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Technology and Physics in the Tokamak Program

The Need for an Integrated, Steady State R&D Tokamak Experiment

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Technology and Physics in the Tokamak Program:
The Need for an Integrated, Steady State R&D Tokamak Experiment

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I. Introduction – The Steady-State Tokamak Experiment (STE)

The Steady-state Tokamak Experiment is a proposed superconducting-coil, hydrogen-plasma tokamak device intended to address the integrated non-nuclear issues of steady-state, high-power tokamak physics and technology. Such a facility has been called for in the U.S. program plan for the mid-1990's, and will play a unique role in the world-wide fusion effort. Information from STE on steady-state current drive, plasma control, and high power technology will contribute significantly to the operating capabilities of ITER, and to the development of controlled thermonuclear reactors.

We propose consideration by the U.S. fusion community of the STE project, an integrated fusion research and development tokamak that will fill a critical role in the world-wide effort to develop practical thermonuclear power systems.

The STE tokamak is intended to address the physics and technology issues associated with steady-state operation under conditions of energy density and flux appropriate to fusion reactors. In this role, the STE complements the proposed CIT ignition tokamak, which is intended for study of short-pulse ignition dynamics and α -particle effects. The combination of these two programs will address the issues of fusion physics that remain, and will make a major impact on the definition and resolution of the technological problems of fusion reactors.

By the end of 1990, the world fusion programs expect to decide on the construction of a large international experimental reactor, ITER, which would begin operation around 2000. The overall objective of ITER is demonstration of the scientific and technological feasibility of fusion power, with the capability of controlled ignited burn and steady-state operation. The present aggressive ITER schedule means that STE will not affect the basic ITER design, but it will provide major support for the development of operating techniques, particularly those relating to current drive and plasma shape and profile control. It will also be the principal facility for the development and testing of high-power, steady-state components for ITER.

The STE project is a necessary step for the U.S. fusion program. Present experiments, in the U.S. and abroad, are examining many facets of the problems of plasma confinement, heating, stability, and current drive. Included among these are the major superconducting-coil devices, Tore-Supra and T-15. The STE will integrate the results of these programs, and add the critical aspect of steady-state operation.

The Technical Planning Activity study,¹ building on the Magnetic Fusion Program Plan of the U.S. Department of Energy, has identified the need for an advanced-geometry experiment to "[confirm] the basis for tokamak reactor viability and [to contribute] to the data base needed for configurational optimization" and a steady-state experiment to "confirm the basis for the introduction of steady-state capability into the long-burn facility and [to] develop the data base needed for optimization of the current-drive technique." The STE satisfies both requirements, and addresses the further TPA concern that "[t]he principal "composite" issue for the tokamak will be to maintain a high-beta equilibrium that is satisfactory from the point of view of current-drive efficiency and effectiveness of impurity control techniques." The role of STE in the fusion program is illustrated in Fig. 1.1, taken from the TPA report.

The focus of the STE program will be on combining three closely related lines of research and development associated with the physics and technology of steady-state operation at high power density. First is the need to develop efficient, noninductive plasma current drive techniques suitable for reactor conditions. Second is the need for a realistic test facility for the development and demonstration of high power plasma interactive components. And third is the need for development and proof of methods of plasma control and optimization, through shaping, boundary condition control, and pressure and current profile control within the plasma. Present programs are making specific contributions to each of these areas. The STE will build on these results, emphasizing the simultaneous demonstration of all non-nuclear aspects of steady-state, high power operation.

An important example of the role of STE is the development and demonstration of effective techniques for noninductive current drive. A number of current drive experiments are now, or will soon be carried out on the DIII-D, MTX, and TFTR devices using energetic neutral beams and electromagnetic waves in the lower hybrid, electron cyclotron, and other characteristic frequency ranges. There are also major current-drive programs abroad (for example at JET, Tore Supra, and JT-60). The STE experiment will use these results to test and demonstrate a current-drive method, in a plasma with reactor-like parameters, with an efficiency good enough for reactor operation, that can sustain an optimum current profile, for times long compared to any relaxation time of the plasma.

Building on the major accomplishments in plasma physics of the past three

¹ Technical Planning Activity Final Report, Argonne National Laboratory Report ANL/FPP-87-1 (January 1987), pp.64-70.

Fig. 1.1 – Roles of STE (steady-state and advanced geometry), CIT (short-pulse ignition), and ITER (long-burn demonstration) in the U.S. fusion program as seen by the TPA activity. Stippled portions indicate the contributions of existing experimental programs.

decades, it is imperative to begin to confront the *development* issues in the fusion R&D effort. A critical point leading to the need for integration of physics and technology is the recognition that the physics performance of a confinement device cannot be separated from the interaction between the plasma and its surroundings, and that control of the plasma boundary conditions may be the most significant technique for plasma optimization.

Economics also drives us strongly in the direction of integration. The fusion program can no longer afford separate facilities for testing each feature of a proposed reactor. The requirement of "relevance" sets minimum standards for size and performance. The cost of meeting these requirements is measured in units of \$100M.

For successful operation of the ITER, and future devices, there are two critical sets of issues that must be resolved. One is the exploration of the physics of alpha particles and ignition. Recent work has indicated that physics performance depends strongly on the method and strength of heating the plasma. Very little is known about the dynamics of ignition or the effect of nuclear heating on energy confinement. The CIT program is specifically intended to address this question. The second is the problem of steady-state, high-power operation coupled with plasma optimization and control.

It is now the time to establish an integrated tokamak R&D experiment to address these issues. A single facility, designed to minimize cost and development time, can provide the information needed to optimize ITER operations and to move toward a reactor demonstration plant.

This document details the role of the STE project in the U.S. and world fusion programs. We define the technical objectives, and present guidelines for the development of specifications. We also present illustrative examples that indicate the range of possible devices, and show the costs and benefits associated with certain important choices.

The STE program, if aggressively pursued, could be in operation by the mid-1990's. The cost of STE is expected to be significantly less than for CIT, because of the reduced development requirements and the absence of nuclear systems and components.

II. Goals and Constraints

The STE will address, in a timely and cost-effective way, the issues of current drive, high power operation and component development, and plasma control, under steady-state reactor-like plasma conditions. These requirements lead to an intermediate-sized tokamak, with superconducting magnets, operating in a neutron-free environment. Use of existing technology, materials, and components are maximized.

The focus of the STE is on addressing the non-nuclear issues associated with steady-state operation at high power density. This involves questions of plasma physics, tokamak technology, and the interactions between the two. Three particular areas of research can be identified:

- noninductive current drive
- high power density plasma interactive components
- profile control and plasma shaping

Noninductive current drive is one key issue. A major criticism of the tokamak concept since its initial conception has been that, with inductive drive of the toroidal plasma current, it is inherently a cyclic device, requiring a periodic shut-down of the reacting plasma in order to reset the transformer. The ability to drive plasma current by noninductive means – using electromagnetic waves (rf) or neutral particle beams – has been demonstrated in principle. However, the efficiency (current per unit power) is poor, and the effect is strongest in plasma regimes far from reactor conditions (e.g., at low density). Within the coming 3–5 years a number of promising techniques will be tested on present devices. The critical issue is the development of efficient techniques providing the capability for driving all of the plasma current under reactor-like conditions, as well as for controlling the profile of the driven current in order to optimize plasma operation.

The key technological issue to be addressed by the STE is the development of components for high-power, steady-state operation of reactor-like tokamak plasmas. While results from high-power test stands, extrapolation from low power operation, and transient operation at high power density all contribute to the development of these components, a true tokamak test facility is crucial. Development and demonstration of high heat flux components – rf antennas, divertor collectors, vessel armor, ... – must be carried out under conditions of particle and radiation fluxes that correspond to a large, burning plasma. In addition, the environment should be amenable to rapid repair and exchange of components.

The third key issue for STE is the optimization of plasma performance through regulation and control of the profiles of plasma pressure and current, as well as control of the conditions in the neighborhood of the plasma surface. Recent experimental results have indicated that the behavior of a tokamak plasma (stability, energy confinement) is critically dependent on the boundary conditions (*e.g.*, H-mode), and on the profiles of plasma pressure and current. The success of the tokamak concept as a confinement scheme for power production may depend on our ability to control these conditions. External control of the pressure and current profiles must be through development of localizable fueling, heating, and (particularly) current drive techniques. Boundary condition control may be even more important. Regulation of particle and energy fluxes, and electric potentials at the plasma surface is known to influence confinement and impurity content, and is essential to the survival of plasma interactive components.

The optimization of plasma shape is closely tied to the issues of profile control. The primary goal of modification of the shape of the plasma surface is to raise the allowable operating plasma pressure. Achievement of the more advanced plasma shapes, which promise significant improvement in pressure limits and the possibility of confinement improvement as well, requires that the plasma current distribution be accurately controlled.

A preliminary study indicates that these objectives can be met with a medium-scale tokamak device, similar in size and field to present research tokamaks. The plasma volume will be less than, but comparable to that of TFTR, and the magnetic field strength would be similar. The STE device will have the capability of continuous operation, in the magnetic field coil systems as well as the heating and current-drive systems.

The need for steady-state operation constrains the choice of magnet systems. While it is conceivable that resistive magnets could be designed to meet this requirement, considerations of operating cost and the need to work in a relevant, integrated environment lead inexorably to superconducting coil systems for both toroidal and poloidal magnetic field.

The successful operation of the STE relies on the generation of reactor-like plasma conditions. The significant exception to operation at these conditions is that this device must be non-nuclear, that is, it should not operate with deuterium or tritium plasmas. There are several reasons for this. A flexible, responsive research program requires good access to the interior of the tokamak for observation and an

extensive complement of diagnostic instruments. The radiation environment of a reactor-like plasma is too severe to allow this. Also, the interior of the device must be readily accessible for maintenance, repair, and replacement of test modules and components. In a nuclear system, such access would be precluded by intense activation of the structure and internal components. Finally, the cost associated with nuclear operation is high.

Even deuterium operation is precluded, because of the high duty cycle of the STE. Present tokamaks accumulate roughly 20,000 seconds of plasma operation per year. STE will reach this level in a few days, and should provide more than 2,000,000 seconds of operation per year. While such extended operating time is an integral part of the STE program, the radiation dose that would accumulate for extended operation at reactor-like plasma conditions in deuterium would lead to severe shielding and activation problems.

In order to hold down the cost of the STE facility, it is necessary to be very careful about using materials and technologies that require significant development effort. One feature of the fusion program has been that each experiment endeavors to push the technological state-of-the-art in as many areas as possible. This approach has led to high performance devices, but often with major delays and at considerable expense. The mission of the STE can be accomplished using proven materials and technologies.

An example is the choice of magnet technology. The anticipated magnetic field strength requirement for the STE is of the order of 5 T at the plasma axis, leading to a variety of possible choices for superconducting material and coil technology. The selection should be based on three things: cost, performance, and the existence of developed and tested configurations.

