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ABSTRACT

Regimes of tokamak operation more favourable to either inductive or non-inductive current drive are discussed. Steady-state operation, while desirable, appears to be limited to 'advanced' operation regimes. Non-inductive and inductive current drive are shown for 'conventional', high current, and 'advanced', low current, concepts. Thermal energy storage and augmentation systems suitable for pulsed systems are shown, and the issues of stress and thermal cycling are introduced. The advances in physics that may lead to attractive steady-state concepts, are shown also to lead to attractive pulsed concepts.

1. INTRODUCTION

In toroidal magnetic confinement devices, a poloidal magnetic field is required for confinement and stability. In tokamaks, this poloidal field is pre-dominantly generated by the plasma current. The desired plasma current I_P consists of two components: the bootstrap current I_B, driven as a result of plasma temperature and density gradients, and the 'externally-driven' current, $I_P - I_B$. This latter current can be driven either inductively, or non-inductively.

Non-inductive current drive (NICD) relies on high energy neutral beam injection [1] or on Landau damping of injected waves [2]. If the externally-driven current is high, the electrical power needed by the NICD system can lead to unacceptably high levels of power re-circulation [3]. Inductive current drive (ICD) uses the tokamaks central solenoid. ICD is highly efficient, so that power re-circulation is not a serious issue. On the other hand, ICD implies pulsed operation. This has two main consequences. First, in order to maintain full or partial electricity production during the periods without burn, some form of energy storage or augmentation is required. Second, fatigue induced by pulsing of mechanical or thermal loads will affect the life-time of certain components [4].

This choice of mode of operation will primarily be influenced by the level of plasma current required, and by the efficiency of non-inductive current drive systems. At plasma currents below, say, 15MA, i.e. in 'advanced' operation regimes, re-circulating power fractions for NICD are of the order of 10% to 15%. The capital cost per MW plant output power is then probably comparable for the NICD and ICD modes, taking into account that ICD introduces less constraint on the operation parameters and could allow higher fusion output. For higher plasma currents, i.e. for more 'conventional' concepts, NICD leads to higher cost due to the increasing re-circulating power fraction. The cost of the pulsed reactor is approximately independent of current, as the increase of scale-size that goes with the increase of current is compensated by an increase of fusion power.

It is of course advantageous if a reactor can operate in an 'advanced' regime with low plasma current, e.g. I_p < 15MA. Lower current will lead to smaller unit size and smaller total capital investment. Smaller unit size will allow easier integration of fusion plants in power generation economies where moderate size plants (< 1GW electrical) are the norm. It is also advantageous if such a reactor can economically work in steady-state. There is no doubt that pulsed operation poses additional technological problems in the first wall and blanket area.

However, the main stream of the present global confinement scaling data points towards devices operating at relatively high current. In particular, the state-of-the-art, and the need to adopt a low-risk approach, lead to ITER being defined as a high current machine [5,6]. At the full ITER current level of 25MA, current will be driven inductively.

The choice of mode of operation - inductive versus non-inductive - for a tokamak fusion reactor still lies some time in the future. It will be influenced by many developments both in plasma physics and in fusion technology. Therefore, it appears premature, at the present time, to attempt a detailed relative costing of pulsed and steady-state concepts.

JET is performing experimental work on all sides of this issue. Synergism between Lower Hybrid current drive and TTMP acceleration of fast electrons by ICRH fast wave heating was demonstrated [7]. A high fraction of bootstrap current was shown in discharges with a high poloidal beta [8]. And, in an AC operation experiment, breakdown of a second plasma within seconds of the termination of a first plasma was demonstrated [9].

In section 2 definitions of some of the terms used are given. In section 3, we show the demonstration of AC operation. In section 4, we discuss some of the issues of inductive operation for a tokamak concept that we will indicate as 'conventional', that is, high current, moderate super-conductor critical field and the tightest aspect ratio permitted by the inductive flux requirement. In section 5, we discuss the 'advanced' concept. We show some of the experimental achievements in 'advanced' regimes, and we point out areas where further research is necessary. We then show how inductive operation could be applied also to tokamaks operating in 'advanced' regimes, and we show a simple thermal augmentation system suitable for advanced tokamaks in pulsed operation. In section 6 conclusions are presented.

2. DEFINITIONS

In this section we define some of the terms used in the paper: re-circulating power and its consequences, the pulsed nature of inductive drive, and the AC and FC inductive drive cycles.

