

HEATING, CURRENT DRIVE, AND FUELING TECHNOLOGY

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Draft 3 Revisions:

Common questions in Sec. 1 changed per revisions by Technology Group.

Sec. 2.4 Table I modified in response to comments from several people.

Sec. 3.2 modified by contributors.

Sec. 4.4 modified by contributors.

1 Introduction

This report is to provide information for discussion at the Snowmass meeting by the Heating, Current Drive, and Fueling Subgroup (H/CD/F). It is an attempt to answer a particular question in the Plasma Support Technology field. The question is:

- 1. What is the potential for and what advances will be required in profile control technologies (plasma heating, current drive and fueling) to enable present, near term, and next step devices to meet their performance goals and ultimate research potential?**

In addition, this subgroup is to contribute to responses to the following questions that are common to all Plasma Support Technology Subgroups:

- A. What are the most important contributions that technology can make over the next 10 years: 1) to improve the vision for an attractive and competitive fusion product, and 2) reduce the cost of R&D for fusion?**
- C. What contributions will Technology make to advancing science? What research areas will be pushing the frontiers of sciences? What constitutes concept**

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exploration, engineering proof of principle and engineering performance extension for fusion energy systems?

The report is organized as follows:

- Section 2 summarizes the requirements of profile control in future experiments, and the status and capabilities of various techniques.
- Section 3 describes the status and R&D needs of specific heating and current drive techniques.
- Section 4 discusses the status and R&D needed for various fueling and disruption mitigation concepts.

2 Heating, Current Drive, and Fueling Needs

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2.1 Background

Heating and current drive technologies are essential for heating plasma to fusion-relevant betas and temperatures and manipulating plasma properties to access advanced operating scenarios (reversed shear, MHD stabilization, turbulence suppression). Significant progress has been made in developing and deploying high-power gyrotrons in the ~1-MW level at 110 GHz and the development of 170-GHz prototype units for electron cyclotron heating/current drive (ECH/ECCD) and fast-wave (FW) antenna arrays in the >1-MW unit size for Ion Cyclotron Heating (ICH) and current drive (via direct electron heating). Progress is also being made in other countries on the development of negative-ion based, high power neutral beams (0.5–1.0 MeV). With the present program emphasis on increasing plasma performance and reducing next-step option costs, the emphasis of the development of these heating and current drive technologies will concentrate on improving power density (higher voltage limits for ICRF launchers), higher gyrotron unit power (2 to 3 MW), increased efficiency gyrotrons featuring multistage depressed collectors, ICRF tuning and matching systems that are tolerant to rapid load changes, and steady-state gyrotrons and actively cooled ICRF launchers for long-pulse/burning-plasma, next-step options.

Fueling is another technology that is essential for the achieving fusion-relevant plasma parameters and manipulating plasma parameters to achieve improved performance (peaking of the density profile for higher reactivity and reducing transport via turbulence suppression). Recent successes include sustained operation above the density limit on DIII-D, high-field side launch with improved density profile peaking, internal transport barrier generation, the development of steady-state pellet injectors operating in the 1.5-km/s speed range, and the demonstration of core fueling in proof-of-principle experiments using accelerated compact toroids (CTs). Pellet fueling technology has also been used recently to ameliorate the effects of major disruptions in tokamaks by delivering massive amounts of low- and high-Z material that rapidly quench the current in vertically unstable plasmas. It has been estimated that eliminating disruptions in tokamaks in the fusion energy development class would increase the lifetime of divertor plasma facing components by a factor of two. Reducing the severity of disruptions could allow the advanced tokamak to operate nearer its ultimate potential. A critical issue for fueling in next-step device plasma regimes is the degree to which profile peaking is needed (for higher density operation and improved reactivity and confinement) and the technological requirements to meet that need (pellet speed, CT density and the physics of CT deposition).

2.2 Needs for profile control

The ability to control the heating, fueling, and driven current profiles will lead to the next level of plasma understanding and (it is hoped) improvement in performance of present-day and next-step fusion devices. There are several potential uses of H/CD/F technologies.

Plasma profile control to improve confinement

The ability to manipulate plasma profiles (pressure, density, current, and/or electric field profiles) *and sustain them in the correct state for long periods* is believed essential for achieving improved confinement and stability (e.g., the “advanced tokamak” modes in tokamaks). In addition, the achievement of high (VH-mode) confinement will require the ability to develop and

sustain a transport barrier near the edge of the plasma. This is generally believed to be possible by manipulating the electric field and/or plasma rotation velocity profiles in the plasma.

The profiles needed are not known, although theoretical work in this field is a hot area of investigation. Furthermore, the *mechanisms* by which the “good confinement” regimes are set up are not well understood. Therefore, it is important (at least at present) that H&CD systems must be designed to allow *flexibility* in the heating deposition and driven-current profiles that they can generate.

Non-inductive current drive

Long-pulse or steady-state operation in tokamaks will require the full plasma current to be driven non-inductively using some combination of current-drive techniques and bootstrap current. The requirement (for tokamaks) that steady-state operation be achievable is in addition to the requirement of plasma profile control described above.

Control of burning plasma

While some methods of burning-plasma modes are calculated to be stable, others are not. In this case, stabilization of the power output of the fusion power may require feedback control of the heating power, fueling rate, or plasma confinement. Schemes to do this have been proposed using heating and fueling.

Plasma-wall interaction/mitigation

It is generally recognized that for future reactor-scale tokamaks (and perhaps stellarators), the use of “simple” divertors will cause heat loads on the divertors and/or first walls that will be difficult to handle at best, and possibly impossible to withstand for long periods of operation. Therefore, some mechanism for spreading the heat load more evenly over the plasma wall is highly desirable. The use of fueling and/or heating techniques to form (for example) detached divertor operation or radiative plasma boundaries may be possible.

Disruption avoidance/mitigation

The requirement that plasma-facing components (PFC's) must withstand disruptions is one of the major engineering drivers for the design of these components in tokamaks. The ability to avoid disruptions, or at least to modify their properties so that they will offer less damage potential to the PFC's, would greatly expand the ability to design a more cost-effective, reliable fusion device.

2.3 Needs for stellarators

Neutral beams for compact stellarators

The PDX/PBX neutral beam system, which will provide the neutral beam heating power on NCSX, is capable of full power pulses for up to 300 ms, and somewhat lower powers up to 500 ms. These pulse lengths should be sufficient for the early phases of NCSX operation. However, longer pulse capability of 3 - 5 seconds would constitute an attractive upgrade for the later years of the program. This would require replacing the present grids with ones with much more cooling capability, and replacing the ion dumps and scrapers with better cooled structures incorporating, in the ion dumps at least, hypervaportrons or swirl tube configurations. All of this technology has existed for many years, and variants of it have been used on numerous machines. However, engineering adaptation, especially of the source grids, will be required to integrate this longer pulse technology into the existing neutral beam system. It is likely that the ORNL beams

on MAST will encounter the need for longer pulses before NCSX does. Therefore, the most cost effective role for the U.S. may be to collaborate with the MAST people, and buy duplicates of whatever adaptation of JET PINI grids and beamline compatible sources they adopt.

RF for stellarators

The compact stellarator program is aimed at developing a concept for steady-state operation with no disruptions and low recirculating power, by combining tokamak and stellarator features. In the current reactor vision of compact stellarators, the plasma will carry a bootstrap current, but will not require a large current drive system for profile control as some other concepts do. However, to develop the concept through at least the proof-of-principle stage, we will likely need RF heating techniques as a research tool to vary profiles and control rotation. These control capabilities are important to the mission of the proposed stellarator PoP program in several ways: 1) understanding the influence of profiles on beta limits, 2) understanding what is required to obtain enhanced confinement, 3) clarifying the control requirements for next steps.

RF scenarios for compact stellarators are beginning to be studied, and one that has emerged is direct electron heating by high-harmonic fast wave. It does not require inside launch but it does involve a novel frequency range (~350 MHz), so we anticipate technology development needs in the area of sources, matching systems, and couplers.

2.4 Present Capabilities

Heating and Current Drive

Table I lists some of the desired capabilities in the heating and current drive areas. The present-day capability of the various H&CD techniques to deliver these capabilities is shown in Table I.

