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Background Notes Relating to Poster Presentation

on

Recent Developments in Engineering and Technology Concepts

for

Prospective Tokamak Fusion Reactors

(with special reference to the NET project, the Next European Torus)

by

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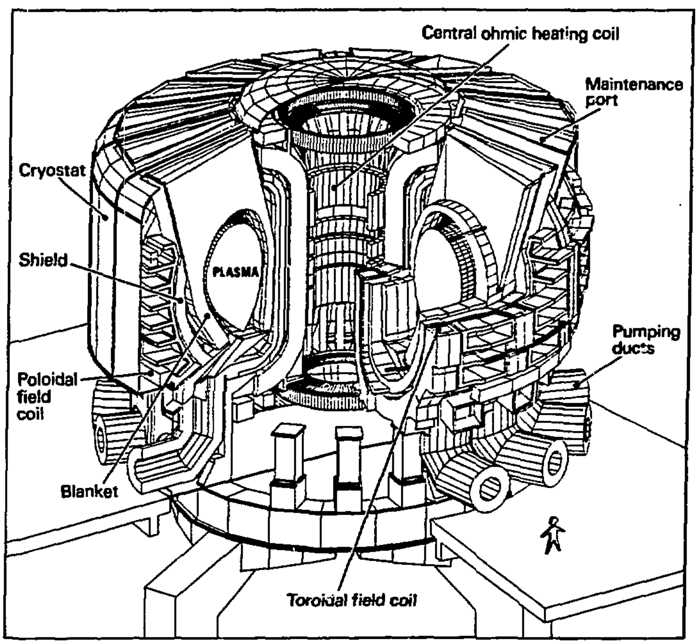

Schematic view of NET-Single Null concept,
now superseded by NET-Double Null, the main subject of this Note

ABSTRACT

The tokamak has become the most developed magnetic fusion system and it appears likely that break-even and possibly ignition will first be demonstrated in existing machines of this type. Yet larger tokamaks could also demonstrate the essential tecnologies for the production of useful power. World-wide, well over a hundred tritium-breeder/heat-removal blanket concepts have been devised and preliminary engineering design studies undertaken, but the effort deployed on breeding and power recovery systems has been very small compared with that assigned to plasma research and development. The European Communities' NET (Next European Torus) project may offer an opportunity to redress this imbalance. The NET pre-design stage now in progress for some three years has selected many of the best features of plasma and nuclear design from the worlds' total efforts in these fields, and the NET concept is described in this paper as exemplifying where magnetic fusion power reactor technology stands today. It is concluded that although there are numerous more advanced types of magnetic-confinement fusion reactor at early stages of their physics development, the tokamak offers the best opportunity for the early demonstration of fusion power. It is considered that appropriate Australian research institutions should seek to make contributions to this program through international co-operation agreements.

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Recent Developments in Engineering and Technology Concepts

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Prospective Tokamak Fusion Reactors

(with special reference to the NET project)

1. INTRODUCTION: Of the numerous conceivable routes to the practical development of nuclear fusion power the Tokamak concept and its allies appear to have become the favoured front runners, with the Tokamak itself in the lead. Funding for the magnetic mirror program has now been severely cut, and the inertial concepts seem to have a very long way to go before they offer any prospect of a feasible power plant: it would appear probable that the main incentives for continued work in the inertial field are military developments and fundamental physics. Of the Tokamak allies, the Reversed Field Pinch (RFP) concept appears to be the most favoured, but although several quite extensive engineering design studies have been completed, the physics developments are as yet a long way behind those of the Tokamak, and the engineering studies must be regarded as somewhat speculative at the present time. However, it is to be noted that one major thrust of the RFP engineering studies has been to highlight the very large materials investment inherent in the various Tokamak reactor concepts, which, on account of their relatively low power densities, appear to demand some 5 to 10 tonnes of materials per MW(th), compared with a target of around one tonne/MW(th) for the Compact Reversed Field Pinch Reactor scheme (Los Alamos).

Although outline sketches have been devised of all sorts of speculative power reactor schemes based on such concepts as Compact Toruses, Z-pinch, Dense Plasma Focus and Muon Catalysed Fusion, none of these can be regarded as feasible projects at the present time and will require in all cases much further physics investigation before engineering concepts can be far developed.

Now that the World's four large experimental Tokamaks are producing, or are about to produce, a great wealth of data on the essential plasma physics of near-reactor-size machines, with a considerable amount of practical experience on plasma engineering, substantial effort is being applied to the conceptual design of the next step. Considerations of likely rather low levels of funding have pushed the United States laboratories towards proposals for one more increment in the investigation of plasma physics, with conceptual schemes for 'burning plasma' tokamaks of various designs being the present outcome. However, in Europe the aim is to design and build a machine in which all the aspects of fusion reactor technology necessary to convert the fusion energy output into useful power may also be fully tested. This machine, the Next European Torus, NET, would be a direct precursor to a power station demonstration ('Demo') reactor.

