A program toward a fusion reactor*

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(Received 5 December 1990; accepted 19 April 1991)

Near breakeven conditions have been attained in the JET tokamak [Fusion Technol. **11**, 13 (1987)], with beryllium as the first-wall material. A fusion triple product ($n_D \tau_E T_i$) of 8–9$\times 10^{20}$ m^{-3} sec keV has been reached (within a factor of 8 of that required in a fusion reactor). However, this has only been achieved *transiently*. At high heating powers, an influx of impurities still limits the achievement of better performance and steady-state operation. In parallel, an improved quantitative understanding of fusion plasmas has emerged from the development of a particular plasma model. Good quantitative agreement is obtained between the model and JET data. The main predictions are also consistent with statistical scaling laws. With such a model, a predictive capability begins to emerge to define the parameters and operating conditions of a DEMO, including impurity levels. Present experimental results and model predictions suggest that impurity dilution is a major threat to a reactor. A divertor concept must be developed further to ensure impurity control before a DEMO can be constructed. A New Phase for JET is planned in which an axisymmetric pumped divertor configuration will be used to address the problems of impurity control, plasma fueling, and helium ash exhaust. It should demonstrate a concept of impurity control and the operational domain for such a device. A single Next Step facility (ITER) is a high risk strategy in terms of physics, technology, and management, since it does not provide a sufficiently sound foundation for a DEMO. A Next Step program is proposed, which could comprise several complementary facilities, each optimized with respect to specific clear objectives. In a minimum program, there could be two Next Step tokamaks, and a Materials Test Facility. Such a program would allow division of effort and sharing of risk across the various scientific and technical problems, permit cross comparison, and ensure continuity of results. It could even be accomplished without a significant increase in world funding.

I. INTRODUCTION

Present fusion research programs are directed ultimately to the construction of a Demonstration Fusion Reactor—DEMO, which should be a full ignition, high-power device. Moderate extrapolation of latest results and considerations of model predictions, taken together with the constraints of present technology, allow the size and performance of a thermonuclear reactor to be largely defined.

The Joint European Torus (JET)[1–3] has successfully achieved and contained plasmas of thermonuclear grade, and has reached near breakeven conditions in single discharges. JET is well placed to oversee the route to a DEMO. Sections II and III of the paper set out the main results achieved in JET.

Furthermore, a clearer picture of energy and particle transport begins to emerge. The critical electron temperature gradient model is a particular model consistent with JET results. The implications of this model are considered in Sec. IV and predictions are compared with JET results.

All aspects of plasma behavior, impurity control, and plasma exhaust must be included in the model used to define the size, toroidal field, plasma current, and operating conditions of a DEMO reactor (see Sec. IV). Such a reactor would produce power in the range 3–6 GW thermal or 1–2 GW electrical power. Most critical is the control of impurities and the exhaust of helium ash at high power. An acceptable solution is most likely to be achieved with a divertor at high densities.

To consolidate the model for density and impurity control, a New Phase is planned for JET with an axisymmetric pumped divertor configuration to operate with a stationary plasma (10 sec to 1 min) of thermonuclear grade. This is described briefly in Sec. V.

The Next Step must bridge the gap from present knowledge to that required to construct a DEMO. This demands significant technological and scientific developments (see Sec. VI). These developments are expected to mature on varying time scales. Attempting to cover them all in a single device will limit the domain of investigation and lead to unacceptable risk of failure, unless a large safety margin is allowed on each component. This would impact on the start time and construction time, with consequential effect on costs.

A Next Step program is needed to address these issues. It seems prudent to envisage international collaboration. The program could comprise several complementary components, each optimized with respect to specific clear objectives. There could be two Next Step tokamaks, and a Materials Test Facility, each constructed in different locations with differing start dates. This theme is developed further in Sec. VII, where details are presented of such a program, and its time scale and cost.

*Paper 1RV1, Bull. Am. Phys. Soc. **35**, 1917 (1990).
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II. JET PERFORMANCE RELEVANT TO A DEMO

JET is a high current, high-power tokamak with a low-Z first wall.[1-3] Furthermore, JET operates with the configuration foreseen for a Next Step tokamak. The plasma current of 7 MA in the limiter configuration[4] and the current duration of up to 30 sec at 3 MA are world records and are over twice the values achieved in any other fusion experiment. Also, 5.1 MA and 4.5 MA are the highest currents achieved in the single-null and double-null divertor configurations, respectively.[5] Neutral beam injection (NBI) heating has been brought up to full power (~ 21 MW) and ion cyclotron resonance heating (ICRH) power has been increased to ~ 22 MW in the plasma. In combination, these systems have delivered 35 MW to the plasma.

The overall fusion triple product as a function of central ion temperature is shown in Fig. 1 for JET and a number of other tokamaks.

A. Fusion performance

Better plasma purity was achieved in JET with a beryllium first-wall, by sweeping the X point and by using strong gas puffing in the divertor region. This resulted in high ion temperatures [$T_i(0)$ in the range 20–30 keV] and improved plasma performance, with the fusion triple product [$n_D(0)\tau_E T_i(0)$] increasing significantly. Such improved fusion performance could otherwise have been achieved only with a substantial increase in energy confinement.

In the hot-ion H-mode regime, the central ion temperature reached 22 keV, the energy confinement time, τ_E, was 1.1 sec, with a record fusion triple product [$n_D(0)\tau_E T_i(0)$] of $8\text{–}9\times 10^{20}$ m^{-3} sec keV. The neutron yield for this discharge was also amongst the highest achieved on JET at 3.8×10^{16} sec^{-1}, with $Q_{DD} = 2.4\times 10^{-3}$. A full D–T simulation of the pulse showed that 12 MW of fusion power would have been obtained transiently with 16 MW of NBI power, giving an equivalent fusion amplification factor $Q_{\rm DT} \sim 0.8$, reaching near breakeven conditions and within a factor of 8 of that required by a reactor. Similar results with ICRF were also obtained at medium temperatures, with $T_e \sim T_i \sim 10$ keV.

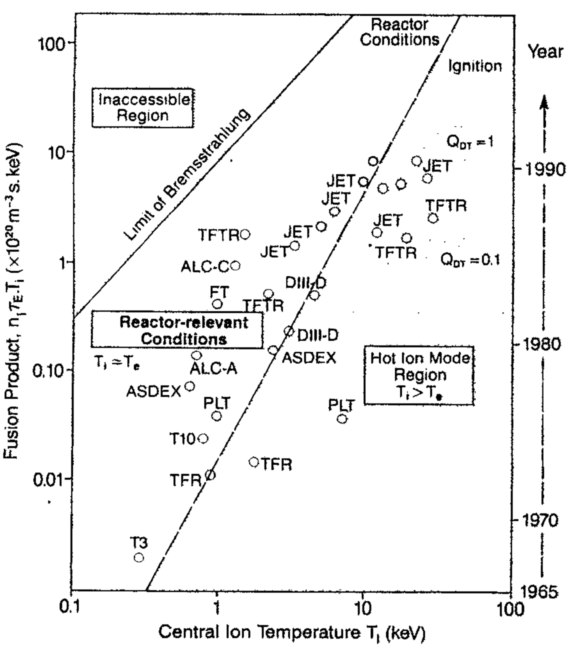


FIG. 1. Overall performance of the fusion product [$n_D(0)\tau_E T_i(0)$] as a function of central ion temperature [$T_i(0)$], for a number of tokamaks.