Table I. Present Heating and Current Drive Technological Capabilities

Desired Capability	Ion Cyclotron	Fast-wave (CRF)	HHFW (direct elec. damping)	IBW (directly/launched)	Mode-conv. H&CD	Electron Cyclotron	O-mode launch	X-mode launch	EBW	Lower Hybrid	Neutral Beam Inj.	Coaxial Helicity Inj.
Heating												
Heat ions		E+		E-							E+	
Heat electrons		E+	E	E-			E	E		E+	E+	
Heat on-axis		E+	E				E	E+		E+	E+	
Heat off-axis		E+	P	E-			E	E+		E+	E+	
Change power dep. profile during pulse		P					P ^g	P ^g			No	
Use small amount of port space		D ^b		D			E+	E+		E ^h	E	
Current Drive												
Drive current on-axis			E		E-		E	E		E+	E+	
Drive current off-axis			P		E-		E	E		E+		
Drive current near edge												E- ^e
Drive current efficiently		E-	E-				f	f		E+		
Change driven current profile during shot		E ^c					P	P ^g		E+		
Change CD efficiency during shot		P	P				P	P ^g		P		
Change CD direction during shot		P	P							P		
Momentum												
Drive plasma rotation		? ^a		E-							E+	
Control rotation velocity profile				E-							E+	
Advanced												
Produce and control transport barrier		P		P			P	P			E	
Stabilize plasma		E										
Lower loop volts/V-sec for startup		E					E+	E+		E	E	
KEY: E Experimentally demonstrated capability P Possible capability; needs tests D Technology development needed + Clear demonstration or superior results - Preliminary demonstration or inferior results	NOTES:											
	a Some evidence from C-Mod, inconclusive											
	b RF power flux through a port up to ~ 10 MW/m ² at present.											
	c Minority current drive (demonstrated) will change local q profile, but does not drive significant net current.											
	d Mode conversion current drive											
	e On small machines											
	f Results consistent with theory											
	g Requires steerable mirror in vacuum											
	h Demonstrated, but needs <i>cooled</i> long-pulse launcher development											

3 Heating and Current Drive

3.1 Neutral Beam Injection Technology

Larry Grisham (PPPL)

Appealing Characteristics of Neutral Beam Injection

There is a great body of data demonstrating the effectiveness of neutral beams in heating magnetically confined plasmas, and a smaller, but substantial, collection of experimental results showing its effectiveness at current drive. The two greatest technical advantages which neutral beam injection enjoys compared to some wave heating technologies are that the energy and momentum transfer from the beam particles to the plasma are well understood (with the exception of TAE effects), and that the coupling efficiency of neutral beams is relatively insensitive to plasma edge conditions. The great boon that follows from these two characteristics is that, in principal, all of the technology development can be done away from the plasma confinement device where it will be used, and likewise, the beam technology can in principal be fully tested before it is ever installed. Because most of the energy transfer occurs through two particle collisions (rather than collective effects, such as waves), and because the details of the edge conditions don't strongly alter the coupling, one can usually have confidence that a fully tested beam will behave as expected when injected into a plasma.

Status of Neutral Beam Technology

Generations of plasma confinement devices have employed neutral beams using positive ion beams as the precursors of the energetic neutrals. On almost all of the devices where these systems were deployed, the positive-ion-based beams have been the day to day workhorses for heating the plasma, and on most machines their reliability has been as good as or better than any of the other major systems comprising the fusion facility. Moreover, with only very few exceptions, positive ion based neutral beam systems have reached a large fraction (75 – 100 %) of their specified power capability within a relatively short time (1 – 3 years), and in at least one case have eventually far exceeded their power specifications.

Unfortunately, the efficiency of neutralization of positive ions declines rapidly with velocity once the ion velocity is well beyond the classical velocity of the bound orbital into which the neutralizing electron has to be transferred. For positive hydrogen isotope beams, this has the consequence that the neutralization efficiency starts declining rapidly at energies greater than 40 keV for hydrogen, 80 keV for deuterium, and 120 keV for tritium. Since the fractional energy components resulting from the dissociation of accelerated diatomic and triatomic ions are slower, and thus more efficiently neutralized, this means that at higher energies, positive ion beams not only become less energy efficient, but they also come to be progressively more dominated by their molecular dissociation components at 1/2 and 1/3 of the acceleration energy.

The solution to the unsuitability of high energy positive ions to the production of neutral beams has long been recognized, and much work has gone into developing negative ion sources and accelerators for heating and current drive on present and future large fusion devices. The binding energy of negative hydrogen ions is only 0.75 eV, so the cross sections for removing the electron to produce a neutral are sufficiently larger than those for inadvertently removing the remaining electron that the neutralization efficiency remains high (59-60%) and nearly constant across a range of many MeV.

Unfortunately, the relatively low 0.75 eV electron affinity of hydrogen means that it is much more difficult to make negative hydrogen ions than positive ones. Extracting a beam of negative ions from the source is complicated by the fact that the electrostatic field pulling the ions from the source plasma is much more effective at pulling the far more mobile electrons along with the beam. Propagating the negative ion beam through the extractor and accelerator structure is much more challenging than with positive ions, because the fragile negative ions are easily neutralized by gas streaming from the source. Ions that are neutralized before they have fallen through all of the accelerating fields and their associated lens effects can produce a large low energy tail on the beam, and these particles will also in most cases have more divergent trajectories. All three of these generic problems are aggravated in the case of large ion sources.

In practice, the developers of large negative ion sources have gotten around the first problem (difficulty of production) by introducing cesium into the sources. In some manner which is not well understood, and subject to considerable lack of agreement, the cesium enhances the extractable negative ion current density by a factor of several, and also reduces the co-extracted electron fraction, but it introduces important complications. The co-extracted electron fraction is further suppressed by the use of magnetic fields in front of and in the extractor grid, but these fields must also alter characteristics of the extraction sheath, making the beam optics more difficult to predict. The premature neutralization problem in the extractor and accelerator sets limits on the source operating pressure, and consequently upon the usable negative ion density.

The first generation of negative ion sources for fusion is now in operation on the JT-60U and LHD devices. As might be expected, the first use of an entirely new and more complicated technology has generated a succession of problems, although apparently less so with the LHD sources, which benefited enormously from having a large, well instrumented, full power test stand for five or six years prior to installation of a beamline on LHD. Due to schedule concerns, the JT-60U negative ion system did not have a test stand capable of testing anything close to a full size source at full power. This has made implementation on JT-60U much more difficult, because time for beam development is much more restricted on a flagship tokamak than on a dedicated test stand. Nonetheless, over the past three years much progress has been achieved in understanding and ameliorating such problems as time-dependent arc characteristics arising from cathode sheath impedance changes due to increasing cesium ion concentrations, spatial non-uniformity due to several possible effects, and excessive premature neutralization due to strongly varying pressures. The result has been that the beam divergence, grid heat loading, pulse length capability, and neutral power have gradually improved.

Despite these improvements, the reliability, predictability, and ease of operation of these sources is still much less than the fusion community is accustomed to in the use of positive ion based neutral beam systems. A small portion of this is due to the fact that the negative ion sources (particularly the JT-60U sources) operate at much higher voltages (500 kV is the JT-60U specification) than their positive ion counterparts. A much more pervasive problem, however, is the exquisite sensitivity of cesiated sources to even tiny water or air leaks. Although the manner in which the cesium enhances extractable negative ion current is subject to controversy, it is fairly clear that part of the effect derives from the reduction in surface electron work functions which occurs with of order half a monolayer coverage on metal surfaces. This cesium reacts aggressively with water or oxygen to form cesium oxide, which raises the work function and alters the surface characteristics because it is an insulator. Even very small water and air leaks which would result in no observable effects in positive ion systems seriously degrade the operation of cesiated sources.

Required Development for Future Applications

The JT-60U negative ion sources were designed to operate at 500 keV, and they can probably achieve this value if there is ever a period when the cesium oxidation is low enough to allow the sources to operate at high enough current densities for perveance match at 500 keV. They have so far operated as high as 400 keV, with much operation at 350 – 360 keV. Electrostatic acceleration of high current beams is generally thought to be feasible for energies at least up to 1.5 MeV, although this remains to be demonstrated. However, developing structures for accelerating high negative ion currents to high energies is an expensive endeavor for which the U.S. technology program is unlikely to find the resources in the current funding environment. One of the lessons which the difficult path of beam optimization on JT-60U has shown is that it is essential to perform full power developmental tests of neutral beam systems, and especially of ones incorporating new technologies, on test stands, rather on major fusion facilities.

