP's must be understood and controlled to reduce heat flux.

Particle control:

The compatibility of a toroidal divertor for particle control with RFP operation needs further intense investigation. Particular issues which complicate divertor implementation are the need for a close-fitting conducting shell to stabilize both internal tearing modes and external kinks, the requirement of low field errors to prevent locking, and a relatively large ion Larmor radius in the edge. Diagnosing the edge and divertor will be particularly important in modeling divertor erosion and performance. Also, the compatibility of RFP operation with a pump limiter should be examined.

Opportunities for resolution:

Since the RFP is a self-organized system in which the plasma seeks a relaxed state via global dynamics, the edge and the core are tightly coupled. For example, lowering impurity influx and controlling edge fueling appears to be a prerequisite for the transition of the plasma to improved confinement modes. This close coupling of the core and edge will challenge existing scrape-off layer and divertor models and may be useful in improving the physics basis of such models. In addition, active modification of the RFP edge via heating and current drive may soon be possible, further elucidating the physical links between the core and edge and perhaps leading to further performance improvements.

There is a particular need for an additional Concept Exploration experiment to investigate divertor geometries in the RFP with a focus on developing an efficient divertor compatible with the RFP (MHD stability, field errors, etc.). This experiment should build on the results of TPE-2M.

Although still very speculative, the successful development of liquid walls for MFE has multiple benefits for the RFP. In addition to the possibility of solving the resistive wall kink mode problem, liquid walls could provide a means to achieve high wall loads, thereby maximizing benefits from potentially compact plasma configurations such as the RFP. Also a liquid wall could greatly reduce thermal stresses for a pulsed reactor scenario which is particularly advantageous for the RFP given its large current requirement and lack of pressure-driven current.

4.2.5 Electric Tokamaks

Contributions from L. Schmitz, UCLA

Introduction

The Electric Tokamak (ET) is a high beta confinement concept with moderately high aspect ratio. The Electric Tokamak relies on three major physics objectives: 1) Achieving near classical ion transport by fast poloidal bulk plasma rotation across the minor radius; 2) Exceeding the conventional plasma beta limit for MHD modes, called the “first stability limit,” by rotational stabilization of kink and ballooning modes; 3) Reducing anomalous electron heat transport due to the high effective aspect ratio produced in a high beta, magnetic well configuration.

Control of impurity release and radiation losses as well as control of bulk plasma recycling is crucial in ET since operation past the conventional density limit is required to produce a magnetic well configuration and/or reach unity plasma beta. The Electric Tokamak is presently at the stage of concept exploration and issues related to this stage and the proof-of-principle stage are mainly discussed here.

Issues and Opportunities for Resolution

Heat Removal

Heat removal from a conventional, high field tokamak is hampered by the narrow scrape-off layer radial power flux profile (width <1 cm) leading to a concentrated heat load in excess of 1 MW/m^2 onto the divertor plates. In a high beta electric tokamak, the scrape-off layer will be wider since the toroidal magnetic field will be smaller (< 1T). The total power exhaust from an Electric Tokamak Reactor (α, n, core plasma efflux), however, will be in the range 0.3-3 GW, and is thus comparable to an advanced tokamak reactor.

The concept-exploration ET at UCLA is a large, low field tokamak (R = 5m, a = 1m, B_t = 0.25 T). The total power input will be < 2.5 MW with a maximum pulse length of several seconds. Due to the large vessel surface area and the low magnetic field, the projected surface heat load will be very low: $0.01\ MW/m^2 < P_{surface} < 0.3\ MW/m^2$, depending on details of the magnetic and divertor configuration.

A proof-of principle ET tokamak will require B_t = 1T and P_{aux} = 5-10 MW with pulse lengths of ten seconds to several minutes. The expected wall or divertor heat load should still be well within conventional engineering limits (< 3 MW/m^2).

Impurity and density control

The Troyon beta limit can be substantially exceeded in ET only if the conventional density limit is surpassed. Recycling (and impurity radiation) control is crucial in order to avoid radiative collapse associated with the density limit, and ET needs to operate at Z_{eff} ~ 1 to achieve this.