2.1 Re-circulating power.

Re-circulating power is the electrical output of a plant taken for the operation of the plant itself. As an example, for a large (6GW thermal) coal power station the re-circulating power is about 5%. In fusion reactors, the main users of electrical power are the cryogenic cooling system for the super-conducting coils, the pumping power for the coolant pumping, the power supplies for the poloidal field system, and, in steady-state machines, the NICD system.

It is generally believed that the total re-circulating power fraction should not exceed 20%. Re-circulating power leads to an increased investment cost, as for a given net plant electrical output a larger fusion power is required, hence a larger machine. It further leads to increased fuel cost, and to production of extra waste heat.

For ICD, pulsed plants the re-circulating power should be averaged over the total cycle. There is no NICD re-circulating power. On the other hand, there is in general more cryogenic power required because of eddy current dissipation during the current ramp-up and ramp-down phases (in a steady-state plant the current can be ramped very slowly). Secondly, there is power required to supply the magnetic energy to the primary coil, a fraction of which is dissipated in the power supplies. These extra contributions can amount to 10 to 100MW, depending on machine size.

2.2 Inductive drive cycles.

A typical inductive cycle is shown in figure 1. Burn is terminated as the plasma current is ramped down. Dwell-time, Burn-off time, and Burn-on time are defined in the figure. Duty-cycle is the ratio of the burn-on time to the total cycle time. During the burn-off time, no fusion power is generated. This leads to an 'energy deficit', equal to the product of fusion power and burn-off time.

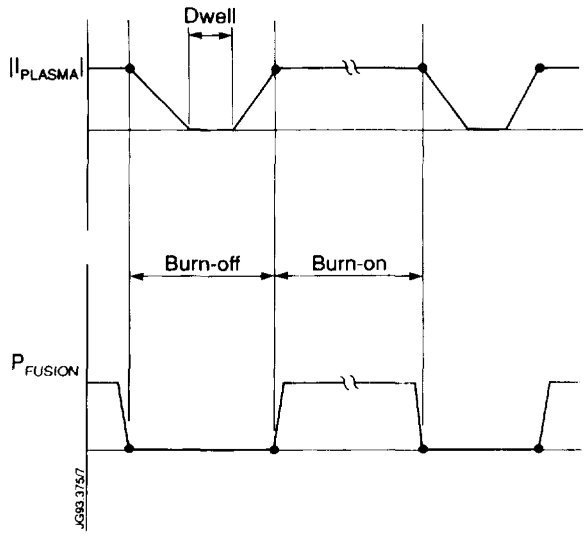

Figure 1. the inductive drive cycle.

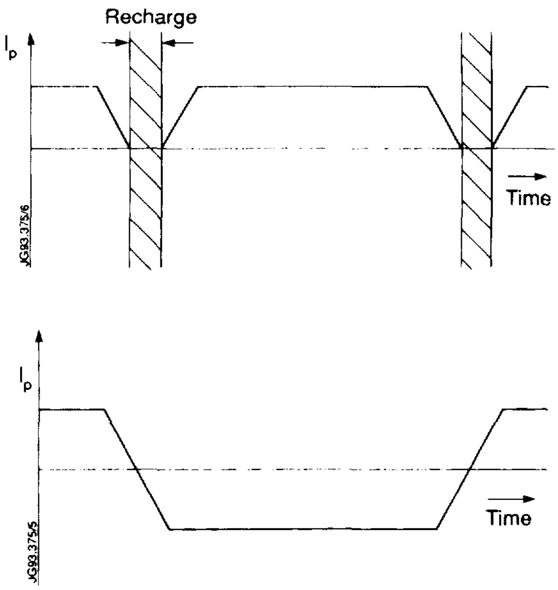

Figure 2. FC and AC cycles.

In figure 2 the two different inductive drive cycles are illustrated [10]. Forward Cycle operation is the standard operating mode of tokamaks, in which the transformer is re-charged before plasma breakdown. The dwell time is determined by the re-charge time, which can be of the order of minutes because of the large magnetic energy involved. In Alternating Cycle operation, plasma current and transformer current are symmetric around zero in successive cycles. There is no re-charge necessary, although flux can be gained by partial re-charge.

There are a number of trade-offs between FC and AC operation; AC is more expensive in flux, i.e. a bigger transformer will be required for a given flat top pulse length; AC operation allows shorter dwell times because no re-charge is required; The changing helicity of the magnetic field lines in AC puts a constraint on the divertor design; FC operation requires a significantly larger primary re-charge power supply.