A problem which is potentially more pervasively troubling, but which would require much less investment by the U.S. to explore, is the question of how to build a negative ion system with the operational robustness of positive ion systems. This essentially reduces to the question of how to either eliminate the necessity of using alkali metals in the sources or, less elegantly, how to nearly preclude the chance of oxygen or water reaching the negative ion sources.

The first option, finding ways of eliminating the need for cesium or any other alkali metal, would be an approach that should necessarily first be explored on a small scale, and an acceleration voltage of a few tens of kV would be adequate for determining the extractable beam. Thus, this would be a low cost program involving only small scale facilities, and could serve as a physically and fiscally viable way for the U.S. program to contribute to the development of a second, more robust, generation of negative ion sources.

The alternative option, of retaining cesium in negative ion sources, but excluding air and water from the contiguous leakable environment, may also be of a scale suitable to U.S. involvement, but would be less oriented towards source physics studies, and more toward developing waterless heat pipe technology appropriate for cooling requirements within the source and beamline, and also to finding the most effective way to combine secondary vacuums and nitrogen gas shrouds to ensure that, if anything does leak into the beamline, it's not air. Developing the heat pipe technology would probably require a test facility with some significant beam power. This could be adapted from existing surplus beamlines, but would probably cost more than would the low power test stand required for exploring how to make adequate current densities in cesium-free sources.

Due to funding constraints, the U.S. fell out of the first wave of fusion negative ion sources. Due to the same constraints, the U.S. cannot readily re-enter the field at the high power end of the beam system, which would require a large investment. However, the U.S. might well be able to contribute at the low energy end through exploring ways to produce much more robust systems, and this could provide us with an entry into the second wave of negative ion sources.

3.2 *Electron Cyclotron Heating Technology*

M. Makowski (LLNL)

Introduction

The physics of EC heating (ECH) and current drive (ECCD) is well developed in both experiment and theory. Experiments have been performed over a period of more than two

decades in mirrors, tokamaks, and stellarators. These experiments have demonstrated effective plasma heating, with reliable access to the H-mode. Experiments have also demonstrated current drive, with recent work on off-axis current drive demonstrating higher efficiency than expected. In other experimental work, plasma startup has been extensively studied, as well as control of sawteeth, ELMs, locked modes, and control of $m = 2$ modes and some types of disruptions. EC waves have also proven to be a useful tool for transport studies.

The fundamental properties of EC wave absorption and propagation lead directly to its application as an auxiliary heating and current drive mechanism in fusion plasmas. The electron cyclotron wave can be launched in vacuum, and it propagates directly into the plasma without attenuation or interaction with the edge plasma. The waves continue to smoothly propagate within the plasma until they encounter the resonance and are locally absorbed. Absorption is generally complete and without the formation of an energetic tail or other non-linear effects. This localized absorption property allows control over the deposition profile and lends this heating method particular flexibility:

- Plasma heating, for achieving H-mode access and ignition or sustainment of a burning plasma,
- Current drive, to supplement inductive current or control the plasma profile,
- Stabilization of MHD instabilities, such as ELMs, sawteeth, and neoclassical tearing modes,
- Breakdown, start-up assist, and wall cleaning.

Electron cyclotron is very attractive when used for bulk plasma heating as an extremely simple system results. More advanced functions, such as MHD stabilization, require more sophisticated systems. In this case, steerable injectors and feedback control systems are required in order to meet the requirements for the broad range of proposed current drive missions. Steerable injection also accommodates a wide variety of plasma equilibria, particularly in the case of advanced tokamak scenarios.

ECH Physics

The functions listed above (see also **Table I of Sec. 2.3 above**) can be provided over a wide range of conditions, including variations in toroidal field and density. For almost all conditions, including the low density, low temperature Ohmic phase of the discharge prior to the H-mode transition, the launched ECH power is fully absorbed in a single pass due to the large optical depth. Control of MHD activity in the plasma is possible as a result of the highly localized deposition. Local absorption in conjunction with modulation techniques has also been exploited for local transport studies.

ECCD has been achieved in several tokamaks using the up-shifted scheme with oblique low-field-side injection of either second harmonic X-mode (T-10, RTP) or fundamental O-mode (T-10).

The highly localized nature of heating and current drive possible with EC waves can be used to control MHD activity in a plasma and to tailor the pressure and/or current profile. For example, in a number of experiments the sawtooth instability has been stabilized through application of EC wave power with careful placement of the EC resonance relative to the $q = 1$ surface. Similarly, heating near the $q = 2$ surface has been shown to suppress $m = 2$ oscillations which can lead to a disruption. Local EC wave power deposition very near the separatrix can strongly decrease the frequency of ELMs in H-mode plasmas, leading to a significant increase in

confinement. Locked modes have also been suppressed through EC wave absorption. All of these applications require only a small fraction of the heating or transport power in the plasma, but all of them also need further physics development before they can be considered for use in next stage fusion devices.

EC wave power has been used successfully for preionization and startup in many tokamak experiments and is used routinely for plasma production in stellarators. It has been demonstrated that the peak loop voltage can be reduced by nearly an order of magnitude through injection of a modest amount of EC power in the ionization phase. During the preionization and startup phase EC wave power can greatly broaden the conditions under which reliable breakdown can occur. In this way, extremely stringent constraints on error magnetic fields and gas fill density can be reduced through EC wave startup assist.

Technology

High frequency, multi-megawatt systems are needed for the proposed applications, particularly off-axis current drive. Over the past decade, steady progress in the technology has been made and recently, sources and systems of the required power and pulse length are beginning to come on line. Several experiments have or plan multi-megawatt ECH installations (LHD, TCV, DIII-D, W-7X, and ASDEX). Power output up to ~ 1 MW has been achieved over a wide range of frequencies at pulse lengths up to 1 s. Also, depressed collectors have been demonstrated over the range 110 – 170 GHz. In short pulse operation, efficiencies as high as 65% have been achieved with a single stage depressed collector. Steady-state operation has been demonstrated only at 110 GHz and at a relatively low power (~ 100 kW).

Transmission technology is generally well developed and relatively mature. Two forms of millimeter wave transmission are in use: overmoded waveguide and quasi-optical. Both types of transmission have been demonstrated and are, or will be, employed in near-term, large-scale experimental applications (waveguide transmission on DIII, TCV, JT-60U, and LHD; quasi-optical transmission on W-7AS and W-7X).

Very high injected-power densities can be achieved with ECH technology; up to $100 \text{ MW}\cdot\text{m}^{-2}$ in principle. However, in an ignition device, due to engineering, shielding, safety, and geometric constraints, this value is reduced to the order of $10 \text{ MW}\cdot\text{m}^{-2}$, which is comparable to or in excess of the power density of other heating technologies.

Studies demonstrate that in order to completely realized the potential offered by electron cyclotron waves, a steerable injection system is required. At least one degree of freedom is needed (either poloidal or toroidal) and two are preferable. However, the design of such injectors remains an engineering challenge due to a combination of the severe environment and demanding requirements. The challenges are even greater in a burning plasma environment.

R&D Needs

The near term goal is clearly to validate physics. This requires prototypical large-scale systems. Fully industrialized aspects of the system, such as single- and multi-stage depressed collectors, high- T_c superconducting magnets, and solid state power supplies, do not need to be implemented immediately. These components are important relative to efficient system performance, but are unrelated to fundamental demonstration of the technology. Based on the above discussion, near-term (next 1 to 5 years) system requirements can be prioritized as follows:

- Reliable and robust sources in the range of 110 – 170 GHz

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- Minimum output power in excess of 1 MW, with a goal of 1.5 – 2.0 MW
- Development of steerable injection systems
- Development of single stage depressed collectors
- Development of efficient internal mode converters producing Gaussian output beams

With no new machine on the horizon, no solid requirements can be established in terms of frequency. However, “generic” development areas can be identified: more economical systems and systems capable of operating at two or more frequencies. Longer-term priorities concern themselves largely with improving the efficiency, economics, and performance of an ECH system for next stage and ignition devices. The goals here include

- Development of very long-pulse/CW tubes and hardening of transmission and injection components under long pulse operation
- System cost reduction through the development of
 - Higher efficiency gyrotrons (single and multi-stage depressed collectors)
 - Higher unit power (2.0 – 5.0 MW) sources
- Development of frequency tunable sources: step tunable gyrotrons and/or FEMs together with broadband transmission systems to deliver the power to the plasma
- Development of higher frequency sources as required

3.3 *Lower Hybrid Range of Frequency (LHRF) Technology*

P. T. Bonoli (MIT), S. Bernabei (PPPL), R. R. Parker (MIT), M. Porkolab (MIT)

The specific question to be answered for the lower hybrid RF area is:

What is the potential for and what advances will be required in lower hybrid profile control technologies to enable present, near term, and next step devices to meet their performance goals and ultimate research potential.?