Edge recycling control is needed to maintain a low edge neutral density and avoid excessive damping of poloidal rotation by charge-exchange collisions.

The ET concept exploration phase will employ titanium gettering or active Ti burial to control plasma recycling and oxygen radiation in H or D plasmas. He discharge cleaning in between shots will also be employed.

ET also affords access for deep fueling by inboard neutral beam injection or inboard pellet launch and thus is a testbed for these core fueling techniques at moderate to high β.

A major innovation in ET is (ICRF) plasma heating at the high field side (due to excellent accessibility of the high aspect ratio configuration). Inboard heating will be used to control the edge electron temperature profile and to counteract the poloidally asymmetric outflow of heat into the outboard scrape-off layer. It may thus be effective in preventing radiative collapse near the inboard plasma boundary.

Distributed large area RF antennas allow good spectral definition (narrow k_{par} spectrum) and reduce the RF coupling power density to very low values (<0.2 MW/m^2). All RF current straps are located outside the vacuum vessel. Polyethylene vacuum windows are used, protected from plasma bombardment by a stainless steel heat/particle shield. The low RF coupling power density will reduce the antenna RF sheath potential and, therefore, antenna-related impurity release. In the concept exploration phase, the sputtering of titanium due to bulk ion flux and fast minority ions is estimated to produce < 200kW of impurity radiation loss. The higher density operation of the proof of principle experiment will aid in impurity screening. The boundary configuration for the proof-of principle experiment will have to be developed based on the experience gained during concept exploration.

Helium transport

To avoid helium ash buildup in a reactor, the effective helium particle confinement time has to satisfy $\tau_p*/\tau_E < 10$. This condition is generic (as well as issues relating to active He pumping) and it is independent of the particular confinement concept. ET offers the unique opportunity to explore the relative transport of D and He in the core plasma, as well as the scaling of particle transport compared to heat transport, at moderate to high β.

Edge/Core transport coupling

If ion transport in the ET core can be improved to a near classical level, a large discrepancy in transport between the closed and open fieldline regions may be expected. This could lead to a steep density/temperature gradient inside or at the last closed flux surface. During concept exploration we will investigate the effects of the edge plasma on the core flow profile and the importance of possible relaxation oscillations.

4.2.6 Spheromaks

Contributions from: T. Jarboe, UW, T. Rognlien, and D. Ryutov, LLNL

Spheromak research, being funded at a comparatively low level, has been focused mostly on the issues of helicity injection and core transport, although some information on boundary plasmas has been obtained. While there have been no extensive studies of the plasma-wall interaction for reactor-relevant conditions, the following plasma-boundary work has been done: a broad conceptual study for existing experimental devices, identification of key issues in spheromak-reactors, and the first steps for numerical simulations. There will be a focused effort made to study edge plasmas in the new Sustained Spheromak Physics Experiment (SSPX) device at LLNL. Many of the results obtained and methods developed for tokamak edge plasmas are applicable to spheromaks, thereby accelerating progress on spheromak edge-plasma research.

Heat removal: Being compact devices, spheromaks must deal with high power densities. Power exhaust is aided by a magnetic separatrix that allows heat loads to be diverted away from the chamber side-walls and dispersed over large surfaces areas in the divertor region. Such divertors could be constructed in the gun electrode region and near the toroidal axis. Also, because spheromaks have no toroidal B-field at the edge, the connection length along **B** between the plasma edge and the divertor surface is much shorter than for tokamaks, giving a narrow scrape-off layer (SOL). Thus, plasma losses to the side wall should be small since the spacing between the separatrix and the conducting wall can be at least 10% of the major radius, as set by MHD stability considerations. Lack of side-wall damage has been confirmed by observations in the Compact Toroid Experiment (CTX) performed at LANL. The width of the SOL is also controlled by the characteristics of edge turbulence which needs more detailed understanding.

The divertor region may also serve as electrodes used to drive edge currents for helicity injection to sustain the spheromak. In a reactor, the power input from such injection is much smaller than the alpha heating power, so the divertor needs to accommodate only slightly more power than for a non-driven edge.