We believe that these trade-offs are secondary to the debate. In the remainder of the paper we will refer to inductive operation without making a distinction between FC and AC.

3. DEMONSTRATION OF AC OPERATION.

Operation of a tokamak in the AC mode was first demonstrated on STOR1-M [11], at plasma current levels of the order of 4kA.

In JET, subsequently, AC operation as demonstrated at the level of 2MA [9], with limiter discharges. AC operation was found easy to accomplish, provided a finite dwell time was left between the successive discharges. This dwell was required for two reasons: first, the polarity of the vertical field must be changed, and this takes time. Second, immediately after termination of the first discharge the neutral pressure in the vacuum vessel is too low to allow a plasma breakdown. Hence, a pre-fill gas puff needs to be introduced, which again takes time. In the JET experiment, dwell times between 50ms and 6s were obtained.

In figure 3, traces are shown for the basic JET AC pulse. The density, under feedback, is the same in both cycles. ICRH power was applied to both cycles. For the same amount of power, the same temperature, effective ion charge and radiated power were obtained. These measurements indicate that there is no degradation of plasma purity in the second cycle.

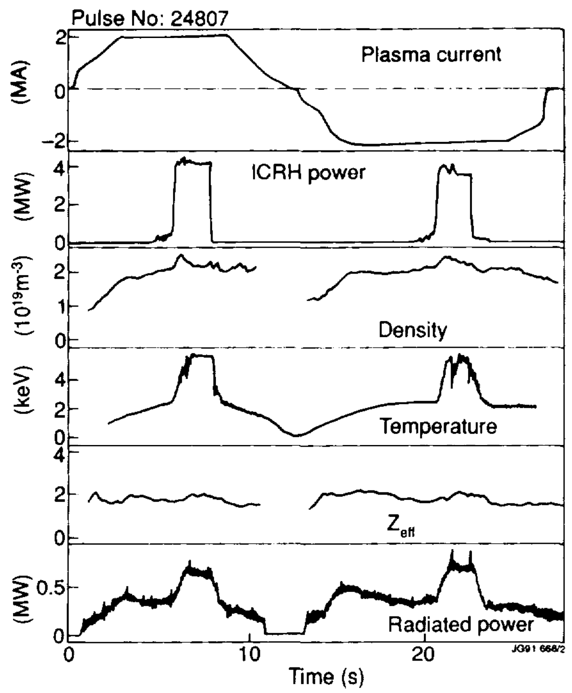

Figure 3. The basic AC pulse.

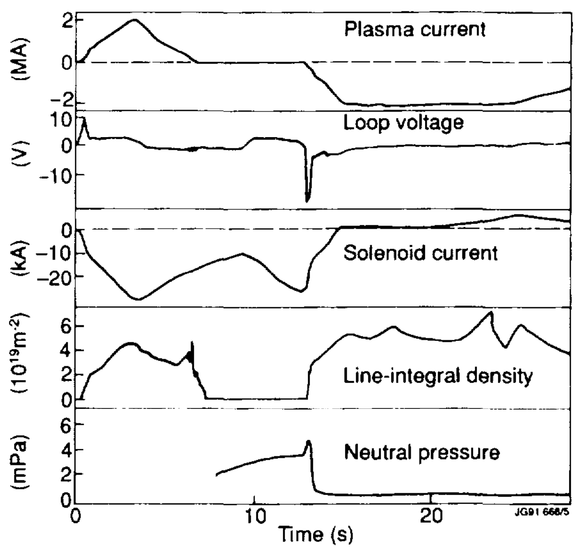

Figure 4. AC pulse with long dwell time.

In figure 4, traces are shown for AC operation with a 6s dwell time between the discharges. The time of the second breakdown was fixed for technical reasons, so the dwell time was obtained by shortening the first plasma. An interesting feature is the partial re-charge of the central solenoid just before the second breakdown. We observe that the neutral pressure, which is low just after termination of the first discharge, is rising during the dwell time. It is, in fact, comparable with the pre-fill gas-puff pressure which can be seen just before 13s. The implication is that the possible presence of impurity gases released from the walls do not prevent the second breakdown from occurring.