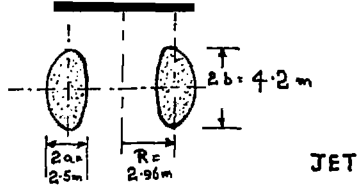
1.1 NEXT EUROPEAN TORUS: NET: The present 'reference design' for NET is the outcome of three years of work in European laboratories, lead by a group at Garching responsible for the overall concept. It has been derived quite directly from the experience gained in the international INTOR study, and clearly incorporates many features drawn from the numerous other design studies on conceptual fusion power reactors which have been developed over the past ten years. The Project Group has available, of course, the immediate experience being built up within the European Community from the JET project.

This paper has therefore concentrated on presenting the essential features of the latest NET preliminary reference design, the NET-DN (Double Null) as described in the two dozen or so papers on this development given at the SOFT Conference in September 1986, with additional comments where appropriate drawn from other reactor studies published in this or other conferences or papers. Most of the accompanying sketches have been taken directly, or adapted, from the SOFT papers.

The planning objective for NET is to complete the present phase of 'pre-design'in 1989, the detailed design to occupy the period from 1990 to 1995. At thas stage there would have to be major decisions on funding and siting: the funding would involve expenditures of several billions of ECU's (dollars). If approved by this date, it would be expected to take some five years to build, and hopefully be on-line by the year 2000. The testing program is envisaged as occupying some 10 years, following which a Demo reactor would be built, possibly to be on line by 2025.

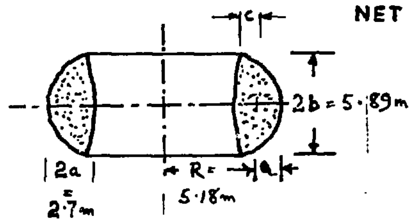
1.2 GENERAL DESIGN PHILOSOPHY: The outstanding general principles of the NET design approach are: to evolve a controlled-ignition pure-fusion deuterium-tritium (DT) burning reactor system which, from the outset, is based upon full remote handling procedures for every stage of its operation, maintenance and modification; and one which is able to test out several variants of both plasma engineering and neutron blanket technology. It would thus be much more flexible in application than a DEMO or a commercial power reactor, and, at 600 MW(thermal), would have a much lower, more easily managed, power output. It should therefore be able to test out all of the most promising ideas for power reactor technology being developed during the next two decades.

1.3 ELECTRIC POWER GENERATION: It is not yet clear to what extent, if any, the output heat of NET would be converted into electric energy. Since the major aim of NET is to test out as many breeding-blanket concepts as possible it may be undesirable to add the extra complication of steam-raising and actual power generation. However, there are many special problems such as titium control which need to be investigated, and one preliminary study has been completed for an electricity-generating blanket module.



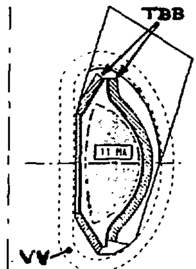
2. ESSENTIAL FEATURES of the NET-DN CONCEPT

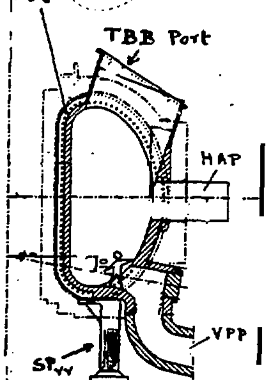
2.1 THE REFERENCE PLASMA: The objective is to produce a double-null divertor-controlled plasma body of some 390 cubic metres volume, and to develop a fusion power of some 600 MW(th) over a burn period of 200 seconds or more, the average power density thus being about 600/390 = 1.54 MW/cu.metre. The plasma cross-section is elongated, b/a = 2.18, and triangulated, c/a = 0.65, with aspect ratio R/a = 3.8. The particle densities are: electrons 1.56E20 and ions (D plus T) 1.20E20 per cubic metre, the aim being the strongly-peaked Asdex H-mode particle distribution, corresponding to divertor operation.

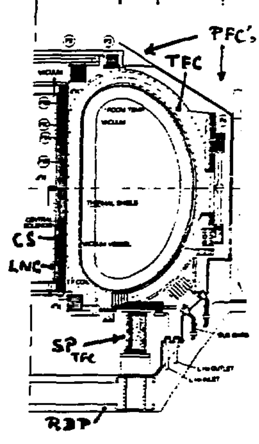


2.2 BASIC REACTOR ARCHITECTURE: The machine has as its principle architectural concept two concentric toroidal structures which between them carry the entire weight of the reactor. Based upon the Reactor Base Plate (RBP) each toroidal structure is carried independantly on 16 Support Posts (SP), the two rings of 16 posts each being interspaced with each other on the same pitch circle. Each SP consists of a laminar arrangement of steel plates, flexible in the radial direction but rigid circumferentially. The inner torus of the pair is a combined Vacuum Vessel and nuclear radiation shield (VV), which also supports the weight of the removable Tritium Breeding Blanket (TBB) sectors; the outer torus comprises the structurally-linked ring of Toroidal Field Coils (TFC's), and fits closely around the outer surface of the VV with a thermal insulation gap between the two: the outer torus also carries the weight of the Poloidal Field Coils (PFC's) together with that of the cryostat vessel containing the liquid helium for cooling both sets of superconducting coils.