However, the best fusion performance was obtained only *transiently* [for one confinement time (~ 1 sec)] and could not be sustained in steady state. Ultimately, the influx of impurities caused a degradation in plasma parameters. In fact, density and impurity control have been the main obstacles limiting JET performance.

B. General plasma performance

As a result of reduced impurity levels with a beryllium first-wall, the general JET performance improved and operation at higher densities was prolonged. In particular:

The pumping of deuterium with beryllium was more efficient than with carbon walls and provided improved density control. This permitted low density and high temperature (up to 30 keV) operation for times > 1 sec;

the density limit increased,[6] and a maximum peak density of 4×10^{20} m^{-3} was achieved with pellet fueling. The density is limited principally by fueling and not by disruptions, as was found with carbon limiters;

sawteeth were stabilized for periods exceeding 8 sec;

in the absence of sawteeth, the central electron temperature saturates at ~ 12 keV, even though the central power to the electrons appears to exceed that to the ions;

with beryllium antenna screens, H modes were obtained for periods > 1 sec using ICRH alone. The confinement characteristics were similar to those with NBI alone;

β values up to the Troyon limit were obtained in double-null X-point plasmas.[6] Thus, in JET, the parameters of a reactor plasma have been reached individually, and near-breakeven conditions have been achieved in single discharges.

III. QUANTITATIVE UNDERSTANDING OF FUSION PLASMAS

The underlying JET results are presented with particular emphasis on their significance for the formulation of a model of transient and steady-state plasmas that can be used for moderate extrapolation to a Next Step tokamak.

A. Density limit

With a beryllium first-wall, the maximum operating density increased significantly by a factor 1.6–2 compared with a carbon first-wall. Furthermore, the nature of the density limit changed and the frequency of disruptions at the density limit were much reduced. Disruptions did not usually occur, and the limit was associated rather with the formation of a poloidally asymmetric, but toroidally symmetric radiating structure (a "MARFE"), which limits the plasma density to within the stable operating domain.

Heating and plasma fueling were varied systematically, using both gas and pellet fueling. With deep pellet fueling and either NBI or ICRH, peaked profiles were obtained (Fig. 2). Just before a density limit MARFE occurred, pellet-fueled discharges reached the same edge density as gas-fueled discharges, but the central densities were considerably higher. The central density depends, therefore, on the fueling method used. The profiles are similar near the edge, but are remarkably flat with gas fueling.

These observations suggest that the edge density may be correlated with the density limit and is found to increase approximately as the square root of power (Fig. 3). This endorses the view that the density limit is determined by a power balance at the plasma edge and the cause of disruptions is related to radiation and charge exchange near the $q = 2$ surface. Thus, where beryllium is the only impurity and when the radiation is low, and confined to the outermost edge, density limit disruptions are not observed.

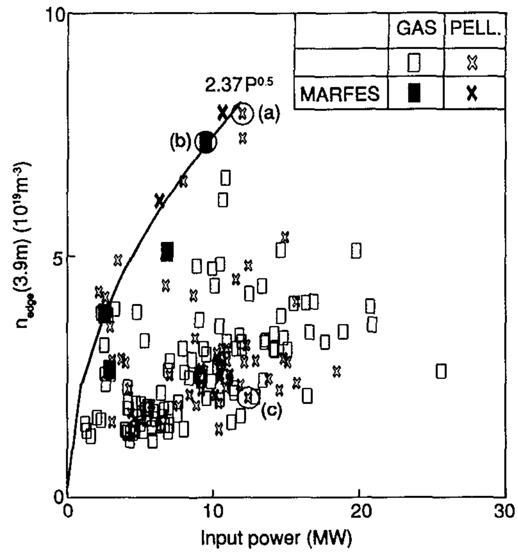


FIG. 3. The edge electron density (n_{edge}) versus input power (P) showing that the density limit occurs at the boundary of the operational domain close to the curve n_{edge} ($\times 10^{19}$ m^{-3}) $= 2.37P^{1/2}$ (MW). The profiles shown in Fig. 2 correspond to the three data points (a), (b), and (c).

B. Density profiles and transport

Density profiles obtained with edge fueling tend to be flatter for the lower $Z_{\rm eff}$ and higher density achieved with beryllium, in contrast to those obtained with carbon, which tend to be more peaked, even with edge fueling. The occurrence of flat density profiles suggests that there is no need for an anomalous inward particle pinch, except, perhaps, on impurities and in the edge.

The relaxation of the peaked density profiles achieved with pellet injection allows an estimate of particle transport. For a 4 mm pellet injected into a 3 MA/3.1 T plasma, the decay of the central electron density is shown in Fig. 4. Following injection, the decay constants are 1.8 sec for the Ohmically heated discharge and 1 sec when ~ 8 MW ICRH is applied. The global energy confinement times are in a similar ratio. Therefore, it is reasonable to assume that particle and energy transport are linked. The measurement of emissivity profiles by the soft x-ray cameras following the injection of laser-ablated, high-Z impurities provides evidence of better impurity confinement in H modes. The temporal evolution of Ni XXVI emission is shown in Fig. 5 for the L and H phases of two similar discharges with ~ 9 MW of additional heating. In contrast to the decaying signal of the L phase, the signal rises rapidly to a steady value which persists to the end of the H phase. This shows that impurities have considerably longer confinement times in the H phase and endorses the view that an edge transport barrier exists, which could be destroyed (for example, by edge localized modes) and on transition from the H to the L phase.

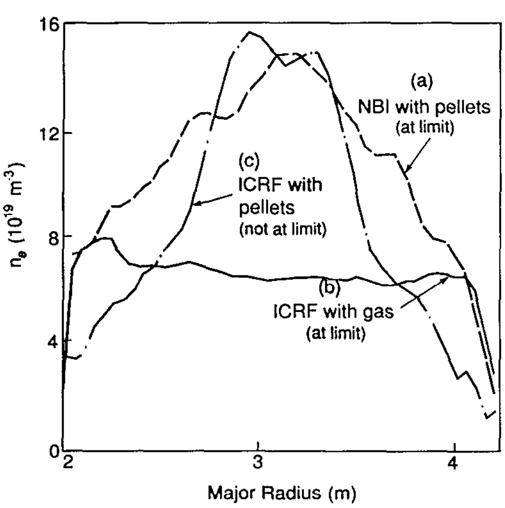


FIG. 2. Electron density profiles for different fueling and heating methods. Profiles (a), (b), and (c) are correlated with their proximity to the density limit (see Fig. 3).

C. Temperature

High ion temperatures have been obtained at the low densities possible with a beryllium first-wall and with the

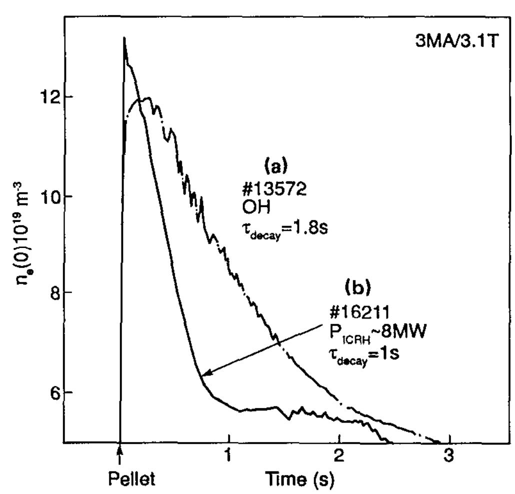


FIG. 4. Decay of central density following pellet injection into discharges with (a) Ohmic heating only and (b) ~ 8 MW ICRH.