III. Fundamental Parameters

The STE concept is based on the establishment of "steady-state", "reactor-like" plasma conditions. These conditions imply a minimum pulse length of roughly a half hour, and a plasma pressure of about 3 atmospheres (about 10 keV at $1 \times 10^{20} \text{ m}^{-3}$).

Pulse Length

The minimum pulse length corresponding to "steady-state" operation depends on the relaxation time of the processes under consideration. For the STE program, the ultimate goal is the demonstration of continuous operation without any constraints on pulse length (other than the patience of the operators). However, for the other program goals of the STE, a shorter pulse length will provide sufficient confidence that the full implications of operation in an asymptotic stationary state have been explored. A good rule of thumb is that the stationary state should be maintained for at least five characteristic time constants. For a simple exponential process, this brings the evolving parameters to within 1% of the steady-state value.

The longest characteristic time constant for a tokamak plasma is the L/R time, for the change of the total plasma current:

$$\tau_{L/R} \simeq \mu_0 \sigma \frac{a^2 \kappa}{2} \left(\ln \frac{8R}{a} - 2 + \frac{\ell_i}{2} \right)$$

For the range of device parameters considered for STE, this time constant can be as long as 400 seconds. Five time constants gives the minimum pulse length of 2000 seconds (about a half hour).

Another magnetic time constant is the skin time, the characteristic time for changing the distribution of the current in a restricted region within the plasma: $\tau_{skin} \simeq \mu_0 \sigma \delta^2 / 4$, where δ is the thickness of the region of interest. For rearrangement of the current distribution within the entire plasma cross-section, the skin time corresponds to the ℓ_i term in $\tau_{L/R}$, which is typically 0.2–0.5 $\tau_{L/R}$, as much as 60–80 seconds.

The other important characteristic times are the equilibration times of components that interact with the plasma. The thermal time constant is $\tau_{th} \simeq \delta^2 / 4D_{th}$, where the thermal diffusivity is the ratio of the thermal conductivity to the heat capacity, $D_{th} = \kappa / \rho C_p$. For example, the thermal time constant for a graphite wall tile is a few seconds.

In almost all plasma experiments long conditioning times are required for optimal plasma behavior. This is due to the time needed for establishing a stationary state for diffusion of gas in the surface layers of the wall materials, and in surfaces not directly exposed to the full radiation from the plasma. Such effects are design dependent, but cannot be eliminated entirely.

Plasma Conditions

The approximate plasma parameter range for STE is constrained in several ways. The primary constraint is the need to minimize the extrapolation from STE results to much larger, burning plasma devices. This leads to the requirement that the plasma energy density within the plasma and the energy flux at the plasma surface be comparable to what is expected for reactor-like systems. To an extent, it will be necessary to rely on the comparability of dimensionless parameters (e.g., collisionality, dielectric constant), but the credibility and applicability of the STE experiment will be significantly enhanced if the dimensioned numbers match as well.

A wide range of densities and temperatures have been proposed in various reactor studies. However, the basic D-T fusion cross-section constrains the allowable energy density (or equivalently, pressure). The fusion reaction rate is proportional to $p^2 \langle \sigma v \rangle / T^2$, where p ($\propto nT$) is the plasma pressure and $\langle \sigma v \rangle / T^2$ depends only on temperature with a peak at about 14 keV (90% of peak at about 9.5 and 20 keV). Maximizing the fusion power density in a reactor thus leads to maximizing the pressure with the temperature held within the optimum range.

With present experiments operating at electron and ion temperatures in the multi-kilovolt range, the STE target parameters should be close to reactor-like conditions. The reference condition that we have selected is an average temperature (electron and ion) of 10 keV.

There is an additional consideration that leads to temperatures in this range. A critical part of the STE mission is the study of plasma-wall interactions under steady-state, reactor-like conditions, and the development of high heat flux components suitable for these conditions. Chemical and physical processes occurring at surfaces facing the plasma depend strongly on the energy distribution of the particles hitting the surface. To carry out the necessary studies, the STE plasma must correspond to reactor plasmas in this respect as well.

Not only temperature, but also pressure and energy flux are constrained. In reactor studies, the heat flux to the wall and the ability of the wall to sustain neutron

damage are principal constraints. These considerations restrict the neutron wall loading to a few MW/m², and the thermal energy flux to the range of 0.1 to 1 MW/m². In order to provide the appropriate environment for testing plasma interactive components, we take an average heat flux of roughly 0.3 MW/m² as the STE design criterion.

The pressure and density goals for STE are set by the goal of carrying out relevant demonstrations of current drive and plasma profile control. Current drive efficiency, for example, has been shown to depend strongly on density, and to exhibit density cut-offs in certain cases. Transferability of STE results requires operation at densities in the neighborhood of $1 \times 10^{20} \text{ m}^{-3}$. The combination of this density and temperatures around 10 keV leads to a plasma pressure of about 0.3 MPa, or 3 atmospheres.

Magnets

The objectives of the STE program clearly require steady state magnets. There are two principal reasons for this. From the point of view of the future of the fusion program, the STE device is the place for integrating the magnet technology required for reactors into an operating tokamak. Demonstration of the feasibility of such integration is one of the prime objectives of the STE program. There is also the very practical reason that the cost of power required for continuous operation of resistive magnets would be prohibitive.

The choice of superconducting magnet technology is more difficult. The options for selection of superconductor are NbTi and Nb₃Sn. The advantage of NbTi is that it is a proven, commercially available technology. The costs of wire and magnet construction are well established and subject to competitive pressure. The material is strong, ductile, and lends itself to reliable coil construction.

The Nb₃Sn material is capable of significantly higher performance in terms of critical field and current. Thus, fairly large eddy-current losses can be tolerated while still maintaining stable operation. The primary difficulty is that Nb₃Sn is brittle and intolerant of strain, making winding difficult and possibly requiring high temperature treatment after winding.

For the STE program, the choice of conductor is a balance between the use of materials and technology that will be relevant to future devices and the cost-effectiveness of proven components and designs. We expect the development programs now in place for ICCS (cable-in-conduit) Nb₃Sn coils, in addition to recent advances

in design and production quality of multifilamentary Nb_3Sn conductor, will have a major impact on the choice of material for the STE magnets. We note that, to an extent, the development of ICCS magnet technology is independent of the selection of superconductor, and may be valuable for STE regardless of the material used.

Under present STE design assumptions, the peak field at the coil is about 8 tesla, which can be provided by either NbTi or Nb_3Sn . As noted, this gives a very large stability margin against eddy-current losses that will occur during initiation of the plasma current. It is possible, particularly with Nb_3Sn , that the coils could be designed for higher field operation (to ~ 8 T at the plasma; ~ 13 T at the coil) with smaller stability margin. If the STE design process shows that this feature is cost effective, it would provide the opportunity for future upgrade to higher current and power density operation.

IV. Tokamak Specifications

We examine a range of tokamak parameters and look at the costs and benefits of size variations. Field and current levels are set by insistence on relevant values of β and q . The critical physics parameters are the scaling of energy confinement and current drive efficiency. Sufficient power density at the plasma surface is also a requirement. An important consideration is flexibility, to allow variation of plasma shape, and to permit possible upgrades in field, current, and power subject to cost effectiveness.

Magnetic Field and Plasma Current

For the STE experiments, the target plasma parameters are temperatures of about 10 keV and density around $1 \times 10^{20} \text{ m}^{-3}$. These parameters can be related to the machine parameters of size, field strength, and plasma current, by insisting that the STE tokamak operate in a plasma regime that is relevant to reactors.

The two key plasma parameters are β , which measures the plasma pressure in terms of the pressure of the external magnetic field, and the safety factor q , which constrains the plasma current. Values of β that are too high, or of q that are too low, lead to deteriorating energy confinement, reduced stability, and eventually disruption. On the other hand, high pressure (high nuclear power density in a reactor) and efficient use of the external field requires operation at high β . High β limits and good energy confinement are strongly correlated to large plasma current, which leads to operation at low q .

A number of stability analyses have found, and experiments tend to confirm, that the maximum stable value of β is proportional to the normalized plasma current, $I_N \equiv \mu_0 I / aB$, for a wide variety of conditions. We define the normalized value, $\beta_N \equiv 100 \times \beta / I_N$ (Troyon obtained the limit of $\beta_N \simeq 3$ for ballooning modes.) The STE device should be capable of operation close to these limits at its highest current and field capacity. If the center of the operating parameter space is near $\beta_N \sim 2$, this condition will be satisfied.

The limitation on q is also well defined. Experiment and theory both indicate that there is a hard stability limit that sets the minimum value of q , and energy confinement studies indicate that there is a deterioration in confinement as q is lowered toward this limit. For elongated plasmas ($\kappa \gtrsim 2$), operation with $q < 3$ appears to be difficult, and may have deleterious effects on stability and confinement.

The actual safety factor can be represented by the approximate "cylindrical q ",

$q_{cyl} \equiv (2\pi a^2 B / \mu_0 I R) \cdot (1 + \kappa^2)/2$, for parametric scaling studies. In defining the STE parameters we take $q_{cyl} \sim 2$. For the strongly shaped, diverted plasmas expected in the STE device, $q_{cyl} \simeq 2$ will correspond to a value of the safety factor at the 95% flux surface of $q(95) \simeq 2.8-3.5$.

With fixed pressure, normalized β , and safety factor, the required toroidal magnetic field strength and plasma current become functions of plasma dimensions alone:

$$B_T \propto \left[\frac{\langle p \rangle q_{cyl}}{\beta_N \left(\frac{1+\kappa^2}{2} \right)} \right]^{1/2} \frac{R^{1/2}}{a^{1/2}}$$

$$I_P \propto \left[\frac{\langle p \rangle \left(\frac{1+\kappa^2}{2} \right)}{q_{cyl} \beta_N} \right]^{1/2} \frac{a^{3/2}}{R^{1/2}}$$

Size and Shaping

We have examined a range of devices consistent with these parameters. The fixed values are $\langle p \rangle = 0.3$ MPa, $q_{cyl} = 2.2$, $\beta_N = 1.7$ (corresponding to $\langle \beta_{95} \rangle = 2.2 I / a B$), and $\kappa = 2$. Where needed, the triangularity is taken to be $\delta = 0.4$. With this selection of parameters, the toroidal magnetic field strength at the minor axis is in the neighborhood of 5 tesla, and the plasma current is 3–5 MA. Specifically, the toroidal magnetic field is taken to be $B_T = 2.5 (R/a)^{1/2}$, and the plasma current is $I_P = 14 a^{3/2} / R^{1/2}$.

The independent variables are size, represented by the minor radius a , and the aspect ratio R/a . The cases considered range from $a = 0.3$ m to $a = 0.7$ m, and from $R/a = 2$ to $R/a = 6$. The major radius ranges from 1 m to 3 m. These values encompass the range of sizes and aspect ratios that are considered for the STE device. The smaller devices have the advantage of lower cost and lower power requirements. On the other hand, the smaller devices do not allow personnel access to the chamber for maintenance and repair. The plasma characteristics of the smaller devices also tends to be dominated by surface effects.

This parameter range does not represent a major extrapolation from the present experimental data base. Thus, empirical scaling rules based on present data should give reliable predictions of STE performance.