4. INDUCTIVE OPERATION FOR 'CONVENTIONAL' REACTOR CONCEPTS.

In this section we consider the operation of large, high current tokamak reactors, with concepts based on the present-day state-of-the -art. As a, perhaps extreme, example, we take a machine operating at a current of 25MA, with a poloidal cross-section close to that of ITER-EDA [6], but with the major radius increased in order to allow for sufficient flux capability (note that no credit is taken w.r.t. ITER-EDA for increasing the aspect ratio, i.e. the current is not lowered). We also assume a maximum field at the super-conductor of 12T. Reactor concepts along these lines have not been the subject of any extensive study. In the following sub-sections, we show typical parameters, consider thermal storage system concepts suitable for such devices, consider the issue of the cycling of the stresses in the TF system, and consider the cycling of components in first wall and blanket.

4.1 Typical parameters

The main parameters are summarised in table 1. We note that the large scale of the device is off-set by a large fusion power (7.5GW), at a 'moderate' average first wall neutron fluence. 'Moderate' is used here in the sense that similar values are quoted in reactor concept studies for smaller tokamaks. This machine necessarily operates at low safety factor $q_{cyl} = 2$. Consequently, poloidal beta and bootstrap current fraction are relatively small. The projected electrical power required to drive the externally-driven current non-inductively, with a current drive efficiency factor $\gamma = I_{CD}\ n\ R\ /\ P_{CD} = 0.5\ 10^{20}$, is about 750MW, or 30% of the electrical output. This effective rules out NICD for this machine.

Plasma current	25	MA
Superconductor field	12	T
Cylindrical safety factor	2	
ITER89P-L enhancement factor	2	
Toroidal β at Troyon limit	4	%
Major radius	10	m
Minor radius	2.7	m
Aspect ratio	3.7	
Fusion power	7.5	GW
Electrical power	2.5	GW
Neutron fluence	3.5	MWm^{-2}
Poloidal β	0.8	
Bootstrap current fraction	0.3	
Projected NICD electrical power	750	MW
ICD burn-on time	2 to 3	hours
ICD burn-off time	4 to 6	minutes
Duty cycle	95	%
Energy deficit	2000	GJ
No. of cycles in life	10^5	

Table 1. Typical values for conventional reactor concept.

This machine allows a central solenoid with a radius of 5m. With this, sufficient flux can be provided for a current flat top time of the order of 2 to 3 hours (AC and FC resp.), with a loop voltage of 80mV. The burn-off time is of the order of 5 minutes, based upon a scaling of experimentally achieved current ramp times in JET with the plasma skin time. The energy deficit is of the order of 2000GJ. With a 40 years life-time of the main structural machine components, the total number of cycles for these components will be about 10^5.

Before discussing cycling problems in detail, it is useful to note that 10^5 cycles is not far from the 60,000 D-T pulses foreseen in the 'Blanket Operating Conditions and Design Guidelines' of the ITER-CDA [5]. It is to be expected, that the design values for the ITER-EDA will be similar to these. Therefore, inevitably, many of the problems associated with cycling will already have to be addressed in the ITER process.

4.2 Concepts for energy storage.

It would be undesirable for the plant electrical output to drop to zero during the burn-off time. Even though in large power station networks (in particular in Europe and Japan) the sudden loss of a few GW production capability is a negligible perturbation and is instantly compensated by specific stand-by facilities, it would still appear desirable to maintain most or all (>80% [?]) of the production. This requires energy storage.

Several concepts for the thermal storage systems have been proposed. There are two approaches: first, there are passive systems, placed in series with the reactor (intermediary) coolant loop, which essentially increase the plant thermal capacity. Second, there are systems with flow control. Examples of the first type are given by Ehst for water and liquid metal systems [4], and by Horiike for a solid steel system[12]. These systems have the advantage that there is no flow control, but the disadvantages that the 'useful' temperature drop of the working fluid is relatively small and that the electrical output inevitably drops. These disadvantages lead to large systems.

Systems of the second type are described by Badger in the UWMAK study [13], and are presently under study by Vieider [14]. In these systems, a small fraction of the intermediary coolant flow is taken off the main flow and into a 'hot' reservoir during the burn-on phase. During the burn-off phase, the cooling of the blanket is valved-off. Heat to the steam boilers is then supplied from the 'hot' reservoir, with fluid being flowed into a 'cold' reservoir. The main advantages are that steam input temperature to the turbines can be kept constant, that the electrical output can be kept constant, and that the 'useful' temperature drop of the storage fluid is equivalent to the full temperature drop over the steam boiler.