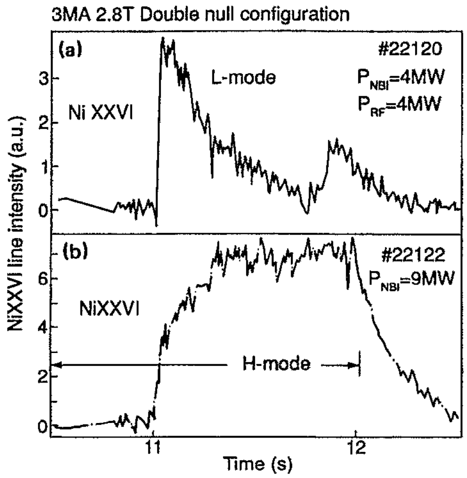


FIG. 5. Temporal evolution of NiXXVI emission in (a) the L phase and (b) the H phase of two similar discharges with ~ 9 MW of additional heating.

better penetration afforded by NBI at 140 kV. Maximum ion temperatures were achieved of up to 18 keV in limiter plasmas and up to 30 keV in X-point plasmas (with powers up to 17 MW). In this mode, the ion temperature profile is sharply peaked and the electron temperature is significantly lower than the ion temperature, by a factor of 2–3. The central ion temperature (as shown in Fig. 6) increases approximately linearly with power per particle up to the highest temperatures, indicating that ion thermal losses are anomalous, but ion confinement degrades little with input power. On the other hand, the central electron temperature saturates at ~ 12 keV, even though with ICRH the central heating power to the electrons appears to be higher than that to the ions. Electron thermal transport is also anomalous and electron confinement degrades strongly with increased heating power. This suggests that electrons are primarily responsible for confinement degradation. However, this does mean that ion losses are necessarily small.

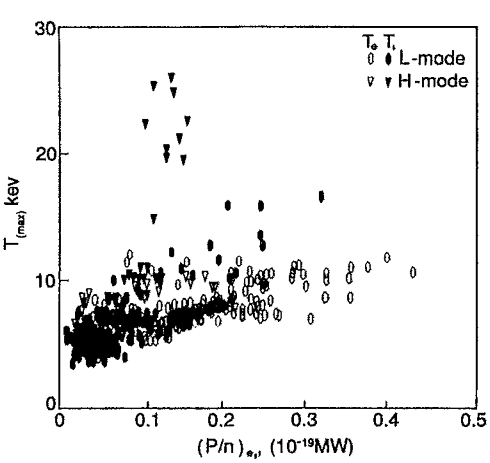


FIG. 6. Central ion (T_i) and electron (T_e) temperatures as functions of power per particle $(P/n)_{e,i}$ to either species.

D. Electron heat and density pulse propagation

The propagation of temperature and density perturbations following the collapse of a sawtooth provide good measurements of energy and particle transport. The decay of the temperature perturbation at different radii in a 3 MA/3.1 T Ohmically heated discharge is shown in Fig. 7(a). This can be modeled with a heat pulse diffusivity, $\chi_{HP} \sim 3.2\ \mathrm{m^2\,sec^{-1}}$, compared with $\chi_e \sim 1\ \mathrm{m^2\,sec^{-1}}$, obtained from power balance considerations. The results in an L-mode plasma, heated with 9.5 MW of ICRH, are shown in Fig. 7(b) and indicate that, although $\chi_e \sim 2\ \mathrm{m^2\,sec^{-1}}$, the same $\chi_{HP} \sim 3.2\ \mathrm{m^2\,sec^{-1}}$ can be used in the simulation to fit the data. Within experimental uncertainties, the same χ_{HP} can be used also for H-regime plasmas and does not depend on heating power.

Simultaneous measurements of the temperature and density perturbations indicate that the particle pulse diffusion coefficient, $D_{PP} \sim D_e \ll \chi_{HP}$.[7] This general behavior must be reproduced by the model.

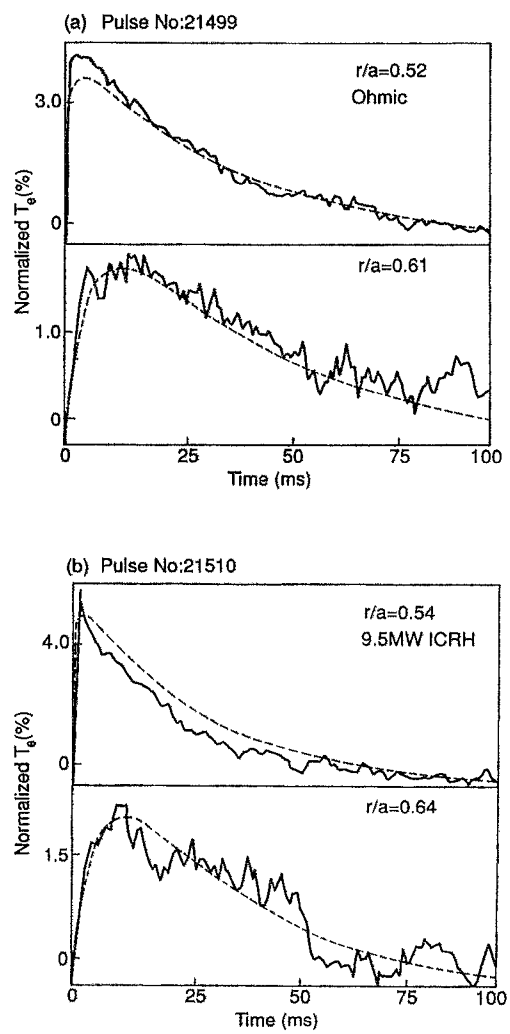


FIG. 7. Temporal evolution of electron temperature perturbations (normalized to the central electron temperature prior to the collapse of a sawtooth) at different radii for 3 MA/2.8 T discharges with (a) Ohmic heating only and (b) 9.5 MW ICRH. Dashed lines are from model calculations using $\chi_{HP} = 3.2\ \mathrm{m^2\,sec^{-1}}$.

E. Global energy confinement

With a carbon first-wall, the energy confinement time improves with increasing current and degrades with increasing heating power, independent of the heating method. With a beryllium first-wall, energy confinement times and their dependences are effectively unchanged: energy confinement does not appear to be affected by the impurity mix (carbon or beryllium in deuterium plasmas).

In the X-point configuration, high-power H modes (up to 25 MW) have been studied. In comparison with limiter plasmas, confinement is a factor ~ 2 better, but the dependences with current and heating power are similar (Fig. 8). These observations are consistent with the same basic mechanism applying over most of the radius, except perhaps near the very edge.