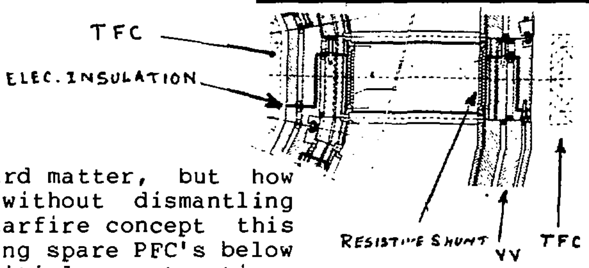
2.2.1 VACUUM VESSEL (VV): This is the most permanent structure of the reactor and is not expected to be replaced during the machine's working lifetime. However, some degree of remote-operation repair will be feasible eg. of seal welds etc. It does not require to be disassembled in order to service or remove breeder blanket components, and is shielded from the direct effects of the plasma by the blanket units. It also provides the (water-cooled) radiation shielding protecting the magnetic field coils. The reactor specification demands that "all coils must be remotely handleable so that they can be removed from the machine and replaced in case of failure. For the poloidal field coils it is necessary that the coils should be removable without the need to dismantle the torus". To remove a TFC it would be necessary to split open the VV and remove a segment of the VV together with its associated TFC. Removing and replacing the upper PFC's



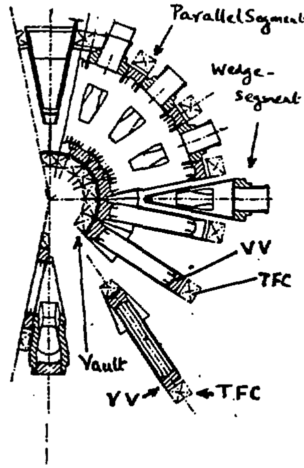




HAP = Horizontal Access Port: VPP = Vacuum Pumping Port

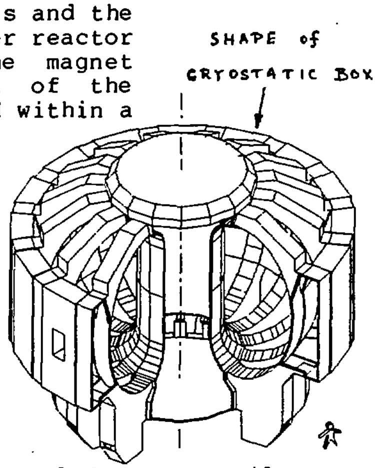


would be a relatively straightforward matter, but how the lower PFC's could be removed without dismantling the VV is not yet clear. In the Starfire concept this problem was dealt with by installing spare PFC's below the torus during the reactor's initial construction. Any of these coil handling operations by remote manipulation would evidently be most complex and demanding: it may be noted that one TFC would weigh 200 tonnes.

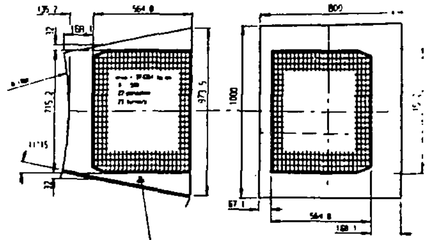


2.2.2 VV SEGMENTATION: The VV is constructed in 16 major segments, each of which comprises two sub-units, the one a cylinder with parallel end-faces, the other wedge-shaped with a 360/16=22.5 degree included angle. The 32 segments are connected to each other by captive bolts within the wall thickness, and, on the plasma side of the vessel, by seal-welded flexible bellows. Each parallel segment is split into two sub-rings electrically insulated from each other by an inorganic insulator, but toroidal electrical continuity is provided to a controlled degree by means of resistive elements which shunt the electrical break, giving a total troidal-direction resistivity of 0.2 milliohms. Consideration is being given to providing the insulating separator in the form of a varistor material to reduce the overload current in the resistive shunts during a plasma disruption.

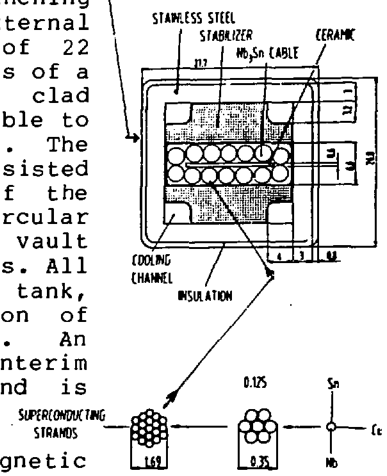
2.3 MAGNET COILS and CRYOSTAT: Both the TFC's and the PFC's are superconducting: in a low beta power reactor this is essential if the power consumed by the magnet system is to be an acceptably small fraction of the gross power output. Both systems are contained within a (very intricate) common cryostatic box.