The requirement of personnel access sets the minimum height of the inside of the vessel at roughly 2 m. The fraction of the plasma dominated by surface effects is

approximately $2\lambda/a$, where λ is the mean free path of a neutral atom at the plasma edge. Setting $\lambda/a \leq 0.1$ gives a lower limit on the product of average plasma density and radius of $\langle n \rangle a \geq 0.5$.

High elongation and indentation have been shown to be beneficial for increasing the equilibrium β -limit or for access to the second-stability regime. In addition, possible improvements in plasma confinement have been suggested. There are several tokamaks now in operation or under construction (DIII-D, PBX, TCA) which have shape optimization studies as a major feature of their experimental program, and we expect that there will be further indication of the benefits and limitations of plasma shaping in the near future.

Stable operation of shaped plasmas depends on the ability to establish and control appropriate plasma current profiles. Generally, the current profile must be broader than attainable in steady-state without local current drive. This capability is a primary feature of the STE program, and it will allow investigation of a broader range of plasma shapes than is possible at present.

For the purpose here of determining the dependence of heating and current drive power, and magnet size on the basic plasma dimensions, we have fixed the elongation at $\kappa = 2$. The reference plasma shape is bounded by a separatrix that provides a double-null divertor.

Heating and Current Drive Power

The auxiliary power required to produce plasmas at temperatures and densities which are appropriate to the goals of STE has been calculated using the MUMAK volume averaged power balance code.¹ Here, we briefly describe the models and assumptions used, and present the results of comparisons for various energy confinement scaling assumptions and neutral beam injection energies, as well as aspect ratio and minor radius.

MUMAK is a zero-dimensional, time-independent power balance code for modeling particle and energy flow in tokamaks. The separate electron and ion equations include terms for noninductive current drive power, heating power, conduction power losses described by empirical energy confinement scaling laws, and radiation power losses. The third equation is a pressure balance constraint. These relations yield the average

¹ M. E. Fenstermacher, *MUMAK - A Computer Code for Modeling Plasma Power Balance and Current Drive in Tokamaks*, LLNL Report UCID-21038, April 1987.

electron density $n_e = \langle n_e(r) \rangle$, and the average electron and ion temperatures (density-weighted: e.g., $T_e = \langle n_e T_e \rangle / \langle n_e \rangle$). Fixed radial profiles are assumed (of the form $n(r) = n(0) [1 - (r/a)^2]^{\alpha_n}$), with $\alpha_n = 0.5$ and $\alpha_T = 1.0$). For these comparisons, we fix both the density and temperature, corresponding to $\langle p \rangle = 0.3$ MPa: $n_e = 1.0 \times 10^{20} \text{ m}^{-3}$ and $T_e = 9.35 \text{ keV}$ ($\langle T_e \rangle = 7.80 \text{ keV}$). Table 4.1 gives the results of these calculations.

In the power balance relations, the conduction power is calculated from an energy confinement time ($P_{i,e}^{cond} = W_{i,e} / \tau_{Ei,e}$, where $W_{i,e}$ is the total energy in the electron or ion fluid). The energy confinement scaling is assumed to follow "saturated H-mode" form:

$$\tau_E = \left[\frac{1}{\tau_{NA}^2} + \frac{1}{(2\tau_{KG})^2} \right]^{-1/2}$$

where the "neoAlcator" confinement time is

$$\tau_{NA} = 0.0625 \left(\frac{\bar{n}}{10^{20}} \right) a R^2 q \text{ sec}$$

and the "Kaye-Goldston" time is

$$\tau_{KG} = 0.055 \frac{\left(\frac{I}{10^6} \right)^{1.24} \left(\frac{n}{10^{20}} \right)^{0.26} \kappa^{0.28} R^{1.65} \left(\frac{M}{\sqrt{2}} \right)^{0.5}}{B^{0.09} a^{0.49} \left(\frac{P}{10^6} \right)^{0.58}} \text{ sec}$$

The safety factor used here is $q_0 \equiv (2\pi a^2 B / \mu_0 I R) \cdot [1 + \kappa^2(1 + 2\delta^2)] / 2 = 2.80$. The confinement times for electrons and ions are assumed equal: $\tau_{Ei} = \tau_{Ee}$.

MUMAK also calculates noninductive current drive efficiencies for electron cyclotron, neutral beam, lower hybrid slow wave, and fast wave current drive, and calculates the bootstrap current as well. For these comparisons we have assumed that neutral beam injection (NBI) and lower hybrid (LH) waves will be used for current drive. Additional electron cyclotron (EC) heating power is used to achieve the desired bulk electron temperature. The NBI model includes the effects of electron trapping as well as the beam energy dependence. The LH models take into account both the accessibility limit and the condition for avoiding strong electron Landau damping of the wave to determine the radius of maximum penetration for the waves.

With fixed plasma density, temperature, and profiles, the current drive efficiencies do not vary over range of devices examined. The calculated NBI current drive efficiency is $\eta_{NBCD} = 0.25$ ($\eta = nIR/P$), and for lower hybrid waves it is $\eta_{LHCD} = 0.35$. The high NBCD efficiency is obtained with the use of 500 keV neutral beams, which

Table 4.1 - MUMAK Cases for STE

case #	1	2	3	4	5	6	7	8	9
R/a	2	3	3	3	4	4	4	4	4
$\bar{R}$ (m)	1.0	1.2	1.5	1.8	1.2	1.6	2.0	2.4	2.8
a (m)	0.5	0.4	0.5	0.6	0.3	0.4	0.5	0.6	0.7
B (T)	3.54	4.33	4.33	4.33	5.00	5.00	5.00	5.00	5.00
I_{bs}	1.19	0.96	1.19	1.43	0.71	0.95	1.19	1.43	1.67
I_{NB}	1.22	0.70	0.94	1.19	0.37	0.57	0.78	0.99	1.20
I_{LH}	2.54	1.58	1.91	2.23	1.02	1.28	1.53	1.78	2.03
I (MA)	4.95	3.23	4.04	4.85	2.10	2.80	3.50	4.20	4.70
P_{NB}	6.6	4.6	7.7	11.0	2.6	4.7	7.3	10.4	14.2
f_{shine}	0.24	0.26	0.25	0.20	0.32	0.22	0.15	0.10	0.07
P_{LH}	7.8	5.5	8.4	11.8	3.5	6.0	8.9	12.3	16.4
P_{EC}	16.5	12.0	10.0	8.0	9.0	7.9	6.5	5.1	2.5
P_{IC}	22.5	15.8	16.8	18.0	12.5	13.2	14.4	15.9	17.5
P_{total} (MW)	53.4	37.9	42.9	48.8	27.6	31.7	37.1	43.7	50.6
τ_E (msec)	74	82	143	219	63	131	221	327	450
P/A (MW/m ²)	1.71	1.26	0.92	0.72	1.23	0.79	0.59	0.49	0.41

case #	10	11	12	13	A	B	C
R/a	5	5	5	6	3.64	2.57	3.18
R (m)	2.0	2.5	3.0	3.0	1.2	1.67	1.75
a (m)	0.4	0.5	0.6	0.5	0.33	0.65	0.55
B (T)	5.59	5.59	5.59	6.12	4.17	4.01	4.46
I_{bs}	0.95	1.19	1.43	1.19	0.78	1.55	1.31
I_{NB}	0.49	0.67	0.85	0.59	0.46	1.46	1.03
I_{LH}	1.05	1.28	1.49	1.08	1.18	2.67	1.98
I (MA)	2.50	3.13	3.76	2.86	2.42	5.68	4.32
P_{NB}	4.8	7.6	11.1	7.8	3.1	11.0	8.4
f_{shine}	0.19	0.12	0.09	0.10	0.30	0.13	0.15
P_{LH}	6.1	9.2	12.9	9.2	4.1	13.3	10.2
P_{EC}	6.0	4.0	2.0	3.5	9.7	8.6	7.5
P_{IC}	12.0	13.5	15.0	12.4	13.4	21.5	18.0
P_{total} (MW)	28.9	34.3	41.0	32.8	30.3	54.4	44.0
τ_E (msec)	183	301	440	382	69	209	195
P/A (MW/m ²)	0.58	0.44	0.36	0.35	1.23	0.80	0.73

penetrate well (the shine-through fraction ranges from about 0.25 for the smaller devices to 0.1 for the larger ones). The LHCD efficiency is high because in these examples it is used for driving current near the plasma edge, and penetration is not a problem. Combining these two methods gives a net efficiency of 0.31.

Even though the selected profiles are not optimized for the bootstrap current, 20–40% of the total plasma current is bootstrap. For these examples, we find that $I_{boot} = 2.38 a$ MA, giving $I_{boot}/I = 0.17\sqrt{R/a}$. With the profile control expected in the STE device, significant improvement in I_{boot}/I is likely – reducing the power required for current drive.

The total power required to heat and drive current in the range of sample devices is shown in Fig. 4.1. This range straddles the transition between confinement dominated by τ_{NA} and by τ_{KG} . The range of power levels required is from below 30 MW for small plasmas at large aspect ratio to above 50 MW. At low aspect ratio and small minor radius, the power requirement is dominated by τ_{NA} . As either a or R/a is increased, the τ_{KG} scaling becomes more important. It is of interest to note the decrease in required heating power with increasing aspect ratio. This is a feature of all of the empirical L-mode confinement scaling rules.

The required current drive power is shown in Fig. 4.2. The power needed is roughly given by $P_{CD} = 45 a^2 \sqrt{R/a} (1 - 0.17\sqrt{R/a})$, where the second term results from the reduction of the external current drive requirement due to the increasing bootstrap fraction. In the range of interest, the dependence of current-drive power on aspect ratio is very weak. Note also that, for the estimated current drive efficiency, the current drive power is less than the required heating power throughout the size range. If either the efficiency is a factor of two lower, or energy confinement is a factor of two better, then the two power levels become comparable. In each case, the difference between required heating power and current drive power is assumed to be supplied by a combination of ECH and ICH.

In Fig. 4.3, we plot the average surface power density. This parameter, the heating power from Fig. 4.1 divided by the plasma surface area is indicative of the capability of the STE device for testing high heat flux plasma interactive components. The required range, heat flux exceeding 0.3 MW/m^2 , is met throughout the parameter space considered, but very high values are quickly reached for small a and small R/a .

In order to provide an indication of the range of performance capabilities of these devices, the operating space of the version with $a = 0.5 \text{ m}$ and $R/a = 4$ ($B = 5 \text{ T}$ and $I = 3.5 \text{ MA}$) is shown in Fig. 4.4. The axes are average density and temperature

Fig. 4.1 - Contours of total power (in MW) required as a function of minor radius and aspect ratio. For the cases considered, this is the heating power; the current drive power is smaller for all cases. The weak dependence on minor radius at low aspect ratio results from the neoAlcator portion of the τ_E scaling rule.

Fig. 4.2 - Contours of power required for current drive (in MW). Note the weak dependence on aspect ratio.