F. Plasma beta

Experiments have explored the plasma pressure (indicated by the β value) that can be sustained in JET and investigated the plasma behavior near the expected β limit in the double-null H-mode configuration at high density and temperature and low magnetic field ($B_t = 1$ T). Values of β_t up to $\sim 5.5\%$ were obtained, close to the Troyon limit $\beta_t(\%) = 2.8 I_p(\mathrm{MA})/B_t(\mathrm{T})a(\mathrm{m})$, where I_p is the plasma current and a is the plasma minor radius.[6] These results are consistent with the highest values obtained on DIII-D.[8] In JET, the limit does not appear to be disruptive at present power levels. Rather, a range of MHD instabilities occur, limiting the maximum β value without causing a disruption. The behavior near both the density and β limits may be interpreted in terms of resonant instabilities which have the magnetic topology of an island. However, ignition will be limited by the capability of the first wall and the divertor rather than by β.

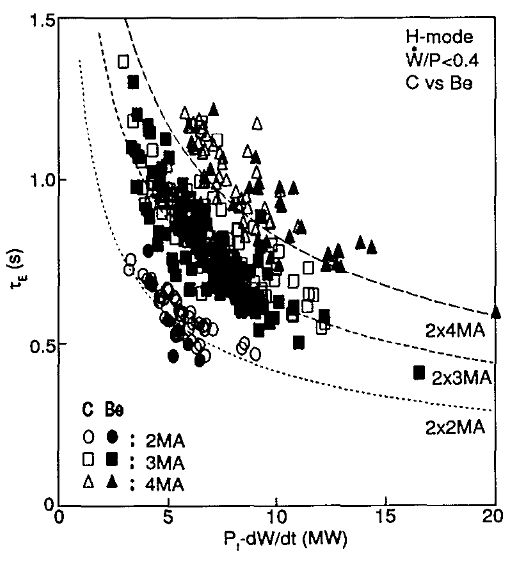


FIG. 8. Global energy confinement time (τ_E) during the H mode as a function of net input power for different plasma currents and first-wall materials.

G. Alpha-particle simulations

The behavior of alpha particles has been simulated in JET by studying energetic particles such as 1 MeV tritons, and ^{3}He and H minority ions accelerated to a few MeV by ICRH.[9] Triton burn-up studies show that the experimental measurements of the 14 MeV neutron rate is in good agreement with that calculated from the 2.5 MeV neutron (and hence the 1 MeV triton) source rate and classical thermalization. The energetic minority ion population with ICRH has up to 50% of the stored energy of the plasma and possesses all the characteristics of alpha particles in an ignited plasma, except that in the JET experiments the ratio of the perpendicular to parallel pressure was above three, while in a reactor plasma the distribution will be approximately isotropic. The mean energy of the minority species was about 1 MeV, and the relative concentration of the ^{3}He ions to the electron density was 1%–2%, which is comparable to the relative concentration of alpha particles in a reactor (7%). Under conditions with little magnetohydrodynamic (MHD) activity, no evidence of nonclassical loss or deleterious behavior of minority ions was observed, even though the ratio of fast ion slowing down time to energy confinement time in JET is greater than that expected in a reactor. The prospects for alpha heating in DEMO should therefore be good.

IV. A TRANSPORT MODEL

A. Formulation of a plasma model

Any model used to predict the performance of a Next Step tokamak must be consistent with the foregoing JET data and with physics constraints. Experimental observations support a model for anomalous transport based on a single phenomenon and MHD limits. This *critical electron temperature gradient* model of anomalous heat and particle transport features electrons which determine the degree of confinement degradation; ion anomalous transport with heat diffusivity χ_i linked to electron heat diffusivity χ_e; anomalous particle diffusivities, D, for ions and electrons, proportional to χ; and anomalous particle "pinch" for impurities alone.

Specifically, above a critical threshold, $(\nabla T_e)_c$, in the electron temperature gradient, the transport is anomalous and greater than the underlying neoclassical transport. The electrons are primarily responsible for the anomalous transport, but ion heat and particle transport are also anomalous. The general expressions for the anomalous conductive heat fluxes are

$$\begin{aligned}
Q_e &\equiv -n_e\chi_e\nabla T_e \\
&= -n_e\chi_{an,e}[\nabla T_e - (\nabla T_e)_c]H(\nabla q), \\
Q_i &\equiv -n_i\chi_i\nabla T_i, \\
\chi_i &= 2\chi_e\sqrt{T_e/T_i}\times[Z_i/\sqrt{(1+Z_{\mathrm{eff}})}], \\
D_i &\approx 0.7\chi_i.
\end{aligned}$$

The critical electron temperature gradient model of Rebut *et al.*[10] specifies possible dependences for $\chi_{an,e}$ and $(\nabla T_e)_c$ and this is explored further in Ref. 7;

$$\chi_{an,e} = 0.5c^2\sqrt{\mu_0 m_i}\left(1 - \sqrt{\frac{r}{R}}\right)\sqrt{1+Z_{\text{eff}}}\left(\frac{\nabla T_e}{T_e} + 2\,\frac{\nabla n_e}{n_e}\right)\frac{q^2}{\nabla q}\,\frac{1}{B_T\sqrt{R}}\,\sqrt{\frac{T_e}{T_i}},$$

$$(\nabla kT_e)_c = 0.06\sqrt{\frac{e^2}{\mu_0 m_e^{1/2}}\,\frac{1}{q}\sqrt{\frac{\eta j B_T^3}{n_e(kT_e)^{1/2}}}}.$$

This model features a limitation in the electron temperature; electron heat pulse propagation with $\chi_{\text{HP}} \sim \chi_{an,e} > \chi_e$; no intrinsic degradation of ion confinement with ion heating power; particle decay following pellet injection and its dependence on heating power; similar behavior of particle and heat transport; and particle pulse propagation as observed with $D_{\text{PP}} \sim D_i \ll \chi_{\text{HP}}$.

Furthermore, in the plasma interior, the same model applies to the L and H regimes and particle and energy confinement improve together. However, at the edge of an H mode, an edge transport barrier forms and the transport might be classical over a short distance ($\sim$few centimeters). In fact, the H mode may be the "natural" consequence of the transport model, since $\chi_{an,e}$ depends on shear, and reduces toward zero near the separatrix. Furthermore, MHD activity reduces on making the transition from L$\rightarrow$H phase. This may imply the stabilization of some other instability at the edge, where the effect of impurity radiation and neutral influxes on MHD might be important in destroying, at least partially, the edge confinement barrier. This instability is apparently easier to suppress in an X-point configuration with high edge magnetic or rotational shear.

To complete the plasma model requires a description of the scrape-off layer (SOL), for which a rudimentary model is also included (see also Sec. V C). However, the spontaneous improvement in edge confinement has yet to be modeled.

B. Application of the model to JET

Several examples are shown to illustrate the applicability of this plasma model to JET experimental results.

A comparison between experimental and computed density and temperature profiles for a 3 MA/3 T discharge heated with 8 MW NBI is shown in Fig. 9. It is to be noted that the temperature profiles are well represented and that the electron density profiles are flat. The Z_{eff} of this discharge is low. No anomalous particle "pinch" is needed to describe adequately the density evolution.

The density and temperature profiles are also well represented for a 3 MA/3 T discharge heated with 8 MW ICRF (Fig. 10). However, it should be noted that the electron density profile is more peaked. The Z_{eff} of this discharge is also higher. An anomalous particle "pinch" is required and in the model this may be attributed to impurities.