In order to answer this question it is instructive to first identify the current profile control requirements of present day and future devices. One of the most important needs of both present day and future tokamak devices is a technique for controlling the off-axis current density, especially in the presence of an internal transport barrier (ITB). Ideally this technique must maintain $q_{\min} > 2$ and $(r/a)_{q_{\min}} > 2/3$ in order to insure MHD stable operation at the $q=2$ -limit and beyond. The current profile control technique must also be shown to be feasible at reactor relevant densities ($n_e > 1 \times 10^{19} \text{ m}^{-3}$) and for long pulse ($\tau_{\text{pulse}} > \tau_{L/R}$, where $\tau_{L/R}$ is the current relaxation time).

The scientific basis of LH current profile control is well-established. Recent experiments in JT60-U have demonstrated maintenance of a low density reversed shear configuration for 6 s at $I_p = 0.85 \text{ MA}$ and $B_0 = 2.0 \text{ T}$ with $(r/a)_{q_{\min}} = 0.6$ using 2.3 MW of off-axis LH current drive

power. In JET optimized shear plasmas, a broad q-profile with negative central shear was formed with moderate LH power (2 MW at 3.7 GHz) during the initial low density current ramp up phase ($n_e(0) = 1.5 \times 10^{20} \text{ m}^{-3}$). An internal transport barrier was also observed in the JET experiments with improved electron confinement and $T_e(0) = 10 \text{ keV}$. However as stated above, in the near term and for next step devices, LH current profile control must be demonstrated at reactor relevant densities with fully relaxed current density profiles.

The present status of LHRF technology can be summarized as adequate to meet the demands of present day experiments, and with some improvements will be able to fulfil the needs of next step devices. Presently, LHRF sources exist primarily as klystrons and are capable of providing up to 0.5 MW/unit (CW) at 3.7 GHz and 0.25 MW/unit (CW) at 4.6 GHz. One megawatt sources CW have also been developed at 1.8-2.0 GHz. A klystron operating at 5 GHz has been developed and achieved 0.85 MW at an efficiency of 65% for 15 ms pulse length. A short pulse klystron operating at 5.7 GHz has reached power levels of 4 MW in the EU. Power fluxes up to 8 MW/m² at 4.6 GHz have been coupled reliably through phased waveguide arrays known as "grills". Waveguide phasing can be varied electronically on μsec time scales so that the grill phasing can be changed rapidly enough to respond to most real time plasma variations. Real time response of the LHRF system to internal transport barrier formation is crucial in order to demonstrate control of the current profile in the presence of an ITB. The power spectrum coupled by a LH grill array is characterized by the refractive index of the LH waves parallel to the applied magnetic field ($n_{\parallel} = c/v_{\parallel}$). Current profile control applications in present, near term, and next step devices will require an effective n_{\parallel} range of $2 < n_{\parallel} < 3$. This range can be provided by conventional launcher designs with phased waveguide arrays. It will also be useful to incorporate into these launcher designs a "compound" spectrum capability where a fraction of the coupled power (< 20%) is characterized by low parallel phase speeds ($n_{\parallel} < 4$). This can be accomplished with either a separate launcher or a dedicated row (or rows) of waveguides within a single launcher.

For LHRF applications in next step devices, CW source development at 5.5 GHz and the 1 MW level will be required. Based on the present state of source development this goal can definitely be achieved. Higher source frequency is needed to avoid damping of the LH pump wave on fusion generated alpha-particles ($\omega/k > v$) and also to provide added margin against depletion of the LH pump wave due to onset of the parametric decay instability. In next step devices an adequate cooling design will also be required at the window location of the grill. The use of a quasi-optical grill design or hyperguide (fixed n_{\parallel}) may provide promising alternatives in a reactor environment. However further technological development is needed in this area.

3.4 Ion Cyclotron Technology

D. Swain (ORNL), J. R. Wilson (PPPL)

What do machines need to meet performance goals?^a

Ion cyclotron heating and current drive have been demonstrated in many fusion experiments. A listing of the present capabilities to meet the needs are shown in **Table I** in **Sec. 2.4**.

^a Heating and current drive *physics* needs of present and future machines will be addressed in other sub-groups.

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In order to meet future needs the present technology must be improved. *Incremental* development can provide sufficient capability/reliability, but significant improvements in power-handling capability would make IC a much more attractive for designers of future devices, since more power could be delivered through fewer/smaller ports.

The long-term goal of the IC program R&D is to develop reliable, advanced ion cyclotron heating and current drive systems that:

- Use limited port space, with a high power density launcher.
- Can run for long pulses, essentially steady-state.
- Can survive in a reactor environment.
- Can work with rapidly varying plasma loads.
- Can control plasma conditions by heating and current profile control.
- Operate over a wide range of density and magnetic fields (already demonstrated).
- Will heat either ions or electrons (already demonstrated).
- Can drive current on- and off-axis in high density discharges.

The *technology* program elements to develop these capabilities are described below. There are several possibilities [e.g., the use of high-harmonic fast-wave (HHFW), mode-conversion current drive (MCCD) and ion Bernstein wave (IBW)] for heating and/or current drive that need further *physics* testing to delineate their potential. These physics experiments are not addressed here, although their success could cause a significant shift in the priorities of the technology R&D program.

Present experimental status

Porkolab¹ has reviewed the experimental and theoretical results of ICRF physics through early 1998. In brief,

- Minority heating (especially H minority in D plasmas) has been demonstrated on several machines with good results. H-factors relative to ITER 89-P scaling >2 have been obtained. He³ minority heating works also, but not as efficiently (at least on JET) as H minority.
- Second-harmonic heating (at $2\omega_H$ and $2\omega_D$) has been observed in several machines. It is not an efficient heater during startup, since the single-pass absorption is proportional to ion beta. However, at higher beta absorption appears to increase as expected.
- Mode-conversion electron heating and current drive has been demonstrated for both on-axis and off-axis cases. Measured power deposition profiles are in good agreement with theoretical calculations.
- Fast-wave electron heating and current drive have been demonstrated on DIII-D. Driven current profiles were in good agreement with calculations, and the current-drive efficiency was observed to increase linearly with electron temperature, also in agreement with theory.
- Impurity toroidal rotation has been measured in several machines (JET, TFTR, C-Mod), but no conclusive comparison with theory has been shown.
- Tantalizing glimpses of the attainment of advanced operating modes with the addition of ICRF heating have been seen on several machines.:
 - Pellet-enhanced plasma (PEP) modes on JET and C-Mod that causes increased confinement in the plasma core.

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- Radiative improved mode (RI) were observed on TEXTOR and ASDEX-U, in which ICRH was used with edge impurity seeding to generate a well-confined plasma where virtually all the power was transferred to the chamber walls by radiation.
- Hot-ion H-mode operation in JET, obtained with ICRF central heating and a current ramp to provide reversed central shear, resulted in the highest neutron production observed in a D plasma. This result shows the effectiveness of combined electron and ion heating using ICRF.

Present experimental results for ICRF technology are also encouraging:

- Antennas have been operated at rf power fluxes through the antenna of $\sim 10 \text{ MW/m}^2$ for pulse lengths $\sim 1 \text{ s}$ in several machines.
- The recent implementation of ELM dumps (e.g., on ASDEX-U) and other circuits to cope with rapid changes in plasma properties have improved the ability of ICRF systems to operate in the presence of large ELMs and other transients.
- Real-time control of the phase between adjacent current straps (and hence control of the current drive efficiency) appears feasible with the use of decouplers. This technique has been demonstrated on DIII-D and TFTR.
- Antennas have been operated with voltages up to 45 kV in some plasma heating experiments (e.g., Tore Supra). However, most experiments operate in the 30–35 kV range. Limits on operating voltage are not understood at present.

R&D needed

Higher-power density launchers – There are two routes that can be taken. Both should be pursued.

- 1– Improve voltage and power-handling capability of present-day launchers by doing laboratory experiments to determine what is limiting in present experiments, devising ways to circumvent these limits, and demonstrating the improvements in operating machine(s).
- 2– Develop new launcher technology concepts. This is less clear-cut, but there may be potential for significant (factor of two) improvements in power-handling capability.