Fig. 4.3 – Contours of surface power density as a function of geometry. This is estimated by dividing the total power by the approximate plasma surface area.

(not density weighted), and the principal contours show the external heating power required at any $\langle n \rangle$ and $\langle T \rangle$. The pairs of small arrows indicate the Murakami and Greenwald density limits. The three hyperbolas represent constant β curves at $\beta_N = 1, 2$, and 3 . The dot-dash curves labelled $0.1-0.5$ indicate the minimum current drive efficiency needed to drive the plasma current at any set of parameters. At the reference operating point, $\eta \simeq 0.2$ would be sufficient.

Coil Specifications

In addition to these physics performance analyses, we have looked at the engineering characteristics of this range of devices. The key feature is the use of superconducting coils in both the toroidal and poloidal field systems. The analysis shows a clear distinction between "compact" ($R/a \lesssim 3$) devices and those with somewhat larger aspect ratio. It should be noted that the exact numerical results of this analysis depend on a number of arbitrary (but not unreasonable) assumptions.

The first important conclusion from this survey is that there are two different characteristic systems, depending on the performance of the poloidal field system. The smaller aspect ratio devices have little space inboard of the TF coils for the PF solenoid. This is illustrated in Fig. 4.5, which shows contours of the inductive capability of the solenoid. This quantity, $\tau = (\Psi - 2LI)/V_{loop}$, is a measure of the capability of the PF system to provide inductive flux in excess of that needed to initiate the plasma current. To the left of the curve $\tau = 0$ there is insufficient room for a solenoid that can bring the plasma current to its design value. For slightly larger devices, the inductive capacity of the solenoid increases rapidly.

The overall size of the coil system (and roughly its cost) is gauged by the coil mass. In Fig. 4.6 we plot the coil mass (toroidal and poloidal field systems) against aspect ratio for various minor radii. For $R/a > 3$, the mass is very closely proportional to R^2 . For $R < 3$, the mass decreases more rapidly. With decreasing size, first the PF solenoid gets squeezed out, and finally there is a limit below which it is not possible to specify a TF coil.

Observations

In these analyses we have avoided direct estimates of cost. Instead we have evaluated the two major elements that enter into a cost calculation, the magnet mass and the heating power requirement. The conversion to cost depends on the conductor and type of construction assumed for the magnets, and on the frequency range of the rf systems and energy of the neutral beams.

Fig. 4.4 - Plasma operating space for the $a = 0.5$ m, $R = 2.0$ m example. The axes are n_e in units of 10^{20} m^{-3} and $T = T_e = T_i$ in keV. Solid lines show power required (MW), and dashed lines indicate contours of constant β_N . The dot-dash curves give the efficiency required at any (n) , (T) if the current drive power is not to exceed the heating power. Short arrows indicate the Murakami (lower) and Greenwald (higher) density limits.

Fig. 4.5 - Inductive capability as a function of minor radius and aspect ratio. To the left of the $\tau = 0$ curve, there is insufficient space for a solenoid capable of establishing the plasma current.

COIL [PF+TF] Mass vs. R/a .

Fig. 4.6 – Estimated coil mass as a function of aspect ratio for $a = 0.4, 0.5$, and 0.6 m. For $R/a \geq 3$, the mass is approximately proportional to R^2 (or $(R/a)^2 a^2$ for the parameters used here).

For very rough calibration, the system with $R/a = 4$ and $a = 0.5$ m has a coil mass of roughly 150,000 kg and requires about 35 MW of heating power (delivered to the plasma).

An indication of the wide range of possible costs associated with the STE tokamak is illustrated by the following tables. Table 4.2 gives the unit costs for representative magnets of various types constructed for the fusion program.² Table 4.3 gives unit costs for a sample of the large plasma heating systems in use on operating tokamaks.³ The cost is normalized to the power delivered to the plasma.

² Adapted from *Unit Cost Basis Notes*, D. B. Montgomery, M.I.T., April 1984.

³ Adapted from *Engineering Study of the Neutral Beam and RF Heating Systems for DIII-D, MFTF-B, JET, JT-60, and TFTR*, W. B. Lindquist and S. H. Staten, 12th Symposium on Fusion Engineering, Monterey CA, October 1987, IEEE Catalog No. 87CH2507-2, p. 795.

Table 4.2 – Magnet Costs (1988 \$)

Magnet	Material	B_{max}	Conductor		Winding	Total
			\$/kA-m	\$/kA-m-T	\$/kg	\$/kg
PLT TF	Cu				13	43
JET TF	Cu				15	60
JET PF	Cu				15	63
DIII PF	Cu				12	82
TFTR TF	Cu				12	100
DIII TF	Cu				12	108
MFTF Solenoid	NbTi	3.3	9.71	2.95	44	122
TFTR BBC PF	Cu				12	147
LCP-GD	NbTi	8	27.40	3.42	100	195
TFTR ring PF	Cu				12	228
MFTF Y/Y	NbTi	7.8	22.43	2.87	123	255
MIT 12T	Nb ₃ Sn	10	42.43	4.24	238	388
LCP-W	Nb ₃ Sn	8	39.65	4.97	410	896

Table 4.3 – Plasma Heating System Costs (1988 \$)

Tokamak	System	Power	Voltage or Frequency	Unit Cost
		MW		\$/W
DIII-D	NB	14	80 kV	5.05
JT-60	NB	20	90 kV	7.71
JET	NB	15	80 kV	5.10
TFTR	NB	25	120 kV	7.56
JT-60	ICRF	5	100-130 MHz	3.25
JET	ICRF	20	23-57 MHz	2.32
JT-60	LHRF	20	1.7-2.3 GHz	2.55

V. Current Drive

A wide variety of current drive schemes are presently under consideration for reactors. Some methods (neutral beams, for example) have had significant experimental tests and have well known benefits and drawbacks; others show promise in theory but have not been tested or verified. Optimization of the bootstrap current is also a major consideration. The STE program will investigate the full range of possible techniques and will demonstrate full, noninductive, steady-state operation.

The desirability of noninductive current drive in toroidal plasmas has led to the beginning of a thorough examination of the variety of techniques that can possibly generate plasma current. These include

- rf methods covering a wide frequency range, including Alfvén waves, ion cyclotron (IC), lower hybrid (LH), fast wave (FW), and electron cyclotron (EC),
- neutral beam techniques involving either existing systems (~ 100 keV) or developmental negative ion beam sources (0.5–1 MeV),
- novel ideas including beat wave generation, helicity injection, and others.

These can also be categorized according to the present status of experimental tests, and the level of technology development required for implementation. For LH, IC, and FW systems, sources are readily available. High frequency EC sources or high energy neutral beams required for reactor-like parameters require development programs (presently under way). Use of FW or IC frequency waves poses a problem in antenna design for a reactor-like environment. The "advanced" ideas (helicity injection, very nonlinear rf, ...) have undergone only the most preliminary analysis and testing.

One possible scenario for initial operation of STE is the use of high energy NB current drive near the axis, with LHCD used for control of the current profile near the plasma edge. EC current drive may also be used at the axis, particularly if the bootstrap contribution is significant. These techniques have already had significant experimental demonstrations. Another possibility is the use of FW current drive at both the center and edge, should the present theoretical optimism be supported by experimental test.

Utilization of the bootstrap current is perhaps of even greater importance than choice of current drive scheme, because it requires no external current source. A

small seed current may be required, and careful tailoring of the pressure profile will be needed in order to optimize the bootstrap effect. With such optimization, it may be possible that 80–90% of the total current can be bootstrap.

In the following sub-sections, we first discuss the bootstrap effect and some general characteristics of noninductive current drive. We then look briefly at the primary possible systems for STE, describing the status of physics and technology, and the prospects for further development.

Bootstrap Current

The “bootstrap” current is a toroidal current predicted by the neoclassical transport theory. There is some experimental evidence (e.g., from low density current drive experiments on TFTR) that it occurs as predicted. However, as with other neoclassical effects, it may be observable only under certain conditions – the presence of microturbulence that increases energy transport may reduce the bootstrap effect. Under certain circumstances, however, the bootstrap current can provide a large fraction of the total toroidal current. The bootstrap current density is approximately

$$J_{bs} \propto \left(\frac{r}{R}\right)^{1/2} \frac{1}{B_p} \frac{\partial p}{\partial r}$$

Several items should be noted. First, this expression is valid only for low collision frequency – if the collisionality exceeds one, the bootstrap current disappears. Also, the density and temperature gradients should be treated separately – bootstrap current density is several times more sensitive to density gradient than to temperature gradient, a matter of concern for H-mode operation with flat density profiles.

The fraction of the total plasma current provided by the bootstrap effect depends strongly on both the current and pressure profiles. In terms of the normalized beta ($\beta_N = 100 \times \beta/(\mu_0 I/aB)$), the bootstrap fraction is

$$\frac{I_{bs}}{I} \simeq C_{bs} \beta_N \left(\frac{R}{a}\right)^{1/2} q \left(\frac{2\kappa^2}{1+\kappa^2}\right)^{1/2}$$

where the coefficient C_{bs} is largest for peaked pressure (density) profiles, flat current profiles, and low Z_{eff} . Under such conditions, almost all of the current can be contributed by the bootstrap effect. For more usual profiles, the pressure must be high ($\epsilon\beta_p \sim 1$) if the bootstrap effect is to provide most of the plasma current. The variation of bootstrap fraction with these parameters is illustrated in Fig. 5.1.

DEPENDENCE OF I_{bs}/I ON q_a/q_0 , α_n , Z_{eff}

Fig. 5.1 – Dependence of bootstrap fraction on peaking of the q profile (q_a/q_0), peaking of the density profile ($n \propto (1 - (r/a)^2)^{\alpha_n}$), and on Z_{eff} .

The result of even partial optimization of the bootstrap effect is shown in Fig. 5.2. Fig. 5.2a shows a case corresponding to the profiles used in the MUMAK analyses (see Section IV). The bootstrap current fraction is 0.28. By raising the safety factor on axis to $q_0 = 1.6$, and peaking the density profile (without changing the functional form) to $\alpha_n = 1.77$, the bootstrap fraction is increased to 0.70. In addition, the seed current required near the axis is very small – about 2.5% of the total current. Most of the externally driven current is needed very close to the edge.

Use of the bootstrap current for a large fraction of the total plasma current provides a strong incentive for development of larger aspect ratio, low current configurations. While this region of parameter space is particularly attractive if operation in the second regime of stability is possible, it may also be advantageous for first stability regime operation.

Finally, it is of interest to note that, for any specified current profile, there is a pressure profile that corresponds to $\sim 100\%$ of the current provided by the bootstrap effect. For example, the simplest case is for a flat current profile, which corresponds to $p \propto 1 - (r/a)^{3/2}$. In such cases, the total pressure and current are constrained so that β_p depends only on aspect ratio and profile factors. For typical cases, $\beta_p \sim 0.5-1$. This configuration, which requires very little external current drive, will be of particular interest for the STE experimental program. One difficulty is that, for diffusive heat transport, a flat current profile near the axis implies a singular ($\propto r^{-1/2}$) heat source. In this context, further study of the behavior of the bootstrap effect in the presence of convective (saw-tooth) as well as diffusive heat transport is needed.