The model has also been applied to the propagation of heat and density pulses following the collapse of a sawtooth in JET. As shown in Fig. 11, the computed changes in T_e and n_e at various radii and times agree well with experiment.

The characteristics of an H mode can also be modeled, assuming classical transport near the separatrix. As shown in Fig. 12, this leads to higher density and Z_{eff}, reduced edge flow (D_α signal simulated), higher stored energy and τ_E, lower χ_e and χ_i, as observed experimentally.

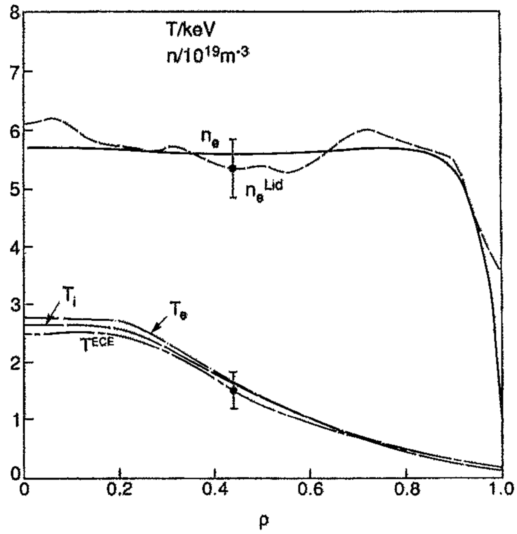


FIG. 9. A comparison between experimental and computed density and temperature profiles as a function of normalized radius, ρ, for a 3 MA/3 T discharge heated with 8 MW NBI.

Thus, we have established good quantitative agreement between the model and JET data. Furthermore, the main predictions of the model are consistent with statistical scaling laws.[10] With such a model, we begin to have the predictive capability needed to define the parameters and operating conditions of a DEMO, including impurity levels.

C. Application of the model to a first DEMO

The parameters of a first DEMO are defined by technology and physics predictions. The minor radius of a DEMO plasma needs to be twice the thickness of the tritium breed-

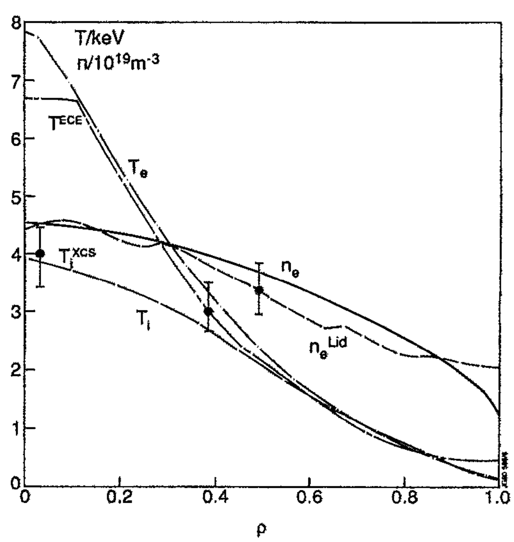


FIG. 10. A comparison between experimental and computed density and temperature profiles as a function of nomalized radius, ρ, for a 3 MA/3 T discharge heated with 8 MW ICRH.

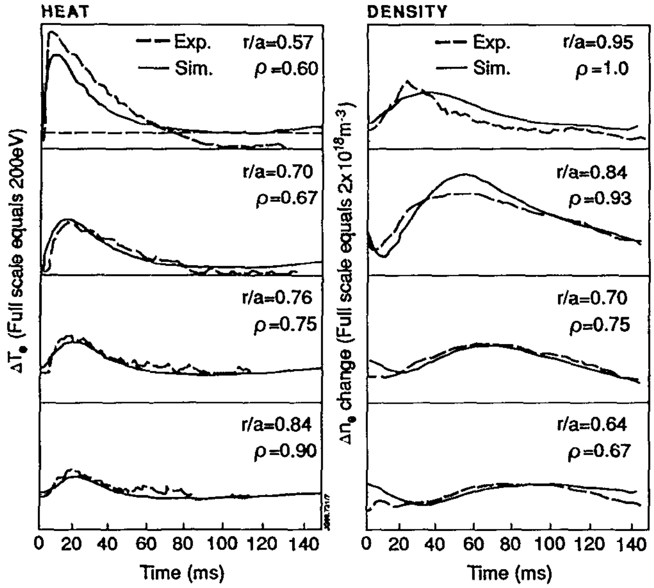


FIG. 11. Application of the model to the propagation of heat and density pulses following the collapse of a sawtooth in JET.

ing blanket thickness, which makes it ~ 3 m and the elongation can be up to 2. A practical aspect ratio of ~ 2.5 sets the plasma major radius to 8 m. Safe operation can be assumed for a cylindrical safety factor of 1.6–1.8. Plasma physics requirements can be fulfilled by operating at a toroidal field no greater than 5 T. This defines a reactor with a plasma current of ~ 30 MA.

The operating conditions will be such that $T_i \sim T_e$; confinement is L mode; D–T mixture, including helium ash; sawteeth (minimum internal control from outside); high density, flat density profile; full impurity control.

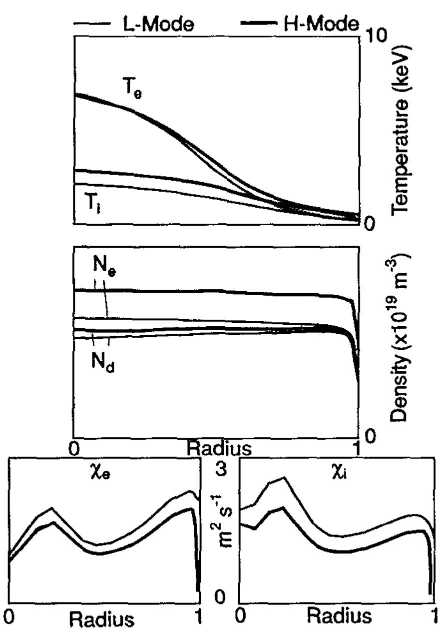


FIG. 12. The characteristics of an H mode modeled, assuming classical transport near the separatrix.

We now model DEMO plasmas using the L-mode transport model which has been tested against JET results, together with a model for sawteeth, β-limit instabilities, and the divertor. A mixture of D–T is assumed and includes helium ash accumulation and pumping. Assuming the provision of adequate impurity control, ignition is maintained in DEMO ($R_{\text{major}} = 8$ m, $a = 3$ m, 4.5 T, 30 MA, $\kappa = 2$) after switching off 50 MW of ICRH (Fig. 13). At ignition, the slightly hollow density profiles (Fig. 14) with edge fueling are sufficient to fuel the center. However, helium poisoning alone can quench the ignition without adequate pumping. To exhaust sufficient helium and maintain ignition requires the pumping of about *1 kg of D–T per hour. Exhaust and impurity control are essential.* In fact, while the H mode has short term benefits for approaching ignition, the long term deficiencies due to helium poisoning are evident (see Fig. 15). Steady ignition conditions can be achieved with a specific level of helium ash.