Long-pulse operation, reactor survivability – R&D is needed to sustain long-pulse, fully noninductive discharges using fast-wave and/or mode-conversion current drive to maintain current and/or current profile control. This is largely a physics experiment, but long-pulse technology capability must also be developed (e.g., to assure launcher survivability for $\sim 10^5 \text{ s}$ of plasma operation).

Present-day systems can operate over a wide range of plasma parameters, and I*** and other reactor design studies have indicated that survivability of IC launchers in a reactor environment appears feasible. However, testing of IC launchers in increasingly long-pulse and high-power conditions with increasingly robust plasmas will be needed to convincingly demonstrate this capability.

Control – In order to work with rapidly varying loads, control current-drive efficiency, and improve reliability, fast automated control systems must be developed that:

- detect arcs in the rf components and protect the system from their effects,

- deliver a large fraction of available rf power to the plasma, even during rapid changes in plasma loading,
- allow real-time control of current drive directivity while maintaining high heating power to the plasma.

¹ M. Porkolab et al., “Recent Progress in ICRF Physics”, to be published in *Plasma Phys. and Controlled Fusion*, 1998.

3.5 Helicity Injection for Current Profile Control

Michael J. Schaffer (GA)

Theory and Background

In a system of nested closed magnetic surfaces, the magnetic helicity H is the linkage of toroidal flux Ψ_T by poloidal flux Ψ_P , with $H(\Psi_P) = 2 \int \Psi_T d\Psi_P$. Integration is from the magnetic axis, and Ψ_T and Ψ_P are full fluxes (not per radian). Since the toroidal safety factor $q(\Psi_P) = d\Psi_T/d\Psi_P$, specification of $H(\Psi_P)$ and $q(\Psi_P)$ are equivalent, and both are equivalent to specification of the current profile in an axisymmetric torus. Viewing current drive as helicity control suggests “helicity injection” current drive techniques that might not otherwise be invented.

Helicity is a global topological property of a volume, not a local property. In an ideally conducting plasma ($E_{\parallel} = 0$) helicity is frozen in and neither dissipates nor diffuses. In slightly dissipative plasmas with nested closed magnetic surfaces H is dissipated at the local rate $2 \mathbf{E} \cdot \mathbf{B}$, or $2 \mathbf{J} \cdot \mathbf{B}$ in simple resistive plasmas. This is the usual Ohmic dissipation of flux.

In order to affect the internal current profile, helicity must be transported to the desired location. Helicity evolution is governed by a conservation equation, the general form of which is

$$\frac{dH}{dt} + \oint_{S(t)} \hat{\mathbf{n}} d^2x = - 2 \oint_{V(t)} \mathbf{E} \cdot \mathbf{B} d^3x \quad (1)$$

$S(t)$ is a (possibly moving) surface enclosing a plasma volume $V(t)$ of interest. The helicity flux vector must be defined so that both it and H are independent of the gauge of the magnetic vector potential. This is nontrivial, is commonly overlooked and might not have yet been done correctly for the completely general case. One derivation gives

$$= 2 \Psi_T (d\Psi_P/dt)/S(t) \hat{\mathbf{n}} + \mathbf{A}_o \times \mathbf{A}_o / t + 2 \mathbf{B} \cdot \mathbf{E} \quad (2)$$

Here \mathbf{A}_o is a vector potential generating the magnetic field normal to the surface S , Ψ_T is a scalar electric potential, and \mathbf{A}_o and Ψ_T are evaluated in the moving frame of $S(t)$. The first right hand term of $\hat{\mathbf{n}}$ contains conventional inductive drive, $2(\text{toroidal flux})(\text{loop voltage})$, which offers no new control. However, it enables “oscillating fluxes current drive” (OFCD) helicity injection when Ψ_T and $d\Psi_P/dt$ are oscillated in phase. The second RH term corresponds to twisting magnetic flux tubes, originating on one side of the surface, penetrating the surface and twisting fluxes on the other side, thereby changing the helicity. The third term describes potential-charged flux tubes penetrating S . It enables DC helicity injection (DCHI) by polarized electrodes in contact with plasma, e.g. when S is a scrape-off layer (SOL). Processes that distort the surface without opening it do not transport by the last two mechanisms, because $S(t)$ can always be defined as the unbroken surface. Therefore, magnetic surfaces must be broken to inject helicity by these “surface-breaking” terms. However, open magnetic surfaces cause rapid energy loss

from hot toroidal plasmas, and such helicity injection current drive concepts might be incompatible with toroidal fusion reactors.

Injection of small, helicity-containing spheromak plasmoids into a larger plasma drives current. The plasmoid injection speed must be fast to penetrate the main plasma magnetic field to a desired depth, where the plasmoid reconnects with the main plasma and deposits its helicity. One might inject spheromaks to different depths to tailor current profiles. Small plasmoids would generate only localized magnetic field stochasticity as they passed inward, and then only for a brief time, so perhaps the global confinement would not be seriously compromised.

Div $\mathbf{E} \cdot \mathbf{B}$ implies the existence of a dynamo in the form of an average $\mathbf{E} \cdot \mathbf{B}$ on a toroidal surface, whose mechanism is excited either by instabilities and turbulence generated internally by the plasma or else by externally launched waves. For profile control, $\mathbf{E} \cdot \mathbf{B}$ must be localized to a desired set of surfaces. Some waves transport helicity and can deposit it locally, but the concept of wave helicity has not been very productive, in part because wave absorption processes can also dissipate much of the injected helicity. Wave current drive is best treated by detailed wave calculations.

Experimental

OFCD has been applied experimentally to RFP's and tokamaks. It was first tested on the ZT-40M RFP. There it raised impurity content, and it was difficult to discern the current drive effect with the available diagnostics. OFCD will be tested again on the MST RFP. Classically, the oscillating fields do not penetrate deeply into hot plasmas. In RFP's helicity can be transported farther inward by surface-breaking terms, due to instabilities generated as the plasma stays near its relaxed state. OFCD was also tested on the Encore, TEXTOR and DIII-D tokamaks. Unless tokamaks have some unknown anomalous helicity transport process, one expects OFCD to work only on the edge. DIII-D had the strongest apparent result. The average loop voltage dropped by 0.09 V, which was ~30% of the drop expected from the OFCD injection rate. Confinement remained good and impurities low, and the limited diagnostics (in 1988) suggested anomalously rapid current diffusion. B_T was modulated on ZT-40M and the small Encore tokamak, but on the larger tokamaks I_T was varied by modulating the plasma cross section area in a fixed B_T .

DCHI has been applied to spheromaks, RFP's and tokamaks. The CTX and SPHEX spheromaks were fully driven by DCHI applied to a poloidally diverted SOL. The edge injection achieved current drive throughout the plasma, but it was accompanied by strong MHD activity, as expected theoretically. Small DCHI electrodes were inserted slightly into the edge of the HBTX, OHTE and MST RFP's in order to modify edge currents. The drive was large enough in MST to modify the edge current sufficiently to improve plasma stability, reduce magnetic fluctuations and increase global confinement. DCHI was applied to the diverted SOL of the DIII-D tokamak and the HIT spherical tokamak (ST). In DIII-D no clear evidence of DCHI was discerned, but edge current profile diagnostics were not available at the time. In contrast, HIT and HIT-II are fully sustained by DCHI. DCHI will be studied in the new SSPX spheromak and NSTX ST experiments, where helicity injection physics will also be investigated.

Single spheromaks were injected into the Encore, STOR-M and Tokamak de Varennes tokamaks. A brief current drive was observed, but cooling by the cold plasmoid caused an overall volt-second loss. Injection did not harm confinement in TdeV.

Technology Issues

The electrodes for DC helicity injection are necessarily plasma facing components and *de-facto* limiter or divertor targets. In a fusion reactor the electrical insulation requirement will complicate the already daunting target design problem. The required current densities are large and can be met at a “cold,” nonthermionic cathode only by highly recycling, cold plasmas. Alternatively, a suitable thermionic surface must be developed. The current flow will reattach the radiative “detached” divertors that are usually preferred for exhaust power reduction. There might be no divertor design that satisfies both the helicity injection and plasma exhaust requirements for reactors. DC helicity injection might be most useful for noninductive startup of toroidal current-carrying plasmas, especially where conventional inductive startup is difficult or impossible. In this application the loss of plasma energy via broken magnetic surfaces is tolerable, up to a point, and electrodes used only for startup can be separated from the main divertor targets.