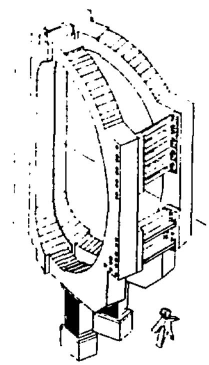
The working life of the coil system is hopefully that of the plant as a whole, but, as mentioned above, provision is to be made to remove and replace any of the coils if necessary. The coils are made from A15 alloy (Nb3Sn, Nb3Al) and are forced convection cooled. This material at 4.5 deg.K can withstand a field within the coil conductors of 11.4 T. To provide additional field strength for the plasma initiation phase of the cycle an auxiliary liquid-nitrogen-cooled normal-conducting coil (LNC) is provided within the bore of the Central Solenoid (CS), and two sets of water-cooled copper coils are situated close to the plasma, just behind the blanket, to provide vertical-stability active-control (VPC's). The radiation durability of any of the coil types, normal or superconducting, is determined by the insulation rather than the conductors, and to this end reliance may be placed on glass/epoxy-resin insulation for the superconducting and LN-coold coils, but ceramic insulation is essential for the VPC coils because they are situated in a very strong radiation field.



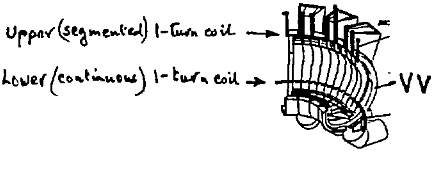
2.3.1 TOROIDAL FIELD COILS (TFC's): The TFC's are required to provide some 5 Tesla on axis. They are somewhat elongated compared to the theoretical bending-moment-free D-shape, and so require special strengthening against in-plane forces. This is provided by a strong steel outer frame or jacket, by sustantial internal strengthening within the windings, and is aided by the external inter-coil support structure. Each TFC consists of 22 flat pancake sub-coils, each formed from 23 turns of a superconductor/copper-stabiliser stainless-steel clad composite conductor, expected eventually to be able to be manufactured in two or three kilometre lengths. The radial-inwards reaction force of the TFC's is resisted by arranging each radially-tapered inner leg of the coil to nest with its neighbours to form a circular vault. Overturning forces are resisted by this vault and by inter-coil struts at the radial extremeties. All the inter-coil supports are within the cryostat tank, which simplifies the problems of thermal insulation of the coils from the environmental temperature. An experimental 'prototype' TFC, using an interim conductor development, has been manufactured and is soon to be tested.



2.3.2 POLOIDAL FIELD COILS (PFC's): The magnetic capabilities of the PFC system determine the control of plasma shape, and the attainable Ohmic Heating current and burn duration. They are formed from the same superconducting material as used for the TFC's, and likewise have forced convection cooling. A protoype PFC is being developed and it is intended to test this on Torre Supra in 1990. The flexibility and capability of the PFC system is such that a 15 MA double-null plasma can be produced (cf. the 11 MA reference plasma).



2.3.3 VERTICAL POSITION CONTROL COILS (VPC): Vertically-elongated plasmas are subject to a particular type of instability which is able to be controlled by (a) the provision of passive conductor circuits specially provided in the blanket units (in the first wall , and in the breeding blanket structure), and also by the inner wall of the VV; and (b) by active control coils situated close to the plasma. This active control cannot be provided by the PFC system for the double reason that they are effectively shielded from the plasma, as far as concerns short-term variations of field, by the VV; and in any case the power required would be prohibitive. Accordingly, the active control is provided by two sets of normal-conducting copper/aluminium coils, water-cooled and ceramic insulated, which are positioned close behind the blanket. The lower coil is continuous, but the upper coil has to be segmented in order to facilitate the removal of the blanket units. It is installed within the blanket unit key plug. The VV serves the important purpose of shielding the rapidly-varying VPC fields from the PFC's thus reducing induced-current losses.

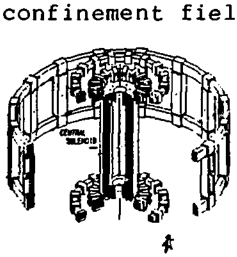


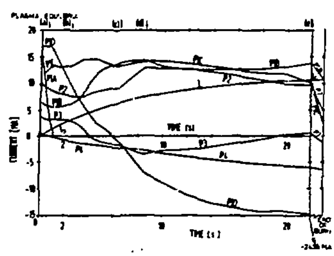
2.3.4 CRYOGENIC SYSTEM: The coil temperature is maintained by forced convection liquid helium cooling at about 4.5 deg.K. The average total thermal load is some 25kW, made up of: TFC's 16kW (8kW eddy currents, 8kW nuclear heating), PFC 4 kW (mostly eddy currents), structure-plus-thermal losses 5kW. Total helium flow rate is 2 kg/s, inlet pressure 6 bar.