Clearly, there are a number of concepts. The choice of system will also depend on the choice of primary coolant and blanket operating temperature range. An R&D

programme is required to identify the most attractive options and to develop systems beyond the conceptual stage. In this context, an attractive proposition may be the use of storage in organic liquids (e.g. ter-phenyl) which can be used at high temperatures under low pressures [15].

4.3 Toroidal field coil stress cycling.

The toroidal field coils in a tokamak, along the critical inner-leg region, are subject to three main forces: the tensile hoop force, which contains the magnetic pressure, the centering force, due to the toroidicity of the system, and the out-of-plane force, due to the component of the poloidal field perpendicular to the TF coil. Of these, the tensile and centering forces are constant in time, while the out-of-plane force varies with the plasma current.

There are two principal ways of constructing the inner-leg region of a TF set. The first solution is wedging the TF coils. This technique is used in many machines, and in the ITER-CDA and SSTR [16] concepts. When wedging, the centering force is turned into a toroidal compressive hoop stress in the wedge. This stress is significantly larger than the centering pressure itself, and is of the order of the tensile stress. In wedged solutions the stress due to the out-of-plane force is essentially eliminated due to the torsion stiffness of the wedged vault. Hence, both relevant stresses are constant in time. Modulating the plasma current presents only a small stress modulation.

The second solution is to buck the TF set onto the primary coil. This technique is used in JET, and planned for the ITER-EDA. The centering force now leads to only a small radially compressive stress. On the other hand, stresses due to the out-of-plane forces become significant. In order to achieve a reduction of the out-of-plane stresses to a level significantly smaller then the dominant tensile stress component, some advanced method of mechanical structure is required. In ITER-EDA a system of shear keys is foreseen to obtain the required torsion strength into the region. Alternatively, one could link the coils in bucked pairs, or bucked triplets. Bucking can lead to significantly thinner TF coils.

For pulsed machines a bucked solution is preferable. The principal reason lies in the pre-compression of the primary coil that is provided by the centering pressure. This allows the construction of a primary coil with a smaller fraction of structural material.

Both in the case of wedged coils, and in the case of bucked coils with an advanced construction, the modulating components of the stress is small. In that case, the fatigue considerations regarding the TF coil set are dominated by the number of shutdown interruptions, when the TF field is switched off. The effect of plasma current cycling will be negligible.

4.4 Thermal cycling.

The bulk of the first wall / blanket / vacuum vessel structure has a thermal time constant much longer than the burn-off time. In addition, cooling to these structures may be valved-off during the burn-off time. Hence, for large components within these structures, no significant temperature excursions will occur during burn-off. Consequently, fatigue considerations will be dominated by shut down interruptions, and not by the cycling.

On the other hand, many of the smaller components in divertor, first wall and blanket will have a thermal time constant significantly shorter than the burn-off time. Examples of such components are divertor high heat flux components, first wall armour, and breeder or neutron-multiplier pellets. In such components, temperature gradients will change and as a result, thermal stresses will be induced. Fatigue considerations for the pulsed reactor are significantly more severe than for the steady-state reactor for such components.

Having said this, on the other hand, the maximum permissible stress for most materials is approximately halved when the number of cycles is increased from 10^3 to 10^6, these numbers being reasonable design numbers for a steady-state and pulsed machine respectively. It would thus appear that it must be possible to find engineering solutions

It is clear that no solutions can be provided to these problems at the present time. In particular, for both the pulsed and steady-state concepts there is a severe concern about the uncertainty of the behaviour of materials under a 14MeV neutron flux. This is reflected in an order of magnitude uncertainty of the life-time of the first wall. For many fusion relevant materials, neutron irradiation tests and helium embrittlement test under a fusion-like neutron flux have not been performed, let alone tests of the fatigue cycling.

5. ADVANCED CONCEPTS

The essential objective of advanced concept research, in a wide sense, is to establish a tokamak operation regime which would allow fusion reactors of a smaller scale, with a smaller total capital investment cost. Preferably, also the capital cost per unit output power should be reduced with respect to that achievable with 'conventional' concepts.

Currently, a narrower interpretation of 'advanced' is common. An 'advanced' tokamak, in this sense, is a steady state machine, operating at low plasma current, with a large fraction of bootstrap current. In the remainder of the paper we will use the word 'advanced' in the narrower sense, and use the indication 'low current' for the wider sense.