While OFCD avoids plasma contact with electrodes, the reactive power is large (power supply issue) and coil currents must be modulated (possibly incompatible with superconducting coils). The frequency is low, $\ll 1$ Hz in reactors. The oscillating plasma parameters and shape will modulate the heat and particle fluxes to divertor targets, making their design more demanding than usual. OFCD induces electric currents in all conducting components in the reactor.

Spheromak injection is the only helicity injection concept to date that offers the conceptual possibility of current profile control. It uses the same technology as spheromak injection for fueling, §4.2 below. The most difficult issues are repetition rate, impurity control and gas control.

Summary

Helicity injection current drive has been demonstrated in some toroidal systems. Its use for profile control has been limited to plasma edges so far, but injection of helicity-carrying spheromaks can in principle control interior current profiles. There is a basic theory, but little can be done quantitatively at present. It is very important to identify helicity injection techniques that minimally open magnetic surfaces. Basic physics is more urgent at present than technology development.

4 Fueling

Fueling system functions for reactors and similar scale fusion devices are:

- to provide hydrogenic fuel to maintain the plasma density profile for a specified fusion power,
- to replace the deuterium-tritium (D-T) ions consumed in the fusion reaction,
- to establish a density gradient for plasma particle (especially helium ash) flow to the edge, and,
- to supply hydrogenic edge fueling for increased scrape off layer flow for optimum divertor operation.

Additional functions related to fueling are:

- to inject impurity gases at lower flow rates for divertor plasma radiative cooling and wall conditioning,
- for plasma discharge termination on demand by either by injection of massive gas puffs or liquid jets/solid pellets.

4.1 Pellet Injection

M. Gouge, L. Baylor (ORNL), P. Parks, T. Evans (GA)

Pellet injection is an established plasma refueling method, and its development has spanned the last 20 years or so. This technology has enabled plasma experimental activities carrying important new results for fusion research. Recent results with the direction of pellet launch along the magnetic field gradient have led to a deeper understanding of the ablation and relaxation physics associated with the penetration of pellets. The basic principle of pellet injection is to accelerate pellets of frozen hydrogen of millimeter size to velocities in the range 100 m/s up to ~4 km/s and to inject them into the plasma. The scrape off layer and, under typical injection conditions, the boundary region, can be penetrated by the pellet, depositing the fuel directly into the plasma core. From the experimental database, scaling relations have been determined which predict the penetration depths into plasmas ranging from current devices to those at reactor dimensions. Pellets are an efficient way to refuel plasmas and innovative concepts like isotopic tailoring to reduce the in-vessel tritium inventory and throughput have been proposed.

In addition to their role in refueling, pellets also have other applications. They can strongly alter particle transport and create transport barriers. For example, the pellet-enhanced-performance (PEP) mode in JET used the initial cooling of the central plasma to create hollow current profiles, now known to produce negative magnetic shear in the core which reduces instabilities and increases confinement. The subsequent density peaking in the core, in combination with the improved confinement, is substantial. Similar experiments are in progress on DIII-D using pellets launched from both the high- and low-field-side. The pellet launch geometry for recent DIII-D pellet launch experiments is shown in Figure 1. Pellets can be injected in four locations: outside midplane, vertically inside the major radius, inside launch at ~ 45 degree angle and inside midplane.

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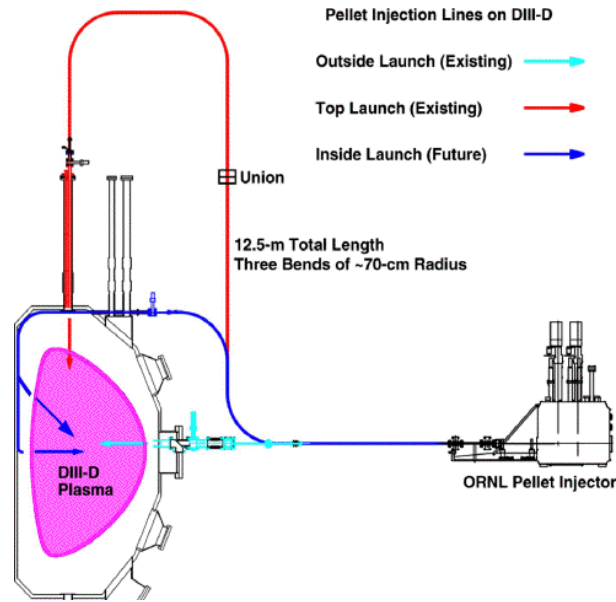


Figure 1. Pellet launch locations on DIII-D.

Pellets injected from the outer horizontal midplane (low field side) show a large discrepancy in the measured fueling efficiency and mass deposition profiles from pellet ablation theory, while the penetration depth compares favorably with theory. An apparent outward displacement of the deposited pellet mass is observed and hypothesized to occur from B and curvature induced drift effects. Injection of pellets inside the magnetic axis from a vertical port and inner wall ports using curved guide tubes has been employed on DIII-D to investigate these effects. The resulting density profiles show pellet mass deposition well inside the expected penetration radius, suggesting that a drift of the pellet ablatant is occurring toward the low field side edge of the plasma (Figure 2 below).

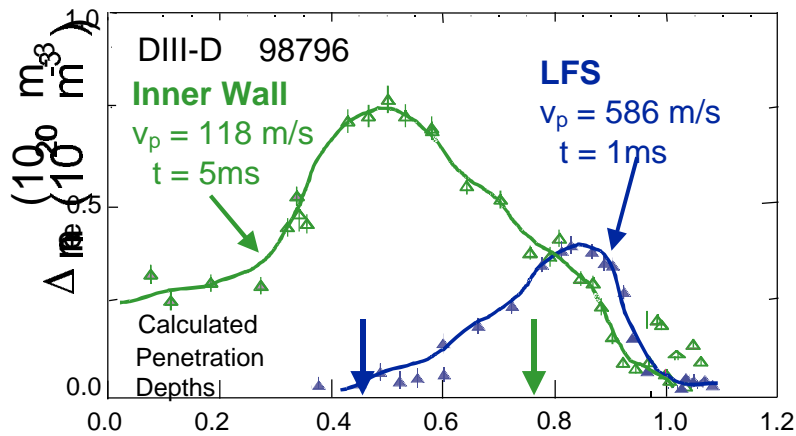


Figure 2. Plasma density increase from a pellet launched from the inner wall (high field side, green) launch compared with low field side launch (blue).

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Modeling of the radial drift of the pellet ablatant by P. Parks, GA is underway. The formation of highly peaked density profiles with pellets injected from the high field side is possible at higher heating power than is possible from pellets injected from the low field side. These peaked density targets are formed for PEP mode experiments where internal transport barriers are formed in conditions where $T_e \gg T_i$. Pellets injected into rotating plasmas are found to induce changes in the toroidal rotation and radial electric field. Small shattered pellets have been used to purposely trigger small ELMs in an attempt to extend the duration of high performance ELM-free VH-mode plasmas. The pellets are found to trigger ELMs in ELM-free conditions and increase the ELM frequency when injected in ELMing H-mode plasmas.

The potential of pellets for density profile shaping has been demonstrated. A strong central peaking could expedite plasma ignition during the start-up phase. During continuous operation in the H-mode, controlled release of ELMs by shallow particle deposition might apply. The ELM intensity and their power load on the divertor—a main concern for tokamaks—can be reduced by increasing the pellet-induced ELM rate. Profile shaping might also be advantageous for helium ash removal from the plasma.

There has been much progress in the physics and technology of pellet fueling. From first experiments with a single, small hydrogen pellet at 100's of m/s the technology has advanced to pellets of all the hydrogen isotopes in longer pulse length systems, with inventories up to about 1000 pellets and speeds in the km/s range. Table 2 below provides a summary of the status of this technology.

Parameter	Status 1978	Status 1999
Pellet isotope	hydrogen (H)	H/D/T
Pellet size	1-few mm	0.5-10 mm
Pellet inventory	1-few	1-1200
Pellet feed rate	N/A	0.26 g/s
Pellet speed	100's m/s	1000's m/s
Reliability	unknown	sufficient

Table 2. Progress in pellet technology 1978-1999.

Deuterium and tritium pellets up to 10 mm in size (see Figure 3 below) have been extruded at rates up to 0.26 grams/sec (for short pulses only); this pellet size and feed rate is sufficient for fueling fusion reactors at the gigawatt power level.