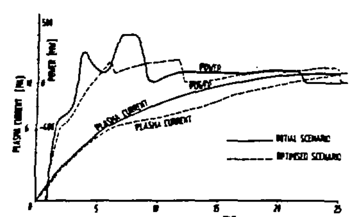
2.4 POWER SUPPLIES:

2.4.1 TFC's: Magnetic energy stored in the toroidal field is some 25GJ. This energy is fed into the coils over a period of some hours, and is expected to remain in the steady-state for several months at a time during normal operation. In event of a serious fault the stored energy must be transferred to, and dissipated in, an external resistor before overheating occurs in the coils. To maintain symmetry of mechanical reaction forces during this transfer alternate TFC's are connected in series to form two coil sets.

2.4.2 PFC's: The power supplies to the PFC's forms an unavoidably complex system involving large flux changes, calling for up to 16kV at coil terminals and 40 kA peak conductor currents, the stored energy in the PF system being around 5 MJ. First, during the first 22 seconds of the total pulse, between them these coils have to initiate and position the plasma, and provide the OH contribution to raising the plasma temperature to the ignition point. This start-up phase is followed by the ignited burn period of several hundred seconds (between 200 and 1000), during which the flux variations in the equilibrium and divertor control fields need only be relatively small, but throughout which the transformer-induced main plasma current must be sustained to provide the poloidal field component of the plasma confinement field.







There are 14 PFC units, 7 of which are in the upper half of the reactor, and 7 in the lower. These upper and lower units are connected in corresponding pairs, so that there are 7 independant power supplies to the 7 coil sets, of which the inner pair of the Central Solenoid (CS) is LN cooled copper, the remaining 6 sets being superconducting. Each power supply unit consists of several rectifier sets (in three of which the polarity can be reversed, and in some of which series. parallel and bypass connections can be made by rapid switching), and resistors which can be switched in to dissipate coil energy as required. The maximum input power during the cycle (during the 22 second start-up period) is some 200 MW, with some 3500 MW having to be dissipated in resistor banks during peaks of energy removal. This can be achieved by 1.7GVA of convertor capacity and 2.6 GVA of disharge resistors.

2.5 EXTERNAL HEATING: No details are yet available concerning the preferred choice of auxilliary heating for NET, but the assumption is that 50 MW of RF energy must be absorbed by the plasma to bring the temperature to the 10 keV required for ignition. This energy would be transmitted through 4 of the 16 Horizontal Access Ports (HAP's). The details of the wave-launching structure, and its corresponding degree of maitainability, would depend on whether the mode adopted was ICRH (Ion Cyclotron Resonance Heating, 0.025 to 0.12 GHz) (loop-array with Faraday screen inside the torus, with ceramic window in the wave-guide), or LHRH (Lower Hybrid, 0.8 to 4.6 GHz) (waveguide array, aluminium oxide or beryllium oxide window, may be placed close to the plasma); or ECRH (Electron Cyclotron, 25 to 140 GHz) (wave-guide, BeO or quartz window, close to plasma).

2.6 PLASMA CURRENT DRIVE: Optimistic reactor design studies aimed at demonstrating the long-term potential of fusion reactors have assumed that RF current drive, either as the sole drive method, or at least as a bridging drive whilst the inductive drive system is re-set, will facilitate steady-state operation, thus avoiding many of the serious disadvantages inherent in pulsed operation. These include great reduction of reactor lifetime due to fatigue stresses, the need for energy storage in relation to the thermodynamic power conversion plant, and many other detailed and derivative difficulties. However, RF (or NB) current drive has not yet been demonstrated as a practical matter, and for the NET design the assumption is made that the current drive will be entirely inductive by the innermost superconducting coils of the central solenoid, and that burn times of at least 200 seconds, possibly 450 s or 600 s, and hopefully 1000 s, will be attainable, with an assumed duty cycle of 0.9.

2.7 FUELLING TECHNIQUES: For the INTOR design the assumed fuelling method is a combination of gas-puffing and hydrogen-ice-pellet injection, and at this stage in development a similar assumption would appear reasonable for NET. The required tritium feed rate for INTOR for the specified fusion power of 620 MW(th) is estimated at 64 g/hr which compares with a consumption rate of 3.6 g/hr to produce the specified 620 MW of fusion power. The remaining 60 g of unburnt T (and the associated 40 g of unburnt D) must be recovered and recycled. In terms of the possible 600 second burn for NET, this corresponds to a consumption of 0.6gT/shot, and a feed of 10.6 gT/shot. The NET and INTOR pre-shot base pressure is 3E-05 Torr, which corresponds to about 0.1 g of tritium gas in the initial charge in the vacuum vessel. Clearly refuelling throughout the shot is essential.
The state of development of pellet injection technology and the in-plasma pellet behaviour are both at too early a stage to be able to define a design. Some estimates suggest that injection velocities as high as 10 km/s will be required to get the bulk of the pellet to the central region of the plasma before too much DT material is lost by ablation, and that to aid this conservation of pellet mass a large pellet of around 5 mm diameter and pehaps 30 mg in weight might be required. On the other hand, if a feed of gas to the outside of the plasma (gas-puffing) can be effectively absorbed, such deep injection of pellets may not be necessary, if indeed pellet fuelling proves to be needed at all.