Current Drive Efficiency

The figure of merit for noninductive current drive techniques is the ratio of current to power required. This is determined by the balance between the acceleration of current-carrying electrons by an external wave or ion beam, and the drag on these particles due to Coulomb collisions with the bulk of the plasma. In the case of neutral beam current drive, the electron current cancels a portion of the fast ion current, but the same Coulomb processes are occurring. A straightforward analysis of this process shows that, very generally, the efficiency can be written as

$$\frac{I}{P} \propto \frac{T_e}{n_e R \ln \Lambda} f(v_{\parallel}/v_{the})$$

where $v_{\parallel}$ is the velocity of the current-carrying electrons and v_{the} is the electron thermal velocity, and f depends on the current drive method. Often this is written

Fig. 5.2 – Pressure (dot-dash) and current (solid) profiles for (a) $q_0 = 1.0$, $\alpha_n = 0.5$ and (b) $q_0 = 1.6$, $\alpha_n = 1.77$. The dotted curves represent the profiles of bootstrap current and externally driven current. Common parameters for these examples are $I_{\text{total}} = 3.5$ MA, $\langle p \rangle = 0.3$ MPa, $R/a = 4$, $q_{\text{cyl}} = 2.23$, and $\alpha_T = 1.0$. Note that for case (b) 50% of the externally driven current is outside a normalized radius of 0.83.

in terms of an "efficiency":

$$\eta_{cd} \equiv \frac{n_{20} I R}{P}$$

where n_{20} is the electron density in units of 10^{20} m^{-3} . Typically, a tokamak reactor might require $\eta_{cd} \sim 0.5\text{--}1.0$ (the Starfire design used $\eta_{cd} = 1.25$ for lower hybrid current drive). Note that the power for inductive current drive can also be cast into this form. Roughly, $\eta_{induct} \simeq 0.2 T_{10} (v_{the}/v_d)$, which is generally in the range of 50–200 (160 for Starfire parameters). Here, v_d is the average drift velocity of the electrons and T_{10} is the electron temperature in units of 10 keV.

At $\eta_{cd} = 0.5$, full parameter operation of the STE ($n_{20} = 1$, $R = 2$, $I = 4$) will require a current drive power of the order of 16 MW. This will be reduced by the presence of a bootstrap current, which may account for more than 30% of the total.

Current Drive Using Neutral Beams

At the present time, neutral beam (NB) current drive is the leading candidate for noninductive current drive in new, near-term tokamak experiments. Neutral beams are particularly favored for current drive near the center of the plasma because of the known penetration and deposition profiles. One concern is that, as devices become larger and densities approach reactor-like conditions, penetration to the plasma axis will become difficult. In response to this problem, active programs are now under way for the development of high energy ($\sim 1 \text{ MeV}$) negative ion sources.

The physics of neutral beam current drive is simply that the circulating fast ions (injected tangentially) constitute a current that is incompletely cancelled by the electron flow. Electrons are accelerated by interaction with the fast ions, but also suffer drag due to the background ions and the trapped electron population. The resulting net current is of the form

$$J_{NBCD} \simeq J_{fast} \left[1 - \frac{Z_b}{Z_{eff}} (1 - G) \right]$$

where $(1 - G(\epsilon, Z_{eff}))$ is the trapped electron correction. The fast ion current J_{fast} is larger than the injected beam current J_b by the ratio of the slowing-down time of the fast ion to the transit time around the torus:

$$J_{fast} \simeq J_b \frac{v_b \tau_s}{2\pi R}$$

These expressions can be cast into the standard form for current drive efficiency:

$$\eta_{NBCD} \simeq 44 \frac{T_{10}}{\ln \Lambda} \frac{A_b^{1/2}}{Z_b^2} \left(\frac{T_{10}}{E_{100}} \right)^{1/2} \left[1 - \frac{Z_b}{Z_{eff}} (1 - G) \right] \\ \times \begin{cases} 1 & \text{for } E_{100}/E_* > 1 \\ (E_{100}/E_*)^{3/2} & E_{100}/E_* < 1 \end{cases}$$

where E_{100} is the beam energy in units of 100 keV, and $E_* \simeq 2.4 Z_{eff}^{2/3} A_b^{1/3} T_{10}$.

The optimum energy is $E_{100} \simeq 1.6 E_*$, or about 780 keV for $Z_{eff} = 1.5$ and a central temperature of 16 keV. The efficiency in this case is about $\eta_{NBCD} \simeq 0.5$. For an STE device with $R = 2$ m, carrying 3.5 MA, about 10–15 MW would be required.

Penetration of the neutral beam to the plasma axis, without excessive shine-through also constrains the energy. The attenuation rate of a neutral beam results from a combination of processes that can lead to ionization of the neutral atoms. These include ion impact, electron impact, charge transfer, and multiple excitation. The optimum energy is roughly given by

$$E_{100} \simeq 2 n_{20} Z_{eff}^{0.4} \sqrt{\frac{5}{2} a R}$$

which is valid for tangential injection of hydrogenic atoms with energy at or above 500 keV. This energy is generally lower than the optimum for current drive. It is about 370 keV for the example considered above, which would reduce current drive efficiency to around 0.3.

The high beam energies necessary for the next generation of devices (such as STE) require development of efficient and reliable negative ion sources, as well as suitable accelerators capable of handling several amperes, and efficient neutralizers of such beams.

Development of both surface and volume sources is being pursued in the community, with the LBL Surface Conversion Source being considered a prime candidate. In the area of accelerators, in addition to rf and induction linacs, electrostatic-quadrupole-focussed variable energy dc accelerators are being considered for energizing the ions (before neutralization). At present, the new generation of rf linacs seems to be capable of good beam efficiency (before neutralization) with the possibility of building cw linacs.

In the area of neutralizers, the gas, plasma, and photon neutralizers are all being considered. The photon or the photo-detachment neutralizer concept appears to be

an effective method of neutralizing high energy negative ions. It can neutralize a large fraction of the negative ion beam, while leaving many of the negative impurity ions unneutralized. A well designed system of this nature would not require the addition of gas, vapor or plasma neutralizers.

Fast Wave Current Drive

Use of fast waves in the ion cyclotron (IC) and lower hybrid (LH) ranges of frequencies for noninductive current drive has the potential advantage of high efficiency, coupled with a technology that is readily available and relatively inexpensive. Furthermore, lower hybrid fast waves are able to penetrate the core plasma at relevant densities, alleviating the problem associated with LH slow waves.

The efficiency of fast wave current drive is given by

$$\eta_{FWCD} \simeq 0.02 T_{10} \left(\frac{9.2}{x} + 5 + 2.8x^2 \right)$$

where $x = c/n_{\parallel} v_{the}$, with $n_{\parallel}$ the parallel refractive index of the fast wave. A plot of $I/P_{cd} = \eta/n_{20}R$ is shown in Fig. 5.3 for $n_{20}T_{10} = 1$ and $R = 1.5$. An efficiency of several tenths of an ampere per watt can be expected.

In determining appropriate plasma and machine parameters for a fast wave current drive experiment, several effects must be taken into consideration. First, the fast wave must reach the plasma core. For $n_{\parallel} < [1 - (\omega^2/\Omega_e\Omega_i)]^{-1}$, the fast waves can be mode converted to slow waves above a critical density. The launched wave should have a $k_{\parallel}$ -spectrum that avoids the slow-wave mode conversion region. Second, the fast wave must be efficiently absorbed by electrons. Third, the launched waves must avoid absorption by ions through competing mechanisms. Fig. 5.4 summarizes the wave parameters for a plasma of $n_{20} = 1$, $T_{10} = 1$, and $a = 0.5$ m. Waves with $n_{\parallel}$ greater than those indicated by the solid curves (labeled by toroidal field) undergo more than 50% single pass absorption. Waves to the left of the dotted curve are reflected through conversion to slow waves. In the band $2 < \omega/\Omega_i < 12$, ion absorption is significant for $B_T \lesssim 4.5$ T. The allowed frequency bands are then $1 < \omega/\Omega_i < 2$, and $12 < \omega/\Omega_i < 20$. Lower frequencies are preferred from the point of view of coupling efficiency, because fast waves are evanescent near the plasma edge, and the evanescence becomes more severe at higher frequencies. Lower frequency waves also have the advantage of suffering less scattering by edge density fluctuations. It is worth noting that the PLT fast wave experiment operated at about $\omega/\Omega_i \sim 35$, with very weak single-pass absorption. The observed density limits are characteristic of slow-wave current drive, and may well be explained by almost complete mode conversion.

Fig. 5.3 - Ratio of driven current to absorbed power for FWCD as a function of parallel phase velocity for three values of average temperature.

Fig. 5.4 - Operating regimes for FWCD as a function of cyclotron harmonic and phase velocity. Good current drive is obtainable for $\Omega_i < \omega < 2\Omega_i$, and for $\omega \gg \Omega_i$. Operation to the right of the 50% absorption line is indicated.

There are both physics and technological concerns about the use of fast waves for current drive. The questions of physics include the issue of whether absorption by α -particles in a reactor plasma would be significant. Preliminary calculations indicate that this can be avoided with proper choice of frequency and wavenumber spectrum. Another physics issue is absorption by trapped particles. Theory indicates that this effect will lead to a significant reduction in current drive efficiency, but that it will remain sufficiently high for operation in a reactor.

The primary technological issue, which applies to all rf heating and current drive schemes using frequencies below the microwave range, is the need for antenna structures on or inside of the vacuum wall, very close to the plasma. Both the survival of such structures in a reactor-like environment, and the effect of antenna material on the edge plasma are major concerns. These are among the major issues that STE is intended to address.

Advanced Current Drive Methods

A number of advanced current-drive schemes have been proposed recently, offering prospects for some combination of improved current-drive efficiency, better access or improved control over current deposition. However, there has been little or no experimental verification of these methods as yet, and further issues of physics and technology require exploration.

The methods we discuss here are presently under study and may be applicable for long-term work on STE. We briefly discuss three nonlinear EC frequency schemes, a possible enhancement of LH current drive, and three methods for "helicity injection":

- o beat waves
- o rising buckets
- o stochastic acceleration
- o lower hybrid burn-through
- o compact toroids
- o dc helicity
- o oscillating fields

The first three of these use the output of a free electron laser and exploit the nonlinearities induced by the high intensity of FEL pulses. The beat-wave scenario studied involves beating two microwave beams to produce a low-frequency longitudinal wave which Landau damps on the electrons. Rising buckets refers to trapping of electrons in the resonant phase-space islands set up by the wave; the islands can be

accelerated by varying the parallel index of refraction $n_{\parallel}$ (for example, by reflecting the microwave beam off a cylindrical mirror) or the magnetic field strength along a field line. Stochastic acceleration arises when cyclotron harmonics overlap; electrons can be accelerated into the MeV range.