There are at present two relatively complete advanced tokamak concept studies [17]. The ARIES-1 study [18] is the most complete device, with emphasis on safety and environmental issues. The SSTR concept [16] is considered to have the best short term feasibility.

In the following sections, we discuss the basics of the advanced concepts, show the progress in demonstration of the advanced operation regime, and identify some areas for further research. We then investigate what the consequences of 'low current' or 'advanced' regimes would be for pulsed operation, show a simple energy augmentation system, and show a proposal for a ten-hour pulsed machine.

5.1 Basics of the advanced concepts

The basic principles of the advanced concepts are as follows: under ITER89P-L scaling [19] (and other scalings), the fusion parameter $nT\tau$ scales favourably with aspect ratio. Consequently, the plasma current can be reduced by increasing R/a. In addition, to reduce the plasma current further, one assumes relatively high confinement enhancement factors, $H > 2.5$ over ITER89P-L. These enhancement factors are to be obtained via tailoring of the current profile. Advanced machines operate at high safety factor $q_{cyl} > 3$. Both the increase in R/a, and the increase in q, lead to a high poloidal β_{pol} 2, and consequently to a high fraction of bootstrap current $f_B = I_B / I_P$ 0.65 β_{pol} /](R/a). Consequently, the externally-driven current $I_P - I_B$ is small, and can be driven non-inductively with a moderate NICD power.

Steady-state operation can be achieved, with a small re-circulating power, of order 10%.

It is important to consider one basic disadvantage of this scheme. Both the increase in R/a, but in particular the increase in q, lead to a reduction of the toroidal β_{tor}. At the conventional Troyon limit, we have $\beta_{pol,limit}$ g q (R/a) and $\beta_{tor,limit}$ g q^{-1} $(R/a)^{-1}$, with g the Troyon coefficient. The fusion power $P_{fus,limit}$ is therefore lower then the level that could be achieved with a given toroidal field set. Because the TF set is usually the most expensive piece of hardware of the device, there is an economic penalty. In a sense, reduced fusion power is the price of steady-state operation. In advanced tokamak concepts the reduction of fusion power is compensated for by operating at a relatively high Troyon factor, and by operating with high field superconductors, with B_{crit} = 16 and 21T resp. for the SSTR and Aries designs.

5.2 Experimental progress in advanced regimes.

Significant progress has been achieved in the demonstration of advanced regimes over the recent years. The existence of a large fraction of bootstrap current in a discharge with high poloidal β_{pol} was first shown by TFTR [20]. In JT-60 [21], the highest fraction of bootstrap current (80%) has been obtained. Both TFTR and JT60 have achieved this with NBI. In JET, a high bootstrap current fraction (75%) was demonstrated with ICRH heating alone [8], showing that, unlike in straightforward neo-classical theory, bootstrap current exists also without strong radial particle flows.

High enhancement factors H for the energy confinement over the ITER89P-L scaling have been observed in the high safety factor, high β_{pol} regime. In figure 5 the achieved enhancement factor versus the parameter $\varepsilon\beta_{pol}$, for discharges in JET and JT60-U is shown . H factors of up to 3.6 have been obtained, where, for reference, the H factor of a standard H mode is 2.

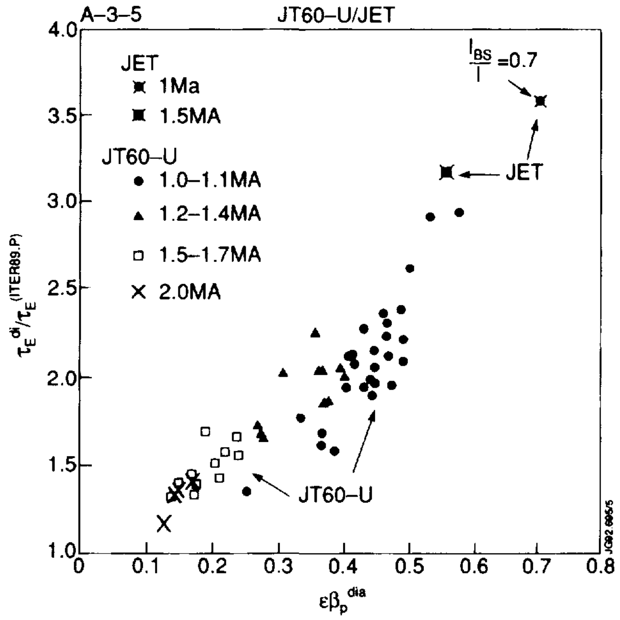

Figure 5. Confinement enhancement factors in the advanced regime.