2.7.1 PELLET INJECTION METHODS: So far a good deal of development of pneumatic-gun and centrifugal-force mechanical injectors has taken place in numerous laboratories, with demonstrated pellet velocities in the range between 1 and 2 km/s. This injector technology will be adequate for pellet inection tests on the existing large experimental tokamaks, but much further development using quite different techniques eg. electrical acceleration of pellets, cluster-ions or plasmoids, may be needed if the higher velocity range is found to be essential for NET or DEMO reactors.

Presumably, in NET, the pellet injectors would operate through one or more of the Horizontal Access Ports (HAP's), possibly several injectors being spaced uniformly around the torus, with a combined injection rate of perhaps 1000 pellets, depending on the required pellet size, in the 600 second shot.

2.8 ASH REMOVAL: The helium from the fusion reaction, the unburnt tritium and deuterium, and various impurity atoms, will leak from the plasma as ionised particles at a great range of velocities. The slower ions will flow along the field-lines at the outer edge of the plasma body and strike the divertor plates situated close to the null points at the top and bottom of the plasma, where they will both give up their kinetic energy and be electrically disharged, leaving the target in random directions. Some of these neutralised atoms will travel back towards the plasma, re-enter it, re-ionise, and once again, in the case of the fuel ions, take part in fusion reactions, possibly recirculating many times before ultimate extraction. The extent of this effect is not yet known with any certainty, but it may be considerable and have an appreciable effect on the burn-up fraction achieved. Other neutral atoms will travel towards the pumping duct and so be removed in the gas phase for eventual re-ciculation or disposal.

2.8.1 DOUBLE NULL: It has recently been found that it may not be necessary to place the divertors close to a pumping duct, and that indeed a divertor placed at the top of the plasma can be effectively pumped by a vacuum duct at the bottom. This has opened the way for the adoption of a double-null plasma with two divertors, but only the one pumping duct at the bottom of the VV. The provision of a pumping duct at both top and bottom was considered impracticable within the NET architectural concept, and for this reason the earlier designs of NET assumed a single-null plasma, with all the complications of an unsymmetric system and less total effectiveness of the divertor plates.

2.8.2 DIVERTOR TARGETS: The divertor plates in NET are mounted at the extreme ends of the inner blanket module, and so can be removed and replaced with the blanket units, their coolant supply system likewise being acomodated in the blanket module system. There are still various assumptions about the performance required of the divertor plates, depending upon the temperature assumed for the particles at the plasma edge. Optimistic assumptions lead to unclad water-cooled copper plates; more pessimistic assumptions have resulted in copper plates protected by tungsten-rhenium alloy or a thin film of liquid tin which can be self-healing in regard to particle sputtering damage.

2.9 PLASMA DISRUPTIONS: In all tokamaks excessive particle density or plasma pressure give rise to major magnetohydrodynamic instabilities which lead to rapid disruption of the confined plasma, resulting in the whole of the stored energy of the plasma being discharged to the walls. The initiation of the disruptive instabilities is connected with the various gradients within the plasma of magnetic field, current and stored energy. It is found that, in large experimental tokamaks, around once in about 100 plasma disharge cycle 'shots', a major instability occurs. This has potentially serious consequences which must be provided against in the design of the machine.

The two major effects are, first, serious localised melting and structural damage to the walls, and second, very large mechanical forces applied to the machine structure due to the interaction between the rapidly decaying plasma current and the magnetic fields of the tokamak coil systems. It is hoped that improvements in plasma stability design will be able to reduce the frequency of major disruptions in power reactors such as NET to once per 1000 shots or less, but it seems unlikely that they will be able to be avoided altogether and so must be accomodated. Less serious 'minor disruptions' are also encountered, which result in a wasting of plasma energy through enhanced plasma thermal conductivity, and must also be taken into account.

The stored energy which is leased in a major disruption comprises two parts: the thermal ene of the ions of the plasma, amounting to some 300 MJ in the 390 cubic tre plasma of NET, and around a third of this quantity in the f of the stored magnetic energy of the poloidal field system. The rmal energy is discharged most quickly, probably in about 10 ms in the magnetic energy decaying in around twice this time. About hal ie magnetic energy appears as resistive heating in the walls, th her half contributing further thermal energy to the plasma as magnetic flux lines cut through the conducting plasma body.