Stochastic acceleration by ECRH resonance overlap was discussed by Menyuk, *et al.*¹ and Villalon and Burke.² For current drive, it has been studied numerically, using the multi-particle orbit code LOREM. Stochasticity requires large fluctuating electric fields, $\tilde{E}/B > 0.1$. The current-drive efficiency is a strong function of the microwave electric field near the threshold, but is only weakly dependent on the initial temperature of the electrons and is thus more or less independent of machine. For $\tilde{E}$ well in excess of threshold, η_{CD} around 0.4 is possible; for more moderate electric fields, values in the range 0.1 to 0.2 would be expected. These values reflect reductions by about 25% each for synchrotron radiation and trapped-particle effects. A major unresolved issue is the role of the parametric instabilities which are likely to exist at these large $\tilde{E}/B$ values.³

Rising-bucket current drive has been evaluated analytically and using the multi-particle near-resonance orbit package from the MCPAT Monte Carlo code. Current-drive efficiencies significantly in excess of quasilinear values can be obtained if the maximum $N_{\parallel}$ can be made close to 1, approximating the condition for cyclotron autoresonance and producing high-energy (100 keV or more) electrons. For a maximum $n_{\parallel}$ of around 0.9 and an effective temperature across resonances (in the portion of momentum space heated by the ECRH) of 2 to 3 times the bulk electron temperature T_e , η_{CD} values of around 0.25–0.3 can be obtained for STE with $T_e = 10$ keV. The scaling is roughly as $T_e^{1/2}$. This efficiency can be achieved at substantially lower $\tilde{E}$'s than for stochastic current drive, if the required spread in $n_{\parallel}$ can be obtained.

Beat-wave current drive has been studied analytically and with the particle simulation code EMONE. It has the significant advantage of allowing precise control over the location of the current deposition, but the disadvantage of requiring two intense microwave sources. The efficiency scales roughly linearly with T_e , giving $\eta_{CD} \sim 0.15$ for STE at 10 keV.

¹ C. Menyuk, *et al.*, Phys. Rev. Lett. **58**, 2071 (1987).

² E. Villalon and W. J. Burke, Phys. Fluids **30**, 3695 (1987).

³ M. Porkolab and B.I. Cohen, *Parametric Instabilities Associated with Intense Electron Cyclotron Heating in the MTX Tokamak*, Lawrence Livermore Laboratory Report UCRL-97477 (1987), to be published in Nucl. Fusion.

Another rf scheme is to use very intense pulses of lower-hybrid power (from, for example, a relativistic electron beam in a klystron) to burn through to the tokamak core (by self-consistently flattening the distribution function). This scheme addresses the principal difficulty with lower-hybrid current drive in reactors, which is penetration. STE would be an excellent facility for testing the concept under reactor-like conditions. This scheme, very recently suggested by Porkolab and Bonoli⁴, is not yet well analyzed. Porkolab and Bonoli estimate that pulses with peak powers in excess of 1 GW might be required in ITER (and hence also STE). This should allow currents driven with lower-hybrid efficiencies at the core.

The helicity injection current drive techniques are very different from the rf methods. The technique is based on generation of dc or low frequency ac currents at or outside of the plasma surface, and allowing this current to penetrate the plasma by relaxation processes.⁵ The current drive efficiency should be very high because the bulk of the electron distribution is accelerated (as with dc inductive current drive). A major issue is the nature of the relaxation process, and its effect on energy confinement. Among the possible techniques for helicity injection are compact toroid (CT) injection, oscillating field current drive (OFCD), and DC helicity injection.

A compact torus is a self-contained equilibrium plasma configuration. For current drive in a tokamak, accelerated compact toruses are injected into the tokamak and stopped in the plasma, depositing, in addition to their mass, their helicity, generating poloidal field in the tokamak. Assuming that the compact tori slow down due to emission of an Alfvén wake,⁶ just reaching the center, and also imposing requirements that the CT reconnect before being expelled from the tokamak and that CT's drive all of the current leads to the conclusion that the power required to drive all the current in STE is about 1/8 the power required to maintain the tokamak energy content. While CT current drive does not scale as nIR , we can normalize to representative parameters for a representative case ($R = 1.8$ m, $n = 0.81$, $R = 1.8$) to obtain a value of $nIR/P = 2.0$. Thus CT's will be efficient current drivers if they can be successfully injected without excessively disturbing the plasma.

DC helicity injection requires electrodes in the edge region of the plasma – for example, divertor tiles may be used. In addition, some of the magnetic flux must

⁴ Private communication.

⁵ T. H. Jensen and M. S. Chu, *Nuclear Fusion* 27 2881 (1984).

⁶ P.B. Parks, *Refueling Tokamaks by Injection of Compact Toroids*, GA Technologies Report GA-A18933 (December, 1987), submitted to *Phys. Rev. Lett.*

intersect these electrodes, as occurs in a divertor. An electric potential applied between these electrodes leads to a toroidal e.m.f. which sustains the plasma current. The effective toroidal e.m.f. is $V = U\Phi/\chi$, where U is the applied potential, Φ is the flux intersecting the electrode, and χ is the toroidal flux in the plasma.

Experiments in the CTX spheromak⁷ and the cold, low-field CDX tokamak⁸ have demonstrated this process. Energy confinement in the former is limited by the ohmic dissipation of a large current flowing in cold edge plasma. This is not a problem in the CDX experiment, where a normal, centrally peaked tokamak current distribution is generated by some anomalous inward pinching process. DC helicity injection is compatible with tokamak divertor configurations and hence it does not *a priori* rule out H-mode confinement. For poloidal divertor tokamaks the required electrode voltage scales as $U \propto BaT^{-3/2}\delta^{-1}$, where δ is the thickness of the scrape-off layer. For $B = 5$ T, $a = 0.5$ m, $T = 10$ keV and $\delta = 1$ cm, $U \simeq 4$ volts. The CTX and CDX experiments have both demonstrated startup from zero plasma current by DC helicity injection alone, and this should also be feasible in large tokamaks.

AC helicity injection (or OFCD) occurs when the internal toroidal magnetic flux χ and the linking external flux ψ are oscillated 90° out of phase. If the injected helicity is efficiently incorporated with little dissipation into the main plasma, it is equivalent to a conventional loop voltage of $V_{equiv} = (1/\langle\chi\rangle)\langle\tilde{\chi}(d\tilde{\psi}/dt)\rangle$. Here $\tilde{}$ denotes the oscillating component and $\langle\rangle$ is the time average. The period of oscillation should be shorter than the skin time of the plasma and longer than the time required for magnetic reconnection to transport helicity to the plasma center.

Preliminary ac helicity experiments have been carried out. Low power OFCD in the ZT-40M reversed field pinch⁹ has driven up to 5% of the total current. Improvement was limited by impurity influx at higher power. A test in the DIII-D tokamak,¹⁰ by oscillating the plasma major radius against a fixed limiter at 5 Hz, also indicated that up to 5% of the current was due to the OFCD. The limitation in this case was the difficulty of obtaining the correct phase relations.

⁷ C. W. Barnes *et al.*, *Phys. Fluids* **29** 3415 (1986).

⁸ M. Ono *et al.*, *Phys. Rev. Lett.* **59** 2165 (1987).

⁹ K. F. Schoenberg *et al.*, *Bull. Amer. Phys. Soc.* **32** 1828 (1987).

¹⁰ S. Yamaguchi and M. J. Schaffer, *Bull. Amer. Phys. Soc.* **32** 1900 (1987).

VI. High Power Components

A key feature of the STE program will be the development, testing, and demonstration of plasma interactive components in a high power density environment. This effort is an integral part of the overall mission of STE, because of the unavoidable linkage of plasma-surface interactions with particle and energy confinement within the plasma. The STE will provide a non-nuclear facility permitting hands-on maintenance and easy replacement of test components and modules.

The design, development, and demonstration of plasma interactive components capable of reliable operation in a high power density tokamak environment is one of the primary goals of the STE program. Although the need for such development has been long recognized by the U.S. fusion research program, incorporation of this work into a working plasma environment has been severely limited, in part because tokamak devices capable of the high power fluxes for extended periods of time have not been available. A second reason for pursuing this effort in STE is the need for testing high power components in a non-nuclear environment, where observation, diagnosis, maintenance and replacement can be carried out without excessive time or money expenditure, and where failure of a test component will not have a catastrophic impact on the overall program.

The third key reason for performing this work in the STE facility is the critical nature of the interaction between the material surfaces bounding the confinement volume and the performance of the tokamak plasma. In recent years, particularly with the studies of H-mode confinement in diverted plasmas, the importance of plasma boundary conditions has been recognized. The transport of fuel particles, impurities, and energy within the plasma are known to be strongly affected by the neutral particle densities, fluxes and energies in the boundary region. For these purposes the boundary layer can be taken to extend from the material surface of the wall (or even a few hundred Ångströms below the surface) to at least one ionization mean free path inside of the nominal plasma boundary (the separatrix for a diverted plasma).

The problems and issues associated with steady state operation of reactors have been summarized in a recent report sponsored by the DOE.¹ This report discusses in

¹ *Technical Assessment of Critical Issues in the Steady State Operation of Fusion Confinement Devices*, DOE OFE Panel on Steady State Issues, Institute of Plasma and Fusion Research, U.C.L.A., Report UCLA-ENG-88-02, January 1988.

detail questions associated with plasma physics and plasma interactive components in view of steady state operation, primarily from the technological perspective. The "physics" issues are seen as:

- o noninductive current drive
- o stability and profile control
- o impurity control
- o particle removal

The "technology" issues are:

- o materials selection
- o heat transfer and thermal stress
- o neutron irradiation
- o erosion and redeposition
- o tritium inventory and permeation
- o disruptions
- o heating and fueling components
- o diagnostics and control

In addition to these issues identified by the panel, the control or elimination of disruptions should be added to the physics list, and design for maintainability to the technology list.

The two physics issues of noninductive current drive, and stability and profile control are the two other major foci of the STE program (discussed in Sections V and VII, respectively). The other two "physics" issues are technology issues seen from the perspective of the tokamak physicists. With the exception of issues of neutron irradiation and tritium, all of these questions result from the problems associated with the high heat and energetic particle fluxes in the boundary region, and can be addressed in a non-nuclear environment. In fact, in some areas testing in a non-nuclear environment is preferable (internal component development, for example).

Materials Selection

The choice of appropriate materials for plasma interactive components (particularly limiter or divertor tiles) is one critical issue which illustrates the interaction between the various requirements and competing considerations. From the point of view of impurity generation, the choice appears to be either high-*Z* materials such as tungsten, molybdenum or tantalum, and low-*Z* materials like carbon or beryllium.

In addition to their good structural properties, high-Z refractory metals are attractive for their very low sputtering coefficients at the low ion impact energies associated with the low temperature, high density divertor plasma mode. However, the divertor plasma must then be able to sweep most of the (few) sputtered high-Z atoms back to the divertor tiles, because they cannot be allowed to deposit on the main torus wall. If deposited on this wall, high-Z atoms can be sputtered into the main plasma by high energy charge exchange atoms. Once in the main plasma, high-Z ions can be catastrophic, because of fuel dilution as well as serious cooling due to intense line radiation.