Although the results obtained on the advanced regime are encouraging, the regime is still in its infancy. The confinement enhancement shown appears only for times of the order 1s. In particular, it has not been demonstrated on a time scale approaching the resistive diffusion time. This implies that the bootstrap dominated current profile is not yet established, except near the plasma edge. Hence, both the stability of the bootstrap dominated current profile, and the survival of the high H factor on the time-scale of the current evolution, need to be demonstrated.

In addition, compatibility of the regime with divertor operation needs to be shown. In particular, the issue of helium accumulation, which is a well known problem for the standard H mode, is crucial.

5.3 Typical parameters for an advanced tokamak concept.

In table 2 typical parameters for an advanced tokamak are shown. These were taken from the SSTR study. We see the operation at a relatively high safety factor, a high aspect ratio, a high poloidal β and a relatively low toroidal β. It is also interesting to note that the superconductor critical field of 16T is crucial for this machine. Although a lower field could be tolerated from the MHD stability point of view (q could be lowered), the fusion power would be significantly reduced.

The NICD power, evaluated at an efficiency $\gamma = 0.5\ 10^{20}$, is 60MW, leading to a re-circulating power of about 120MW or 11% for the CD system.

Plasma current	12	MA
Superconductor field	16	T
Cylindrical safety factor	3	
ITER89P-L enhancement factor	2.6	
Toroidal β at Troyon limit	2.7	%
Major radius	7	m
Minor radius	1.7	m
Aspect ratio	4.1	
Fusion power	3.0	GW
Electrical power	1.3	GW
Neutron fluence, average	3.0	MWm^{-2}
Poloidal β	2.0	
Bootstrap current fraction	0.75	
Projected NICD electrical power	120	MW

Table 2. operating parameters for an advanced tokamak reactor.

5.4 Inductive operation of a low current concept.

In this section we consider the option of driving the amount of externally-driven current (3MA) for the advanced concept machine (previous section) by ICD. In doing so, we implicitly assume that the current density profile, generated by the inductive drive and the bootstrap current, is stable. This is, of course, by no means guaranteed. The present exercise is intended as an illustration; it is not intended to argue that a machine like SSTR should be operated in a pulsed mode without making further changes to the operating scenario. In fact, ICD would allow in a very similar machine an operation at somewhat higher plasma current, and lower bootstrap current fraction. Such operation would be to a lesser extent bootstrap dominated, could be expected to be MHD stable, and would produce a significantly higher fusion power.

In a machine with parameters as in the previous section, with an advanced construction of the toroidal field set leading to relatively thin TF coils (0.9m), it would be possible to include a primary coil with 3m radius. For this coil a critical field of 16T is assumed. The typical parameters that could be obtained with ICD are shown in table 3, where the first and second number refer to FC and AC operation respectively.

ICD burn-on time	4 to 5	hours
ICD burn-off time	1 to 2	minutes
Duty cycle	99	%
Energy deficit	400	GJ
No. of cycles in life	10^5	

Table 3. Inductive operation of a low current concept.

We observe from the table that the premises that lead to attractive steady-state operation, also lead to attractive pulsed operation; the low current reduces the inductive flux requirement, he large aspect ratio allows for a large primary coil, the high field super-conductor leads to a large flux capability, the high bootstrap current fraction reduces the required loop voltage (about 25mV), and finally, the combination of high q operation of a small minor radius plasma leads to a short skin time and avoidance of the major external kink resonances, hence to a fast plasma current ramp time.

The energy deficit for this machine is relatively small, 400GJ as compared to 2000GJ for the conventional concept. The duty cycle is very high.

5.5 Energy augmentation for a pulsed, low current concept.

An energy augmentation system, suitable for pulsed operation of a low current concept, was proposed in a study by the AEA Technology [22]. A sketch is shown in figure 6. In this system the standard steam boiler is replaced by a boiler of higher thermal capacity. This boiler is equipped with an additional gas burner, which is switched on during the burn-off time of the machine. The bulk of the energy deficit is supplied by cooling of the blanket, boiler and other system elements. The temperature of the blanket is reduced by about 100K over a burn-off time of typically 100s. The function of the gas burner is to super-heat the steam, so as to maintain constant steam temperature to the turbines. With such a system, it was proposed that over a burn-off time of about 100s, the drop of electrical output power would be less than 20%. The system was costed at about 60M$ capital investment, to include the enhanced boiler.