The stored thermal energy is tranported to the walls by ion bombardment and X-ray emission, the relative amounts depending on the content of partially-ionised impurity atoms in the plasma. In INTOR the assumption was about a third as X-rays. However, since both types of energy transport have very short penetration ranges in the wall materials, in effect they both cause essentially surface rather than volumetric heating.

In tokamaks the disruption energy flux is principally directed at the inner walls, but it is not uniformly distributed, and peaking factors of around 10 are commonly assumed for analysis purposes. The concentrated flux of energy causes intense local heating, melting and vaporisation. The vapour film may play an important part in reducing the rate of heating, so providing some self-protection. The vaporised material is lost from the wall, but some of the melted metal may remain partly in place and re-solidify, so restoring some of the integrity of the wall. The wall system will have to be designed to accomodate say one in a 1000 of the possible half to a million burn cycles of the reactor, ie. some 1000 or more major disruptions over the reactor lifetime. To this end attention is being focused on protective refractory tiles, metal (eg. tungsten) or ceramic (eg.graphite), mounted on the first wall, and cooled by thermal radiation to the first wall. In NET the tiles would be replaceable, either individually by remote manipulator, or with the blanket module.

3. CONFINEMENT, STABILITY AND SCALING:

It is expected that developments in plasma cross-section geometry, confining-field feed-back control, current profile control, distribution control of auxiliary heating, and density-distribution control via pellet or other refuelling techniques, will enable the parameters required for NET to be attained, but so far these have not been fully demonstrated.

3.1 BETA LIMITS: Theoretical analysis and experimental findings indicate a scaling law for beta:

$$\beta_m = \frac{gI}{aB}$$

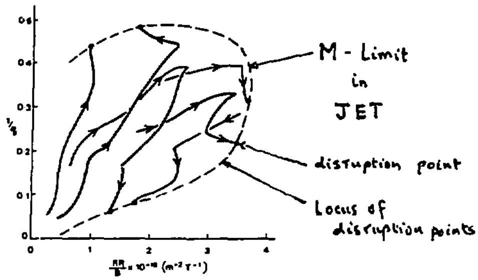
beta(total)(max) = g I(MA)/a(m)B(toroidal, axis)(T)

where g, the beta scaling factor, depends on the plasma elongation and triangularity (inter alia) and has values in the range 2.8 to 3.5: it is hoped to achieve a value of g close to 3.5 for NET, which for I=10.8 MA, a = 1.35 m, and B = 5 T corresponds to the target value of beta of 5.6%, but this in turn is dependant on attaining a value of q close to 2: the value specified for NET is q(separatrix) = 2.1.

3.2 DENSITY LIMIT: The density limit is primarily determined by the onset of major plasma disruptions, and is characterised by the Murakami scaling factor (M) and 1/q (= fraction of a poloidal turn per toroidal-direction orbit of the flux line):

Murakami factor = (average electron density)(major radius)/(B toroidal)

For the values specified for NET: n(electron,av.) = 1.56E20 per cu.m., R = 5.18 m, B (toroidal) = 5 T, M = 16E-19 per sq.m.Tesla, this would correspond to a limiting position on the M:1/q plot at the extreme upper RHS of the stability area. A value of M = 17E-19 for q(a) close to 2 has been achieved in Doublet III.



3.3 CONFINEMENT: A very simple scaling law has been adopted by the NET Team:

(electron confinment time, seconds)

$$\left[\tau_e = \text{Const.}\,\bar{n}_e\,(10^{20}\,m^{-3})\,a(m)\,R^2(m)\,q\right]$$

= (constant)(av.electron density)(mnr radius)(mjr radius squared)(q)

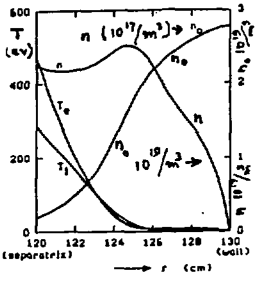
A value of the constant = 0.017 is consistent with the specified value for energy confinement of 2 seconds during the ignited burn (particle confinement is around three to five times the energy confinement time). Other empirical scaling laws give rather similar predictions for the NET confinement time.

It appears that a value of the confinement parameter
(energy confinement time)(plasma density) > 5.5E20,
with (plasma average temperature) > 10 keV,
must be achieved to realise ignition in NET.

$$\tau_E \bar{n}_e > 10^{20}$$
$$\bar{T}_e > 10\,\text{keV}$$

3.4 CONTROL of IGNITED BURN TEMPERATURE: This is as yet a relatively undeveloped subject, but the general principal would be to arrange for enhanced leakage of energy from the plasma body as ignited temperature increased. This might be performed through variable field-topology ripple-induced losses, but is at present thought more likely to be achieved through inherent small changes in beta introducing a measure of self-regulation through increased energy losses.