Low-Z materials such as graphite (or carbon-composite) divertor plates are attractive because, among other features, the fusion plasma can tolerate greater contamination by the carbon impurity. However, carbon is chemically sputtered at appreciable rates by plasma bombardment at any energy level. This puts severe constraints on the capability of the divertor plasma to prevent carbon backstreaming to the main plasma edge, and requires very effective local redeposition. At present it is not known which option is more serviceable, or whether a new and better concept is required. Operation with limiters, ergodic divertors, or other primary plasma-wall contact configurations leads to similar problems.

There are many other issues of materials selection which will be addressed in the STE program [in many respects, most technological problems reduce to either finding the right material, or designing around the inadequacies of available materials]. Fabrication of high heat flux components is an issue. This becomes particularly important in the development of techniques for reliably joining plasma-facing tiles to substrates and to coolant tubes. We must also resolve the conflict for plasma-facing components between the requirements of continuous high heat flux capability and of transient or disruption resistance.

Another issue is the development of materials such as ceramics and aluminum alloys for some reactor components. These are thought to be useful in reactors because of their low activation characteristics, but in other respects their utility is severely limited. STE will provide a facility for testing components made of these and other novel materials under realistic conditions.

An Example - Carbon

At present, carbon materials are the most attractive choice for high power plasma interaction components, steady state or otherwise, largely because of their unique

combination of high-temperature thermal conductivity, strength, low thermal expansion and low vapor pressure with low atomic number. Although the operational data base is growing as more fusion experiments incorporate carbon components, to date it is limited to just a few grades of graphite. Carbon-carbon composites (carbon reinforced by carbon fibers) are relatively new materials, but already they are offered commercially with a wide range of property combinations. Improved composites are continually being announced, and more can be expected as research continues. New conventional grades of graphite are also being developed. To a considerable extent, properties of composites can be tailored to a specific application, but at significant cost.

The capacity of carbon-based materials to handle high heat fluxes is shown by tests at the Plasma Materials Test Facility (PMTF) at Sandia National Laboratory. This facility uses electron and ion beams to test candidate material thermal and mechanical responses to intense surface energy deposition. Several carbon-carbon composites now surpass 100 MW/m^2 deposited nonuniformly for 1 second in notched samples, reducing the sensitivity to transient heat loads.

One principal drawback to carbon materials is their susceptibility to chemical sputtering. Chemical sputtering is greatest at intermediate material temperatures (roughly 400–800 C), and it remains large even at low particle bombardment energies. It is somewhat reduced by operation at 1000 C or more, which is also compatible with a power producing reactor. Another alternative for controlling chemical sputtering is alloying, but this has not yet been pursued to any great extent. The alloy C-SiC (10% wt SiC) as a 0.1 mm-thick pyrolytic coating on a POCO AXF-5Q graphite substrate was found to greatly reduce chemical sputtering. The normal physical sputtering remained unchanged. Thermal tests at PMTF and *in situ* tests at selected neutral beam armor and limiter positions in Doublet III for over 11,000 plasma shots showed greater resistance to surface damage than neighboring TiC-coated tiles. However, the pyrolytic coating process was expensive, and graphite alone suffices for present fusion experiments, so no further tests or development have taken place.

The other major drawback of carbons is their relatively low resistance to neutron damage. This manifests itself as a rapid increase in swelling and deterioration of mechanical properties above a threshold neutron fluence that is several times lower than that tolerated by metals commonly used in nuclear applications.

Heat Transfer and Thermal Stress

Heat transfer performance is a major driver of component design for high power, steady state operation. This is in contrast to design for pulsed duty, where thermal shock and fatigue issues dominate. The thermal stresses that develop when materials are exposed to high heat fluxes are often the most severe loads to which the material is subjected. Temperature gradients associated with the steady state conduction of 10 MW/m^2 range from approximately 25 C/mm for copper and diamond to over 1000 C/mm for stainless steels, with irradiated graphites at high temperature developing about 250 C/mm.

A figure of merit can be obtained for comparison of different materials with regard to thermal stress limits. The maximum allowable product of heat flux and thickness is

$$\max(Q \cdot d) \simeq \frac{2\sigma_y(1-\nu)\kappa}{\alpha E}$$

where σ_y is the yield stress, ν is Poisson's ratio, κ is the thermal conductivity, α is the coefficient of thermal expansion, and E is Young's modulus. The value of this figure is given in Fig. 6.1 for a variety of materials.* For example, tungsten has a figure of merit of about 15 mm-MW/m^2 , or the maximum thickness at 10 MW/m^2 is about 1.5 mm, which leaves little margin for error.

If the stresses do not exceed elastic limits, then the fatigue resistance of most materials is adequate and will probably not be life limiting. The best thermal stress figures of merit are found in special graphites and copper alloys. Thermal shock survival is still a requirement that constrains thermal, electrical and mechanical designs, if plasma disruptions cannot be suppressed. The elimination of the disruption threat would lift these constraints and yield improved component performance. Heat transfer to liquid coolant does not directly affect plasma interaction, but it is an unavoidable bottleneck that strongly influences component design.

In addition to thermal stress considerations, the development of adequate cooling schemes for limiter blades and divertor tiles is an important issue. In particular, these components must be able to cope with significant variations from the nominal operating conditions for which they are designed. In addition to the engineering design of the cooling tubes, ducts, or channels, effects of high velocity flow – including erosion and corrosion – must be examined.

* Adapted from Ref. 1.

Fig. 6.1 - Thermal stress figure of merit for a variety of candidate materials.

Erosion and Redeposition

Material erosion and redeposition by plasma are critical issues for steady state operation, and must be investigated in a realistic machine and plasma environment. Erosion occurs by physical sputtering, evaporation and by chemical reaction with hydrogen. (The latter is significant only for carbon materials.) Some of the eroded material is pumped out, but most of it is redeposited, usually at some distance from the point of erosion. When local redeposition occurs, it reduces the net erosion rate.

In addition to the problem of impurity generation, erosion is important because high heat flux components are necessarily thin and highly stressed, and so are sensitive to material loss. This problem is especially severe for limiters and divertor plates. Basic erosion phenomena are being studied in specialized test stands, such as PISCES at UCLA. Fusion reactor conditions will be more complex, in part because impurity ions generated by the reactor components themselves sputter much more than hydrogen. Runaway impurity self sputtering is a phenomenon that must be avoided. Erosion control has to be achieved by a combination of materials selection and control of the incident particle and energy fluxes. Redeposition is strongly coupled to plasma and geometry, and test stand findings must be verified under actual operating conditions.

Edge Plasma Conditions and Transients

The technology-oriented assessment of steady state issues in the report of the Panel on Steady State Issues¹ places emphasis on the need for control of plasma characteristics, particularly the edge conditions, and on the elimination of transients such as disruptions. These are cited as "physics" rather than "technology" issues, and are passed to the tokamak physics community as problems to be solved. At the same time, the physics community is relying on technologists to address the recycling and impurity control questions.

In fact, these are intimately related issues. With the high power fluxes and energetic particle distributions expected in the boundary region of a reactor plasma, the problems of edge plasma control, impurity sources and sinks, transient effects, recycling, and exhaust must be addressed simultaneously. The STE will be the optimal facility for this task.

VII. Plasma Control

A necessary feature of a successful tokamak reactor will be the ability to exercise local control of plasma conditions. Particularly important are the current, density, and temperature profiles, and the boundary conditions at the plasma surface. The STE program will concentrate on developing and testing techniques for control, as well as demonstrating the benefits gained in improved confinement and higher β limits.

The control of plasma profiles and boundary conditions is crucial for operation of a steady-state tokamak reactor. The stability and transport properties depend on the distribution of current, temperature and pressure within the plasma, and on the flux and energy distribution of neutral particles entering the plasma at its boundary. Further, the achievement of optimum performance in a tokamak will depend on maintaining an appropriate plasma shape, and on controlling the location of the magnetic separatrix that defines the divertor. Such plasma shapes can only be achieved with a tailored plasma current distribution.

Current Profile Control

In all tokamak devices to date, the profile of the toroidal plasma current is closely coupled to the temperature profile which, in turn, is determined by the heat sources and loss mechanisms. With localizable noninductive current drive, the current distribution within the plasma can be decoupled from the temperature. This technique will lead to profiles with improved stability and reduced transport losses.

Recent results for JT-60¹ have shown that with lower hybrid current drive, the energy confinement can improve relative to inductive drive. This improvement is attributed to the generation of current profiles with better transport and stability properties than those which follow from the ohmic temperature profile and Spitzer conductivity. In addition, the DIII-D neutral beam current drive experiment has produced quiescent plasmas free of MHD activity.²

There are a number of changes that can be made in plasma stability using current profile control. For example, it can be used to modify the period of "sawtooth"

¹ JT-60 Team, *Energy and Particle Confinement of JT-60 Plasmas with High Power Heating*, to be published in *Plasma Physics and Controlled Nuclear Fusion Research* (Proc. 11th Int. Conf. Kyoto, 1986), IAEA, Vienna (1987), A-II-2.

² T. C. Simonen *et al.*, *Neutral Beam Current Driven High Poloidal Beta Operation of the DIII-D Tokamak*, General Atomics Report GA-A19182, February 1988, submitted to Phys. Rev. Letters.

oscillations, and thus regulating the transfer of thermal energy from the core to the periphery of a plasma. It can prevent disruption by stabilization of tearing modes.³ The overlapping of low mode number tearing mode magnetic islands is generally believed to be the cause of major disruptions in tokamaks. Edge localized modes (ELMs) have been identified with confinement degradation in H-mode plasmas and edge heating or current drive may be used to suppress ELMs. If the ELMs are ballooning modes at the edge, as suggested by some theories, driving a current at the edge can stabilize these modes and may push the plasma into the second stability region.

If confinement time improves with plasma current as the major tokamak experiments now suggest, there is considerable incentive for operation at very high current, even to the extent that the safety factor on-axis q_0 is below 1, provided the plasma remains stable. With external shaping and global current profile control, theory has suggested that ideal MHD modes can be stabilized for $q_0 > 0.65$. If, in addition, resistive modes can be stabilized by local current profile control, a stable low- q operating regime will be possible for STE.

Electron cyclotron heating and current drive have a number of advantages when used for this purpose. The absorption physics is well understood and the heat deposition is well localized, as indicated by both theory and experiment. In experiments thus far, confinement degradation with increasing EC power has been minimal. EC systems also have the particular technological advantage of short wavelength, requiring only waveguide systems of small dimensions.

Pressure Profile Control

Pressure profile control is another important aspect of the STE experiment. Analogous to current profile control, local pressure profile modification may be used to stabilize pressure driven modes such as ballooning modes which may control both the β limit and the confinement properties for high β plasmas. For example, in diverted plasmas, edge-localized modes (ELMs) that are thought to be ballooning instabilities limit the energy confinement in the H-mode. These burst-like instabilities also generate very high transient heat loads on divertor surfaces. The giant ELMs are correlated with the steepening of edge pressure gradient. With additional edge heating, both the amplitude and the frequency of ELMs have been controlled in some

³ V. S. Chan and G. Guest, *Stabilization of Internal Kinks in Tokamaks Using Electron Cyclotron Heating*, Nucl. Fusion **22** (1982) 272.

experiments. Clearly, edge plasma control is necessary to preserve or even improve on the good confinement properties of the H-mode.