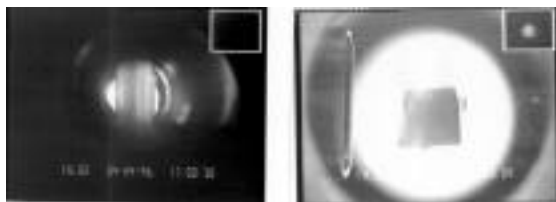


Figure 3. Pure tritium extrusion and pellet.

Pellet speeds up to about 1.2 km/s have been achieved with centrifuge accelerators for small (1 to 2 mm) pellets: and, single-stage light gas guns have accelerated larger pellets (4 to 10 mm) up to speeds in the 1- to 2-km/s range. Two-stage light gas guns operate repetitively with 3-mm pellets at 1 Hz and speeds in the range of 2.5 to 3 km/s. Single 4-mm pellets have been accelerated to speeds above 4 km/s with this technology. This maturing of pellet fueling technology has benefited from a pellet fueling system on virtually every major tokamak and stellarator experiment in the past two decades. The technology to deliver intact pellets at the highest possible speeds around curved surfaces (guidetubes) is under development. This is a complex issue and depends on the pellet speed and temperature (strength) as well as the guide-tube radius of curvature, its diameter relative to the pellet size, and its cross-sectional shape. The speed dependence of penetration for high-field-side or vertical launched pellets is not known, and it may turn out that incremental speed capability from 1–2 km/s to 3–5 km/s (for vertical launch) would be attractive. Nearly continuous pellet feed formation has been demonstrated (~1-hour pulse length) by a new screw extruder concept developed in the Russian Federation to produce 2-mm hydrogen pellets. This needs to be extrapolated to deuterium and tritium feed and larger pellet sizes using this technology or variants such as gear or double-screw extruders. Centrifuges need to be extended to pellet sizes in the 4- to 10-mm range and the allowable speeds quantified. The speed will be limited by the strength of the frozen hydrogen, which is being investigated in the development of high-field-side pellet launch technology. Engineering issues include reliability for long pulse (100 to 1000s of seconds) fueling systems in the 95 to 99% range, minimization of the hydrogenic (tritium) inventory and closed cycle reprocessing of the pellet erosion and/or propellant gases. For high field side launch major research and development issues include:

- quantify guidetube dimensions (radius of curvature, inner diameter, shape, surface finish) and pellet properties (dimensions, shape, isotope, temperature) for maximizing the delivered speed of intact pellets into the plasma,
- related to this, understand the pellet speed dependence of pellet penetration and, more importantly, mass deposition and fueling efficiency, for pellets launched inside the magnetic axis to take advantage of the favorable magnetic gradient,
- based on the speed requirement, which is expected to be quite modest, optimize pellet acceleration techniques and guidetube delivery systems. It may be that a mechanical pellet punch (with or without some modest amount of light propellant gas) could be used as the pellet driver. This would remove an issue with the use of pneumatic injectors on reactor-scale devices, namely the reprocessing of large amounts of tritium-contaminated propellant gases.

4.2 Deep Plasma Fueling Using Accelerated Compact Toroids

David Hwang (UC Davis), Harry McLean (LLNL), Roger Raman (Univ. of Washington)

Introduction

Two types of CT accelerators have been developed for fueling applications. The first produces "spheromak" CT's in a magnetized plasma gun and is the more developed of the two. The second technique produces "field reversed configurations" (FRC's) by purely inductive methods, which might reduce entrained impurities. Spheromaks contain roughly equal toroidal and poloidal magnetic fields and have moderate betas. The FRC's, or frCT's, have negligibly small toroidal fields and operate at betas above 50%.

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Recent design studies for the ITER tokamak show that there are several advantages and requirements for a deep plasma fueling system. First, from a design code analysis, it is found that deep fueling can lead to more efficient reactor operation by lowering the ignition margin H factor,¹ and second, from simple particle confinement estimates, deep fueling can greatly improve the burn fraction of the reactor fuel. Recent deep fueling studies on the medium size TdeV tokamak, using spheromak-like compact toroids (SCT's), have demonstrated over 30% increases in particle inventory without disruptive effects on the discharge². Moreover, the most recent TdeV and STOR-M experiments show that SCT's may be able to initiate H-mode behavior in the discharge^{5,3}. The recent experiments at University of California, Davis have demonstrated that repetitive operation of a SCT injector fueling a small tokamak (DDT) can be achieved at 10 kJ energy levels using passive, saturable-core, switching inductors.⁴ Unlike other fueling methods such as gas puffing and pellet injection, the penetration mechanism of the SCT into a magnetized plasma is the displacement of the magnetic field energy. An added attractive feature is that it can be used to inject plasma into a vacuum equilibrium field structure thus useful for other helical confinement devices.

Various "Spheromak" CT injection experiments are referenced below^{4, 5, 6}. FRC's suitable for refueling large tokamaks have been produced and studied at the University of Washington.⁷

At the recent advanced fueling workshop,⁸ it was pointed out that the foremost requirement for a fusion reactor application is that both the fueling and exhaust systems must be continuously operated⁹. In other words, the repetitive nature of the fueling system is an essential criterion for a reactor relevant system. Another requirement for a rep-rated SCT system concerns with the injector electrode geometry and material. The electrode geometry impacts the overall injection efficiency. The electrode material and operation will also influence the impurity level in the SCT that is delivered to the reactor, while wall-erosion will limit the electrode lifetime. Over the past few years, the SCT injection program has been focused on the demonstration of the basic principle of the injection of SCT's into a toroidal plasma. Up to now, in most of these experimental goals can be achieved using existing technology. However, in order to extrapolate the results of the small experiments to a reactor relevant system, several technological issues need to be addressed.

Technology Development Issues

There is a large amount of overlap between CT fueling and other fueling methods as regards to the required technology development. Listed below are important issues for CT fueling, but which also may be covered by development for other technologies.

Pulsed power systems

Electrical acceleration process of the SCT injector requires the development in the pulsed power areas, especially in energy storage and switching. The critical needs are in the areas:

- 10 Hz repetitive operation high-energy storage capacitors.
- 10 Hz repetitive high voltage/high current switching.

Demonstration of pulsed power systems operating at 1-10 kJ, 100-200 kA, 10-20 kV with 10 Hz repetitive rate needs to be carried out. Since many capacitors will be used in a bank, reliability is critical. The present method used is to operate at much less than the nameplate ratings. Investigation into the optimal switching system should emphasize on reliability and lifetime issues.

Gas handling system

Since tritium will be eventually used in future reactors, the gas handling system will require extremely high reliability and efficiency of operation with a small gas recirculation ratio. The three areas of gas-handling concern are:

- a. Fast pulsed valve for SCT initiation and best gas utilization efficiency.
- b. Gas exhaust in formation region. (Possibly mitigated by better gas valves)
- c. Tritium handling system in cooperation with other fueling methods.

Development is needed in injector valves that are fast enough and provide an adequate quantity of gas (the shape of the gas pulse) while operating at a rep-rate of 10 HZ.

Materials and Heat removal systems

Even with additional optimization to the electrical efficiency of the injector through careful pulsed-power design, heat removal from the surfaces of the CT injector will require active cooling and careful selection of materials. The areas requiring demonstration are:

- a. Electrode cooling at high rep-rate.
- b. Impurity control
- c. Material selection & conditioning.

Technology development should take advantage of past results from other applications on electrode erosion reduction.

Near-Term Technology Developments:

A technology program for SCT fueling should encompass the major technological areas listed above in order to meet the needs of present experiments

- a. Engineering demonstration of high energy, 10 Hz, repetitive pulsed power system.
- b. Electrode material development and selection.
- c. Fast gas valve development.
- d. Electrode heat flux removal.

Long-term Technology Developments:

Longer term developments concern the needs of reactor applications.

- a. Continued electrode material development.
- b. Tritium compatibility.
- c. Radiation compatibility.