Impurity and density control: Experimentally, carbon and oxygen limited the energy confinement for earlier devices (*e.g.,* CTX , although the walls were all metal), but improved with clean-up techniques such as discharge cleaning and titanium gettering. The electrodes and side wall are potential sources of impurities, but this has not been observed in experiments such as CTX. Lack of electrode impurities may be due to high voltages appearing across the electrode only briefly during discharge initiation. Density is controlled by gas puffing in the gun electrode region, and a sufficient edge density is needed to maintain the edge current for helicity injection. Additional density control could be provided by pellet injection.

<u>Helium pumping/control:</u> Helium removal should be similar to divertor tokamaks. Resources permitting, special experiments on diffusion/retention of impurity ions in the hot plasma region in spheromak core can be undertaken in the SSPX device.

<u>First-wall/divertor lifetime;</u> Conducting side-walls are needed for MHD stability, but their distance from the separatrix should prevent excessive erosion - see (1) above. Electrode surfaces may show higher erosion than passive divertor plates for near-term experiments, but this should not be true for reactors where the power flow associated with helicity injection will be very small compared to the alpha heating power. Wall activation by neutrons is similar to tokamaks, but here the simply connected geometry of the first wall and blanket, and the smaller size, would facilitate a more rapid and economical replacement of solid first-walls, allowing high power density in the spheromak.

With its simpler geometry, the use of liquid first-walls could be easier in spheromaks. In addition, the essentially poloidal B-field at the edge would not impede poloidal flow of a liquid metal. For using non-conducting liquid first-walls, an important issue is the MHD stabilizing effect from a conductor behind the thick, non-conducting liquid; estimates indicate this may be feasible.

<u>Edge/core transport coupling</u>: The conducting side-wall is a barrier to magnetic helicity, so the helicity is transported into the core rather than to the wall. It may be possible that helicity-injection current-drive for reactor-grade plasmas can be done with only electron fluid turbulence and the edge plasma would not be transported inward. Thus, particle transport may be similar to a tokamak. The power and current (via helicity) are both fed to the core from the edge region in the formation stage, presumably by magnetic turbulence.

4.2.7 Field-Reversed Configurations (FRCs)

Contributions from L. Steinhauer, UW

High level of thermal isolation: An open coil arrangement, as in an FRC, allows an arbitrary separation from heat collection surfaces, and greatly mitigates the pumping required to control recycling of deuterium or impurities. This may allow nearly complete thermal isolation of the plasma from cold boundaries and would allow the plasma boundary to run hot, as is present FRC experiments. A hot boundary reduces the susceptibility to ITG/ETG driven transport. *Issue*. (1) In near steady operation can recycling be reduced to such a level to allow "hot boundary" operation? (2) Experimental measurements are needed of the edge plasma and the jet plasma between X-points and collector plate), in particular the density and flow rates.

Edge/core transport coupling: The core of an FRC may be a relatively stable minimum energy state (MES) based on two-fluid theory. However, because of rapid losses of particle in the scrape-off layer (SOL) make it likely that the SOL structure will depart from a MES, giving rise to local instability or enhanced transport. *Issues*. (1) Do edge processes regulate the overall transport rate (bulk confinement)? (2) Can the edge properties of an FRC be modified to reduce the transport of particles and energy as in the tokamak H-mode? (3) What active or passive controls can be applied at the edge to bulk confinement of the FRC? (4) Why is the outflow of plasma in the SOL of FRCs anomalously slow (it does not seem to be the result of recycling)?

Fueling and density control: Refueling in long-pulse experiments is accomplished by a combination of feeding gas to the plasma edge and deep injection of frozen D_2 pellets. In FRCs relaxation processes may simplify refueling. Based on two-fluid theory, an FRC should rapidly find (and continue in) a high-β minimum energy state with significant flow shear. If so then any plasma supplied at the edge, whether by shallow pellet refueling, gas puffing, or merging with another FRC may be uptaken into the core as a consequence of the relaxation. *Issues*. (1) Can edge density injection lead to core fueling by the relaxation process? (2) Can this be done without compromising the "hot boundary" feature of FRCs?