It is fundamental to control dilution in a reactor. There are two primary sources of dilution: target plate impurities and helium ash. With impurity control, ignition is achieved with edge fueling and high pumping; high density and flat profile; sawteeth and L-mode confinement; and recirculation of a kg of D–T per hour. The H mode does not improve steady-state ignition due to the better confinement of ashes. At present, impurity control is envisaged to require high-density operation. Under these conditions, the use of current drive seems precluded. However, one hour steady state requires only 50 V ($\sim 10\%$ of inductive volt seconds) and this can easily be provided.

A divertor concept for impurity control must be developed further. The fueling, impurity control, and exhaust capability of a Next Step will be dependent on whether deuterium and impurities (including helium) accumulate in the plasma center. The production and transport of helium ash toward the plasma edge (where it must be exhausted) will depend on the relative importance of energy and particle confinement, the effect of sawteeth, the effect of the edge transport barrier in the H mode, and the behavior of the scrape-off layer plasma.

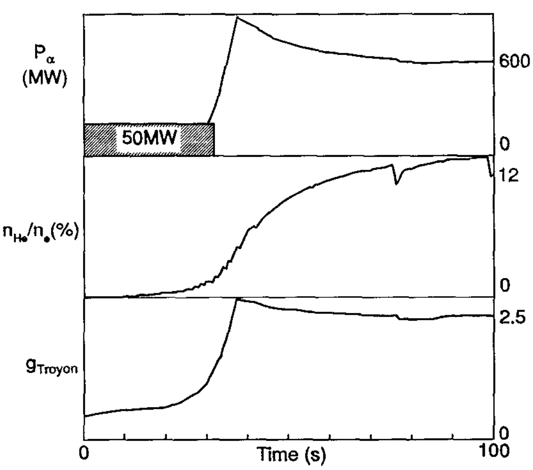


FIG. 13. Model of DEMO plasmas using the L-mode transport model which has been tested against JET results.

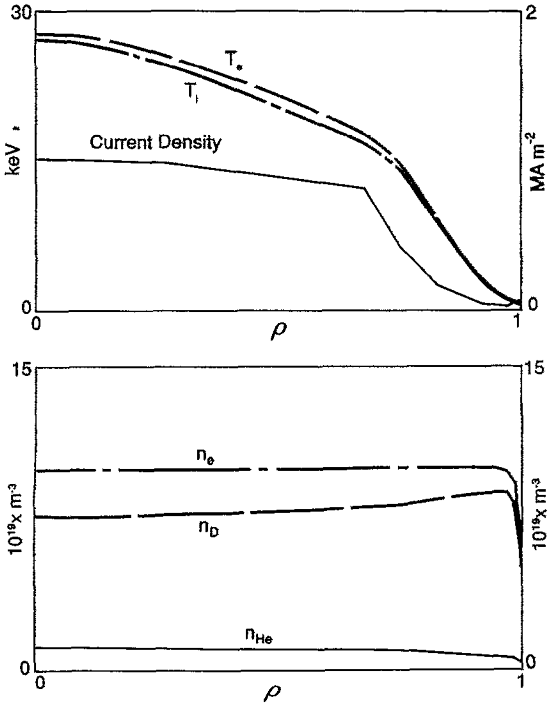


FIG. 14. DEMO profiles at ignition as a function of normalized radius ρ.

V. SOLUTION OF OUTSTANDING ISSUES

A. Impurity control and the new phase planned for JET

Plasma dilution is a major threat to a reactor. *Active* impurity control represents a solution to this problem, and this will be tested in JET in a New Phase planned to start in 1992.[3] First results should become available in 1993 and the project should continue to the end of 1996.

The aim of the New Phase is to demonstrate, prior to the introduction of tritium, effective methods of impurity control in operating conditions close to those of a Next Step tokamak, with a stationary plasma of "thermonuclear grade" in an axisymmetric pumped divertor configuration. Specifically, the New Phase should demonstrate the control

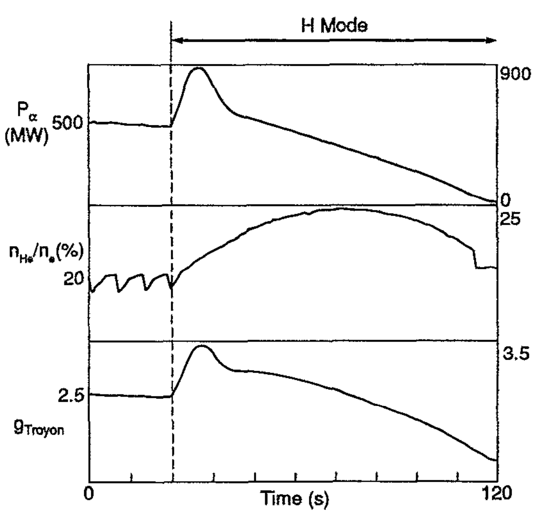


FIG. 15. Long term deficiencies due to helium poisoning in the H mode.

of impurities generated at the divertor target plates; a decrease of the heat load on the target plates; the control of plasma density; the exhaust capability; and a realistic model of particle transport.

B. Key concepts of the JET pumped divertor

Sputtering of impurities at the target plates presents a major difficulty and such impurities must be retained close to the target plates for effective impurity control. This retention may be accomplished by friction with a strong plasma flow directed along the divertor channel plasma (DCP) toward the target plates.[11-13] The plasma flow will be generated by a combination of gas puffing, injection of low-speed pellets, and recirculation of a fraction of the flow at the target plates toward the X point. The connection length along the magnetic field line between the X point and the target plates must be sufficiently long to allow effective screening of impurities.

C. Modeling the edge plasma

Calculations for JET New Phase[13] show that impurities can be retained near the target plates for plasma flows, typically $\sim 10^{23}\ \mathrm{sec}^{-1}$ near the X point. The steady-state distributions of Z_{eff} (with beryllium impurities), for conditions in the SOL and DCP, with and without flow, are shown in Fig. 16(a). These results are obtained for an electron density $\sim 10^{20}\ \mathrm{m}^{-3}$ at the target plates. At target densities approaching $10^{21}\ \mathrm{m}^{-3}$, the reduction of erosion and the plasma flow associated with high recycling at the target plates ensures impurity control. In addition, the calculations show that the ion temperature in the SOL can be substantially larger than the electron temperature [Fig. 16(b)]. In present JET discharges, probe measurements indicate, that at low density, the electron temperature at the target plates is lower than the ion temperature determined from broadening of D_α emission and power balance considerations.[14]

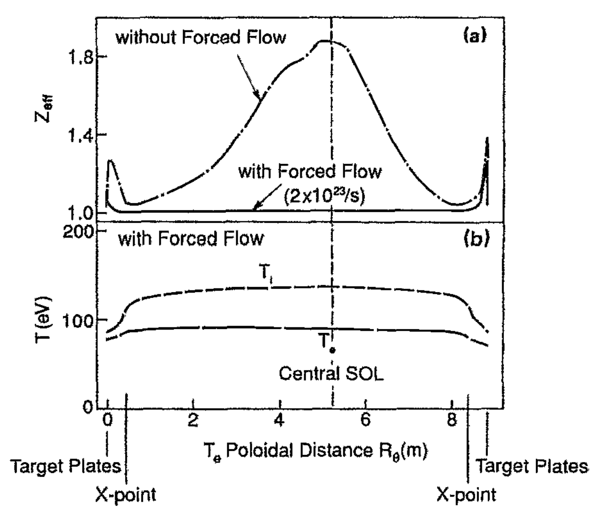


FIG. 16. Poloidal distribution in the SOL and DCP between target plates of (a) the effective ionic charge Z_{eff}, for cases with and without flow and (b) the electron and ion temperatures for the case with flow.