¹ Perkins, L. J., et. al, ITER report 22

² Raman, R et. al. Nuclear Fusion **37**, No.7, 1997 (p967)

³ Raman R., Paper presented at US/Japan Workshop on Advanced Fueling at LLNL on Dec 2, 1997, Livermore Ca. (Proceedings published see Thomassen, et al, *Fusion Technology*, 33, (1998) p.86)

- ⁴ H. S. McLean et al., *Fusion Technology*, 33, (1998) p.252
- ⁵ C. Xiao et al., "Improved confinement induced by tangential CT injection in STOR-M," *Proc. 17th IAEA Fusion Energy Conf.*, IAEA-F1-CN-69/EXP3/02 Yokohama, Japan (1998).
- ⁶ N. Fukumoto, M. Nagata, T. Uyama et al., "Compact torus injection experiments on the H.I.T. test stand and the JFT2-M tokamak, *Bull. Amer. Phys. Society*, **42** (1997) 1961
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4.3 Gas and Impurity Injection

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Present and past generation tokamaks and helical devices have relied heavily upon gas puffing for building up and sustaining the plasma density, and a great deal of success has been experienced using this method, in which the plasma particle sources are strongly localized at the plasma surface. There is growing evidence, however, that gas puffing alone may not be satisfactory in larger, steady state plasma confinement devices with relatively thick and dense scrape off layers, and that some form of "deep fueling", where the particle sources are located near the plasma center, will be desirable and perhaps mandatory. This can be seen in Table 1 below where the fueling efficiency of gas and pellets is compared. It can be seen that the fueling efficiency of gas injection, especially in larger devices, can be an order of magnitude lower than pellet injection. Nevertheless, gas injection is straightforward technology and relatively simple to make tritium-compatible and is the fueling method of choice for influencing scrape-off layer and divertor region parameters. *For larger devices, gas injection has relatively modest influence on the plasma density and pressure radial profiles.* It does, however, with recycling sources, establish the boundary condition for the plasma and neutrals edge density.

Device	Gas Fueling Efficiency (%)	Pellet Fueling Efficiency (%)	Remarks
ASDEX	20	30-100	high density
PDX	10-15		high density
Tore Supra	1	30-100	ergodic divertor for gas fuelling
JET	2-10	20-90	active divertor
TFTR	15		low density DT
ASDEX-U		8-50	

DIII-D	10	40-100	active divertor
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Table 1. Tokamak fueling efficiency

Impurity pellet injection systems (typically small lithium or carbon pellets) have been developed for wall conditioning and plasma diagnostics. More recently, there has been interest in repetitive impurity pellets to foster enhanced radiation in the outer plasma and divertor regions and large (“killer”) pellets for a controlled, preemptive plasma shutdown in anticipation of a major disruption. These systems typically operate at room temperature or higher cryogenic temperatures, but require similar technology for pellet feed and acceleration as are used on H/D/T pellet fueling systems. For disruption mitigation, studies of massive He gas puffs into DIII-D (Tom Jernigan et al.) resulted in peak halo current reduction up to about 50% and also a reduction in the toroidal spatial nonuniformity. In a series of experiments with Ne, Ar and methane pellet injection into DIII-D (Todd Evans et al.), it was observed that peak halo current amplitudes were reduced by up to 50% in triggered VDEs with both neon and argon killer pellets. Halo current toroidal peaking factors were reduced from 3 to 1.1 for these experiments. Cryogenic liquid jets are also under development (Paul Parks, GA et al.) for disruption mitigation. Low Z impurity pellets (e.g. LiD) may be option if there are no or modest runaway electrons generated.

Gas or pellet impurity injection systems running continuously at a fraction (few-10 %) of the hydrogenic fuel input will be needed for burning plasmas to control the amount of plasma power radiated in the plasma outer boundary. This can reduce the directed heat loads on divertor surfaces. Thus this is, essentially, profile control of the plasma energy content to spread it out over a much larger surface area.

The technology for impurity gas injection is mature, although for specific applications the response time of the system (which depends on gas species) will need to be quantified and it may be a requirement for the dosing (control) valve to operate in high background magnetic and nuclear radiation fields. Major R&D issues for impurity pellet injection include development of steady-state pellet production and feed hardware optimized for the pellet material (i.e. lithium, carbon, nitrogen, argon...) and, for killer pellet injectors, high reliability for a single large pellet or liquid jet on demand.

4.4 Disruption Mitigation

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This section divides naturally into four subsections: detection, thermal energy mitigation, magnetic energy mitigation, and a summary of technology development needs. Detection is clearly the most vital component since the whole object is to minimize the deleterious effects of disruptions. However, many of the mitigation techniques may require recovery time afterwards, so false alarms are clearly to be avoided. Thermal energy mitigation has been the area with perhaps the most success to date with several techniques showing promising results experimentally. Finally, developing techniques for mitigating the effects of the tokamak’s poloidal magnetic energy decay requires more work and probably an integrated response of multiple tokamak systems.

Disruption Detection.

Disruption detection is the most critical component of any disruption mitigation scheme. The ideal detection method would operate quickly enough to allow a staged response of the

tokamak's control and heating systems to the plasma disruption, beginning with the most benign mitigation technique appropriate to the situation following, if necessary, with more drastic techniques such as "killer pellets". At the present no generalized detection technique has been developed that is appropriate to all (or even most) tokamaks.

Several groups have been involved in the exploration of techniques of disruption detection on a wide variety of tokamaks. Parameters which have proven useful guides include n_e approaching the Greenwald limit, q values close to 2, $n_N > 2.5$, values above which eddy current stabilization of vertical instabilities is no longer effective, MHD mode locking, etc. In conjunction with the project formerly known as ITER a database of disruption parameters has been established. It is hoped that this database will aid the development of effective detection methods which may be generalized to future devices. Meanwhile, existing experimental programs continue to develop their own detection methods using locally developed databases and such techniques as neural nets to evaluate their own tokamak data.

Thermal Energy Mitigation.

Assuming the tokamak's position control system (and MHD and heating feedback systems, if appropriate) are unable to cope the impending disruption, the first line of defense is probably some method of dumping the plasma's thermal energy. Several techniques have proven effective in existing devices. These include impurity pellets, impurity doped pellets, and massive, low-Z gas puffs. All of these have proven capable of dumping the plasma thermal energy in less than 1 ms through enhanced radiation. Other promising methods, such as a low-Z (e.g. D_2) liquid jet injection, are under development.

In tokamaks with very large plasma currents such as ITER, the use of impurity atoms to enhance the radiation will increase the likelihood of enormous runaway electron currents by the collisional avalanche mechanism. Such runaway currents can lead to very localized heating and destruction of the plasma facing components (PFC). For this situation the best mitigation technique would probably be to rapidly increase $n_e > 10^{21}$ with low-Z atoms (D_2). This eliminates bound electrons in the impurity ion/atoms as targets and provides sufficient density to damp out any runaway electron seed. Techniques which may accomplish this include very large deuterium pellets, massive deuterium gas puffs, and the liquid deuterium jet. The large pellets may prove very damaging to the first wall in cases of miss-fires if high speeds are required, and the penetration of a massive gas puff into an ITER class device is unknown at present. The liquid deuterium jet may prove the best compromise if technologically feasible.

Magnetic Energy Mitigation.

Even with the thermal energy dissipated, the poloidal magnetic energy remains which must be removed on an inductive time scale. If a vertical displacement event (VDE) is allowed to continue, this energy will be dissipated by halo currents which flow through the outer edge of the plasma into the PFC. While the bulk of this energy ultimately ends up as radiation, the local contact points are subject to severe heating. An even greater problem is the force generated on the PFC by the interaction of the poloidal component of the halo currents and toroidal magnetic field. These forces can produce potentially severe damage to large and/or high field tokamaks as has been demonstrated in such diverse experiments as JET and Alcator C-Mod. The best technique for dealing with this problem is not obvious. If the tokamak control system is robust enough, holding the plasma centered or at least away from the most sensitive parts of the inner wall while the current decays is probably the best solution. Failing this, the rapid thermal dissipation techniques like impurity pellets or massive gas puffs have proven capable starting the

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current decay before the plasma has contacted the PFC, thus minimizing the halo currents. Again, this points to the necessity of a good disruption detection method coupled with a flexible response from all the tokamak systems.

Technology Development Needs

The development needs for disruption mitigation follow directly from the preceding sections. In the near term disruption detection and develop of a low-Z mitigation techniques such as liquid jet injection need increased attention. The development of detection techniques in particular need a comprehensive, broad based effort. Techniques and procedures can proceed in parallel with existing tokamak experiments with virtually no interference between the two. The development of low-Z mitigation techniques is another parallel development which does not interfere with existing tokamak experimental programs. In the longer term the final area requiring increased effort is the integration and testing of the mitigation techniques into the tokamak control system. At the present, rapid response of the control system is the only method believed to be able to minimize the magnetic energy effects on the PFC. This last area, of course, will require either additional tokamak experimental time or the displacement of other experiments. Finally innovative ideas in this area need to be encouraged.