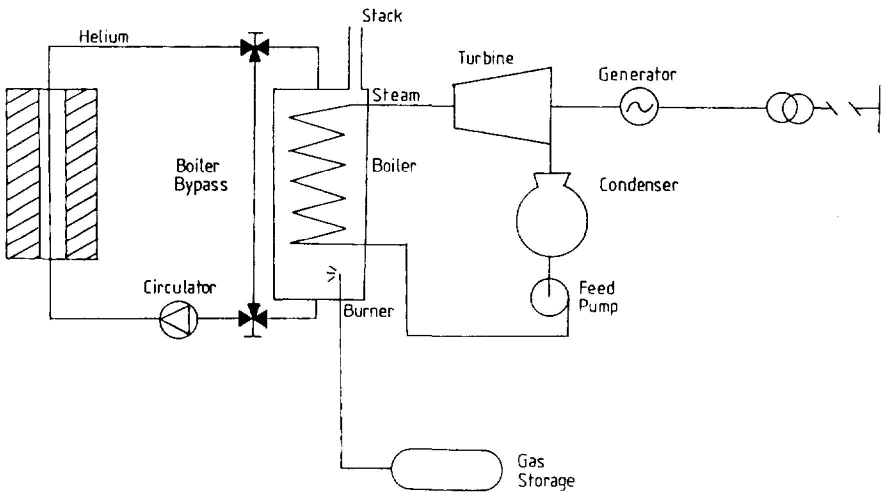

Figure 6. Energy augmentation system (source report AEA FUS 205).

5.6 An ICD advanced reactor with an ultra-long burn-on period.

For the first few generations, it can be assumed that fusion power plants will be a minority, in networks surrounded by fission and thermal plants, and supplemented by pumped storage facilities. Therefore, the capability to load-match the electricity demand is not an urgent consideration at present. If, on the other hand, one considers networks with a substantial fraction of fusion stations, load matching becomes desirable. In a network, there are two principal ways of load-matching. One can vary the power level of all or most plants, or switch a number of plants on and off.

For NICD plants, load-matching may put an additional constraint on the economics of the operation. It is unlikely that NICD plants can be operated economically at a power level significantly below their nominal value. The plasma current still needs to be driven, and the density can not be reduced arbitrarily. Re-circulating power may then be a large fraction.

In this context, it is interesting to note proposals for an ultra long pulse ICD reactor [23]. On the basis of the observation that the demand for electricity varies by a factor 2 to 3 between day and night, an ICD machine with a pulse length long enough to cover the day-time demand is considered. Based on the 'conventional' concepts, the major radius of a 10 hour tokamak would be of order 15m, which is unattractively large. On the other hand, under the assumptions of the 'advanced' concept, a major radius of about 8 to 10m would suffice. The increase in radius would be compensated by an increase in fusion power. Such a machine would not have a thermal storage. Thermal cycling problems would be much reduced, because the machine would be switched on and off relatively slowly.

6. CONCLUSIONS

There is a strong incentive to develop tokamak operating regimes at lower plasma current. This would open the way to economic fusion reactor concepts of a smaller scale size. Although the smaller scale size may or may not reduce the ultimate capital cost per installed MW, smaller machines will be better suited to certain power production systems. JET can make a strong, and short term, contribution to this work, by exploring 'advanced' operation in a wide sense: a search for enhanced confinement and stability regimes accessible with profile modification, and for improvements in NICD efficiency.

If low current operating regimes are comprehensively demonstrated, this can lead to reactors that can be operated in steady-state, using NICD. However, the advances that lead to steady-state operation becoming an attractive option, also make pulsed, ICD operation of these machines attractive. In particular, without the constraint of the necessity of a high bootstrap current fraction, such reactors can be operated at lower safety factor and at high β_{tor}, thus maximising the fusion power.

If 'advanced' operating scenarios are not, or not fully, demonstrated, one may have to accept that at least the first generations of tokamak fusion reactors will operate at high plasma current. Pulsed operation may then be mandatory. In view of the many issues of R&D affecting the economics of pulsed and steady-state reactors, it is pre-mature to make detailed analysis of the relative cost of pulsed and steady-state systems.

Both the pulsed and the steady-state concepts share the need for a comprehensive and experimental qualification of a variety of fusion relevant materials and joining techniques.

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