D. JET program in the New Phase

The New Phase of JET should demonstrate a concept of impurity control; determine the size and geometry needed to realize this concept in a Next Step tokamak; allow a choice of suitable plasma facing components; and demonstrate the operational domain for such a device.

A schedule for the JET program incorporating the New Phase is shown in Ref. 12. The earliest date to have a pumped divertor in JET is 1993. By the end of 1994, all information on particle transport, exhaust and fueling, first-wall requirements, and enhanced confinement regimes needed to construct a Next Step tokamak, should be available. This concept could be developed further toward lower temperature, higher density, e.g., cold radiating plasma or gas target. Such concepts would have to be proven further for the inclusion in a Next Step tokamak.

VI. TECHNOLOGY OF A DEMO

DEMO is a large and complex device. It would include a magnetic system (with superconducting coils), a divertor with high-power handling capability and low erosion, an exhaust system for impurities and helium "ash" products, a first-wall that is highly resilient to 14 MeV neutrons, a hot blanket to breed tritium, a plasma control system and a D–T fueling system. Auxiliary heating systems are required for start-up. DEMO would operate either semicontinuously with long pulses (2 h or so) or continuously with noninductive current drive. It must achieve high reliability, high duty cycle, and a high safety level. Activation and tritium inventory must be minimized. A full remote handling maintenance capability is required.

The construction of a DEMO must be preceded by an extensive test of the concepts and technologies required. Plasma facing components and first-wall materials with a high resilience to 14 MeV neutron radiation at a power flux of 2–3 MW m^{-2} must be developed and tested. Divertor target materials with high-power handling capability and low erosion (e.g., low-Z materials, beryllium, carbon and carbide fibers, and silicon carbide) must be developed and tested. An efficient, high-flow D–T fueling system with pellet injection needs to be demonstrated. The viability of present or emerging concepts for helium ash exhaust, pumping, and energy exhaust still needs to be proved. Tritium breeding blankets must operate at high temperatures but have not yet been tested. Furthermore, in the area of superconducting technologies, viable high-temperature superconducting materials are desirable for sufficient safety margin and these need thorough testing. A credible physics concept still needs to emerge for current drive, to sustain continuous plasma operation. However, this last requirement may prove to be unnecessary when semicontinuous operation (2 h or so) has been demonstrated, since this would be sufficient for a commercial reactor.

In view of the time needed to develop and incorporate emerging technologies required for a DEMO, a precisely defined or optimized engineering design cannot yet be proposed. A reasonably broad strategy would be to develop, partly in parallel, the main components of a fusion reactor. In addition, it is clear that more than one DEMO will be required (as in the early development of fission). All the physics and technological issues for a DEMO must be addressed, so that it can be constructed from proven elements. A well-defined *program* toward DEMO is essential.

VII. A PROGRAM TOWARD DEMO

The scientific feasibility of ignition under conditions required for a DEMO device must first be demonstrated: that is, high-power long pulse operation in fully ignited plasmas ($Q_{DT} = \infty$). Tritium breeding in hot blanket modules should also be tested. Advanced divertors and concept development aimed at improved efficiency must also be pursued.

Technology must still emerge on such items as pumping and helium extraction; the use of low-Z plasma facing materials (e.g., beryllium, silicon carbide, carbon fibers, and carbide fibers); the nature of breeding materials and coolant and structural materials for tritium breeding blankets. The resistance of highly sensitive materials (e.g., insulators, first-wall, etc.) to high-neutron fluences should be evaluated. Industrial development is also required for superconducting coils which can withstand neutrons and ac losses. It is expected that the remaining technology can be scaled up from existing knowledge.

Many of these issues are expected to mature on differing time scales. A *single* tokamak addressing all these issues would be close to a DEMO. Even with a large safety margin on each component, the risk of failure would be unacceptable. To incorporate all innovations that are likely to reach maturity throughout the lifetime of a single facility requires a design that lacks the precise definition offered by well-targeted objectives. There would also be an impact on the starting and construction times and on the consequential costs. A large degree of complexity would be introduced and this would place a practical limit on intended flexibility. We are not yet in a position to take such a large step. It is therefore not possible to address all issues on a *single* device.

An ITER *program* is the way to address the different Next Step issues. Several complementary facilities, each with separate, clearly defined objectives, would reduce scientific and technological risks; allow flexibility to accommodate new concepts; allow cross-checking of results; be more practical in managerial and industrial terms; and offer flexibility in location and time scheduling.

This program toward DEMO would minimize the risks and overall costs and would ensure an efficient use of world resources.

In such an optimized program, the three main issues of long burn ignition, concept optimization, and materials testing would be separated and addressed in different facilities which would be constructed as technologies mature. The engineering design for each facility can be defined precisely, thereby allowing a high degree of confidence that objectives would be met. A minimum ITER program should comprise the following:

P1—a thermonuclear furnace: the core of a first reactor;
P2—an advanced tokamak for concept optimization;
P3—a materials test facility.

In support of this ITER program, "National Programs" comparable in size will be needed.

The main details of such machines are set out below.

A. P1—A thermonuclear furnace: The core of a first reactor

The primary objectives of P1 would be the following:

to demonstrate sustained high-power operation of a fusion reactor core of 2–3 GW thermal power produced for up to 12 h per day for periods of several days at a time ($\sim$2000 h total);

to provide a testbed for the study and validation of tritium breeding blanket modules in full reactor conditions;

to achieve a cost/unit thermal output relevant to the establishment of fusion as a potential economic energy source (1ECU/thermal watt; 1 ECU $\approx$ 1.4\$);

to achieve a high level of safety and have minimum effect on the environment.

The design philosophy of P1 would be the following:

to make full use of the scientific and technical experience gained from JET and other tokamaks;

to minimize the need for developments by using established techniques;

to reduce complexity and increase reliability at reasonable cost;

to provide a high safety margin in achieving design specifications for the magnetic field and plasma current.

The objectives could be achieved in a tokamak with 30 MA, 4–5 T, major radius 8 m, minor radius 3 m, and elongation of 2. The main parameters of P1 are shown in Table I. Impurities would be controlled actively by high-density operation and a pumped divertor. The approach to ignition would utilize ICRF heating with H-mode confinement and in the monster sawtooth regime, while long pulse ignition (0.5–1 h) would be sustained with X-point L-mode confinement at high power and also with high-frequency low-amplitude sawteeth. With sustained ignition conditions, blanket modules would be tested under neutron fluxes of up to 2 MWm^{-2}. Conventional copper coil technology should be used. An early start in 1994 would be possible, once the results on impurity control become available from the New Phase of JET.

B. P2—An advanced tokamak for concept optimization

The primary objectives of P2 would be the following:

to develop advanced divertor concepts at high power;

to improve plasma performance by profile and shape control;

to test efficient continuous operation with current drive;

to demonstrate the viability of advanced superconducting technology in a large, high-power and high-field tokamak.

The basic geometry and concept of DEMO would be expected to evolve due to the emergence of "new" physics, which would be demonstrated on P2.