Pressure profile control in the bulk plasma can be achieved straightforwardly by local heating or by particle fueling. Pellet injection has been demonstrated experimentally to peak up density profiles sufficiently to stabilize ion temperature gradient modes, a prime candidate responsible for anomalous transport in high density regimes. For reactor-like parameters, pellets ablate quickly and penetration to the center of the plasma becomes a severe problem. Either higher velocity pellet injectors (to ~ 10 km/s) are needed or alternative fueling systems such as compact toroid injection must be developed.

Boundary Region Control

Another challenge is controlling the plasma edge. The edge conditions are important to both the plasma and the material walls. Reduction of the particle flux into the plasma with a divertor leads to the transition to the H-mode confinement regime. However, associated with this regime is an improved particle confinement that leads to impurity accumulation in present experiments. To control impurities, the generation of impurities at their source (the walls) and the flux of impurities into the plasma should be regulated.

Among possible techniques for control of particle and energy flow in the boundary region are biasing the divertor and driving edge current. With the availability of high power rf sources, another idea is to use radiation to directly influence the particle flux. The idea is when a travelling wave is launched into the plasma, the particle distribution is distorted asymmetrically in toroidal momentum space. This alters the parallel frictional force and results in a radial particle flux. The direction of this flux depends on the direction of the travelling wave. A preliminary study with ion cyclotron waves shows that the ion Bernstein wave is more effective in influencing the particle flux than the fast wave. Furthermore, a poloidally asymmetric wave number spectrum is needed. This presents a unique requirement for antenna design.

Finally, high power systems open up the possibility of creating anisotropic plasma pressure. Theory has shown that for some very specific pressure anisotropy, it is possible to influence the stability of ballooning modes. If this is achievable, one may consider second stability regime operation in steady-state.

VIII. Role in the U.S. and World Program

STE will play a unique role in the U.S. and world fusion programs in the 1990's, because it will integrate the features of steady-state noninductive current drive, stable high- β operation using advanced shaping and profile control, and continuous H-mode confinement and impurity control with steady-state, high heat flux plasma interactive components.

The key elements of the STE program are (1) steady-state, noninductive operation, (2) shape, profile, and boundary region control for high β , impurity control, and good confinement, and (3) development and demonstration of high heat flux components for continuous operation in a reactor-like tokamak environment. While each of these features will be explored in existing or soon-to-be operating facilities, it is the crucial interaction between them that requires an integrated test in a new facility (STE).

For reference, a short list of relevant world fusion facilities that will be operating in the 1990's are listed below, along with a representative sample for STE.

Major Tokamak Facilities (by year of operation)

	TFTR	JFT-2M	Textor	JET	JT-60	DIII-D
location	USA	Japan	FRG	EC/UK	Japan	USA
start operations	1982		1982	1983	1985	1986
major radius (m)	2.5	1.3	1.8	3.0	3.1	1.7
minor radius (m)	0.85	0.35	0.5	1.25	0.9	0.65
elongation	1.0	1.7	1.0	1.6	1.0	2.0
toroidal field (T)	5.2	1.4	2.6	3.5	4.5	2.2
plasma current (MA)	2.5	0.6	0.6	5.0	2.7	3.5
heating power (MW)	30	7	5.5	38	40	14
flat-top time (sec)	3	0.5	3	15	5	5
edge	lim	div	p.l.	div	div	div
β limit ($3I/aB$) (%)	1.7	3.7	1.4	3.4	2.0	7.3
density limit ($I/\pi a^2$)	1.1	1.6	0.8	1.0	1.1	2.6
program emphasis	high p conf. D-T oper.	rf shaping	boundary	high β conf. D-T oper.	high p conf. divertor	high β conf. shaping

	PBX-U	T-15	Tore Supra	FT-U	C-Mod	
location	USA	USSR	France	Italy	USA	
start operations	1987	1988	1988	1988	1989	
major radius (m)	1.6	2.4	2.3	0.93	0.65	
minor radius (m)	0.30	0.70	0.75	0.31	0.21	
elongation	2.3	1.0	1.0	1.0	1.8	
toroidal field (T)	2.2	5.0	4.5	8.0	9.5	
plasma current (MA)	0.8	2.3	1.7	1.6	3.0	
heating power (MW)	11	14	20	8	5	
pulse time (sec)	1	5	30	1.5	1	
edge	div	lim	p.l.	lim	div	
β limit ($3I/aB$) (%)	3.6	2.0	1.5	1.9	4.5	
density limit ($I/\pi a^2$)	2.8	1.5	1.0	5.3	21.7	
program emphasis	high β 2nd stab.	high p conf.	long pulse rf	high B rf	high B rf	
	Asdex-U	MTX	JT-60U	T-14	CIT	STE
location	FRG	USA	Japan	USSR	USA	USA
start operations	1989	1989	1991		1996	
major radius (m)	1.65	0.64	3.4	1.25	2.1	2.0 (1.2-3.0)
minor radius (m)	0.50	0.12	1.0	0.27	0.65	0.5 (0.33-0.60)
elongation	1.6	1.0	1.5	1.7	2.0	2.0 (1.5-3.0)
toroidal field (T)	4.0	9.0	4.2	14.5	10.0	5.0 (4.0-6.0)
plasma current (MA)	2.0	0.4	6.0	3.9	11.0	3.5 (3.0-5.0)
heating power (MW)	12	4	65	20	20	35 (15-50)
flat-top time (sec)	5	0.5	5	2	5	∞
edge	div	lim	div		div	div
β limit ($3I/aB$) (%)	3.0	1.1	4.3	3.0	5.1	4.2
density limit ($I/\pi a^2$)	2.5	8.8	1.9	17.0	8.3	4.5 (3.2-7.4)
program emphasis	conf. impurities	rf	conf. heating	high B ignition	ignition α phys.	steady-state high power control

The only new fusion facility in the world that would equal the integrated physics and technology test capabilities of STE is the proposed International Thermonuclear Experimental Reactor (ITER). ITER is expected to begin operation around 2000, later than any other device listed here. In a sense, the operation of ITER would integrate the features and results of the STE program with the physics and technological aspects of D-T burning (i.e., ITER can be viewed as a complete integration test, at an estimated cost of 4-6 billion \$).

The overall objective of ITER is to demonstrate the scientific and technical feasibility of fusion power, with capability for both controlled ignited burn and steady-state operation. A balanced domestic U.S. program as called for in the DOE Magnetic Fusion Program Plan and the Technical Planning Activity could enhance these twin ITER goals with two major new U.S. facilities in the mid-1990's: a copper-coil D-T burning device (CIT) for demonstrating the feasibility of ignited operation, and a superconducting-coil hydrogen device (STE) for demonstrating the feasibility of steady-state operation. Since the present data base is being used to design CIT, STE, and ITER, the benefit for ITER of early experience with ignition and steady-state operation in CIT and STE will be to enhance operations and to provide much more flexible test-beds for physics and technology developments. Data from CIT and STE could also come in time to guide selection of additional plasma heating, fueling, current drive, and control systems on ITER.

Regardless of any decision to build ITER, or of the ITER schedule, demonstrations of ignition in CIT and steady-state in STE would, by themselves, constitute early and essential milestones on the path to any practical tokamak power reactor. Ignition in CIT would be essential to understanding the effects of dominant α -particle heating. Achievement of a practical steady-state reactor entails several critical issues in addition to the role of α -particles and basic current drive efficiency, however. An issue central to the feasibility of any steady-state reactor, or even of pulsed reactors that would need very long pulses to be practical, is the question of whether or not the optimum current profile shapes necessary to realize the high β for economical operation can be continuously maintained. Another vital issue is the question of whether or not plasma impurity levels can be controlled in a stationary state, at relevant high heat flux levels on the first walls and divertors.

The world fusion program currently lacks a facility to address these critical issues in the mid-1990's, particularly a steady-state device with diverted H-mode plasmas at relevant high elongation and β values. France's Tore Supra and Russia's T-15,

superconducting-coil tokamaks with circular cross-section, will not likely achieve a magnetic divertor separatrix for impurity control, nor an H-mode for good confinement, nor β much greater than 1%, nor currents much larger than 1 MA, which are major features expected to be needed in ITER or in any practical tokamak reactor. Just now coming into operation, Tore Supra and T-15 will significantly extend the tokamak data base towards longer pulse operation. By 1994-5, when STE would be expected to begin operation at the earliest, Tore Supra and T-15 will have been operated six years or more, ample time to accomplish what these machines are capable of. STE would then extend the results of Tore Supra and T-15 to higher current and temperature, even longer times (in steady-state), and add the important feature of high-heat-flux divertors, H-mode, and higher β through shaping and profile control.

The issue of profile control using noninductive current drive and local heating and fueling can be broken into three sub-issues, the first two of which can (and should) be addressed to a considerable extent in present machines. 1) In existing pulsed experiments, current drive techniques which can be extended to steady-state can be used to show how to initiate the current profile necessary for high β . 2) In existing pulsed experiments, driver techniques can be tested, which extrapolate to a power efficiency suitable for reactor application. 3) In a steady-state experiment with reactor-like properties, the driven plasma must be shown to maintain the requisite profile for times long compared to the L/R time, impurity accumulation time, and all other characteristic relaxation times. It is this third subissue which requires a steady-state, superconducting-coil device such as STE, with the cross-section shape, current capability, and heating power to achieve reactor temperature, β , and heat flux levels.

Short pulse demonstration of noninductive current drive forms part of the necessary data base for initiating STE. A wide variety of systems are presently under investigation. One system is the use of high energy neutral beams or electron cyclotron frequency radiation to drive current in the plasma core, and lower hybrid frequency radiation for current drive in the edge region. This combination can meet the minimum efficiency requirement for STE, give sufficient profile control, and have adequate penetration. It requires the development of high power sources (150-500 keV beams or 150 GHz rf sources) not presently available. Another system is the use of fast wave rf power at a high harmonic of the ion cyclotron frequency. Theory indicates good penetration and current drive efficiency, and high power sources (few 100 MHz) are readily available. A high power experimental test is required.

The capability for carrying out the required development and testing exists in the present U.S. program. High energy neutral beams and high frequency rf sources are now under development. The testing of current drive techniques and demonstration of totally noninductive drive at high β are the key components of the DIII-D tokamak program for the next five years. In addition, data on driver development will come from MTX, Alcator C-Mod, and TFTR, as well as foreign tokamaks.

The lead-time for the STE would be 5-6 years from project initiation to initial operation, approximately 1-2 years for detailed design and about 3-4 years for construction and installation. To provide the necessary support for ITER operations, STE would require a start of construction about 1991-2, with operation beginning 1995.