The design philosophy would be to test new concepts in a low activation environment at a reasonable cost. Flexibility should be built in to allow tests of new concepts throughout the lifetime of the device. This includes advanced divertor concepts (e.g., gas target, radiative target, cold plasma target, etc.) and advanced superconductors. The P2 objectives could be realized with a large tokamak operating at high power (typically 200 MW), 10–12 MA, 6–7 T, minor radius 1.5 m, major radius 5 m, and elongation of 2. The main parameters of P2 are shown in Table II. During the major part of the program, the tokamak would not operate in tritium and therefore would not ignite. Some technological developments are still required and therefore, construction could start somewhat after P1, perhaps in 1998.

C. P3—A test facility for materials

The objective of P3 would be to test materials under very large neutron fluences. This could be realized in a test bed operating continuously and providing high fluxes of neutrons (2–10 MWm^{-2}) over surfaces of 0.04–0.05 m^2. Its construction could start early in 1995, so that results should be available for the design of a DEMO in 2005.

D. Time schedule and costs

A time schedule for the design, construction, and operation of devices within an ITER program is shown in Table III. The overall cost for such a minimum program is estimated at 7 BioECU, not including operation costs, spread over ten years. This is made up of capital costs of P1 at 3.5 BioECU; P2 at 2 BioECU; and P3 at 1.5 BioECU.

In parallel, National Programs would be expected to continue on scientific and technological development in support of the International Program, which would cost a comparable sum (making a total of $\sim$ 14 BioECU over a ten year period). This should be compared with present world funding of fusion research of 1.2 BioECU/annum (i.e., $\sim$12 BioECU over a ten year period).

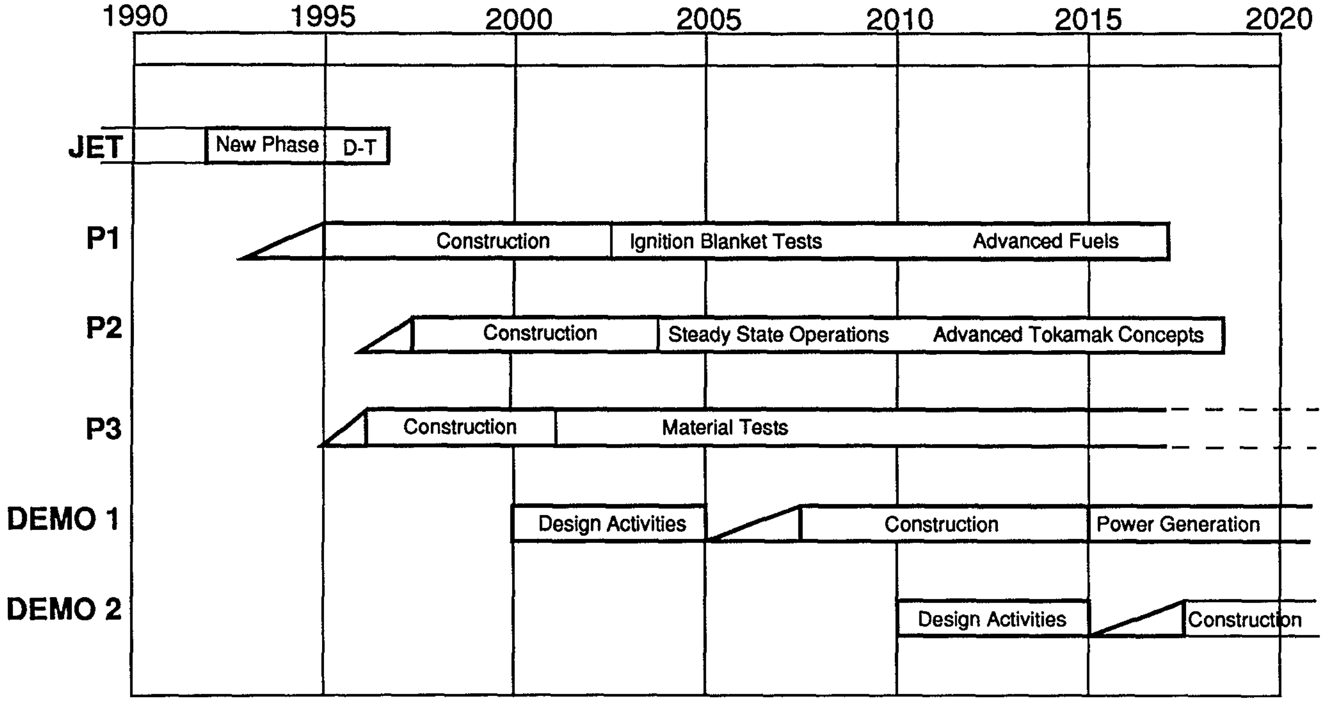
TABLE I. Parameters of P1.

Parameters	
Plasma major radius (R_0)	8 m
Plasma minor radius (horizontal)	3 m
Plasma minor radius (vertical)	6 m
Toroidal field strength at R_0	4.5 T
Plasma current	30 MA
Flat-top pulse length	1000–4000 sec
Volt seconds	425 V sec
Additional heating power (ICRF)	50 MW
Fusion power	0.5–4 GW

TABLE II. Parameters of P2.

Parameters	
Plasma major radius (R_0)	5 m
Plasma minor radius (horizontal)	1.5 m
Plasma minor radius (vertical)	3 m
Plasma aspect ratio	1.8–2
Toroidal field strength at R_0	6–7 T
Plasma current	12 MA
Flat-top pulse length	up to continuous
Current drive/additional heating power	200 MW

TABLE III. A time schedule for an ITER program.



VIII. CONCLUSIONS

The following points have been made.

(a) *Individually*, each of the plasma parameters n, T_i, and τ_E required for a fusion reactor have been achieved in JET; *in single discharges*, the fusion product of these parameters has reached near break-even conditions and is within a factor of 8 of that required in a fusion reactor.

(b) However, these good results were obtained only *transiently*, and were limited by impurity influxes due to local overheating of protection tiles.

(c) The quantitative understanding of fusion plasmas has improved with the development of a specific plasma model, which is in good quantitative agreement with JET data. The main predictions are also consistent with statistical scaling laws.

(d) With such a model, we begin to have a predictive capability to define the parameters and operating conditions of a DEMO, including impurity levels.

(e) Present experimental results and predictions of the model suggest the importance of controlling dilution in a reactor. The divertor concept must be developed further to provide sufficient impurity control;

(f) A New Phase is planned for JET, to demonstrate effective methods of impurity control in an axisymmetric pumped divertor configuration.

(g) Based on present progress, there is confidence that sufficient knowledge will exist to begin the construction of the "core" of a fusion reactor within the next three to four years.

(h) A single Next Step facility (ITER) is a high risk strategy in terms of physics, technology, and management, and it does not provide a sufficiently sound foundation for a demonstration reactor.

(i) A Next Step program comprising several facilities would make more effective use of world resources; is well within the capability of world research; would provide a wider and more comprehensive database; and could even be accomplished without a significant increase in existing world funding.

(j) With concerted effort and determined international collaboration, we have the resources to proceed with such a program, toward a DEMO starting operation in $\sim$2015.

ACKNOWLEDGMENTS

The authors are grateful to members of the JET Team, without whose dedication and hard work, the results quoted in this paper would not have been available.

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