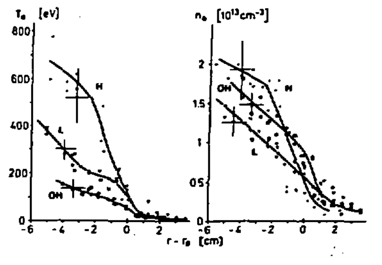
3.5 PLASMA EDGE REGION: Compared with the average plasma temperature of some 10,000 eV, the plasma edge temperature is expected to lie beween 100 and perhaps a few hundred eV, with a particle density between 2 E19 and 5 E19/cu.m. The plasma edge conditions greatly affect the amount of recycling of out-leaking ions which have become neutralised at the divertors or the first wall, which in turn affects the achievable burn-up of T and D, and the pumping duty required of the vacuum pumping system. These factors are affected beneficially if the plasma edge temperature is low (around 100 eV) and the density high (around 5E19/cu.m.



Plasma edge conditions, INTOR

In INTOR it is thought possible that these conditions could be achieved, and it is predicted that for these circumstances only some 1% of the flux of helium ions into the divertor would require pumping, the remainder recirculating (thus saving the duct area needed to intercept a sufficiently large fraction of the divertor-scattered neutralised particles), whilst the burn-up fraction could be in excess of 5%.
The relatively low temperature of the ions leaking from the plasma is expected to minimise the amount of sputtering of heavy atoms off the divertor targets, and the relatively high plasma density around the plasma edge is expected to reduce penetration of the plasma by these heavy neutral (impurity) atoms.
In INTOR the 620 MW of fusion power gives rise to 40MW of electromagnetic radiation to the first wall, and 4MW of particle energy (undergoing charge exchange), whilst 80MW of particle energy flows to the divertor plates (single null arrangement), of which about half is ion energy and half electron energy: around 10% of this energy leaks out on to wall regions neighbouring the divertor plates, which therefore have to be appropriately protected.

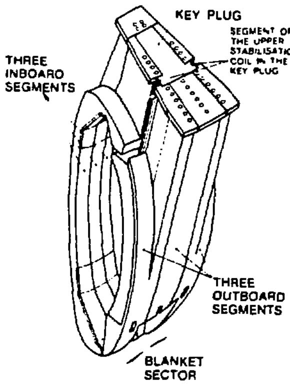
3.6 H-MODE OPERATION: It was found during experiments in ASDEX some years ago that when the poloidal divertor was in use, so that a separatrix was formed, the confinement time was significantly improved, with enhanced peaking of the particle density distribution. This condition has come to called the 'H-mode', but it appears simply to be a characteristic of divertor operation. The improved confinement produced by the H-mode is taken into account in the energy confinement scaling law given above. Although there is improvement with ohmic heating, neutral beam heating causes a degradation of confinement.



H-mode : Asdex [7]

4. FIRST WALL and TRITIUM-BREEDER/HEAT-REMOVAL BLANKET

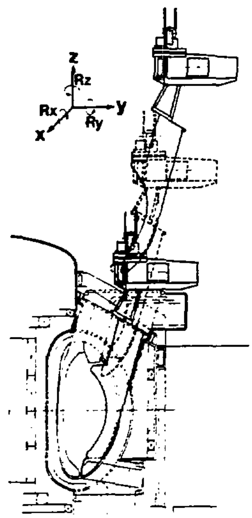
4.1 GENERAL CONSIDERATIONS: A basic feature of the NET concept is that the first wall, divertor targets and breeding blanket form an integral unit, so that all three major components of the reactor may be removed and replaced routinely during the reactor operating cycle. The various schemes so far developed employ either: first wall structures separated from the blanket and separately cooled, or first walls closely coupled to the blanket structure and cooled by a common coolant flow arrangement. In all cases there are 3x16 = 48 first wall segments, 3 segments forming one blanket sector, in each the outer and the inner blanket-unit rings. The segments are placed side-by-side but separated by a 2 cm gap; each segment consists essentially of a curved box-shaped structure enclosing the breeding blanket components, and has attached to its radial walls the electrically-conducting saddle loops which aid in short-term plasma stabilisation. These conductors have to be provided with cooling channels and must be protected by the first wall.



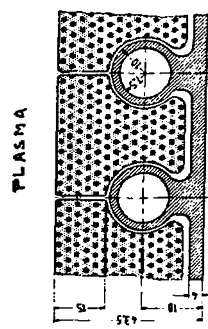
4.2 BLANKET CONCEPTS for POWER REACTORS: Over the past two decades many tritium breeding blanket concepts have been taken to various degrees of design detail, but despite several recent overall evaluations of the numerous possibilities, none has so far emerged as the clearly favoured candidate. A recent U.S. survey examined 130 concepts, and concluded that a liquid-lithium, self-cooled, vanadium-structure concept was the most to be preferred for tokamak reactors, with helium-cooled lithium oxide in a ferritic steel structure to be the most promising of the solid breeder concepts. The Japanese reactor studies have included several possible options, although the conclusions are not so clear-cut, but seem to favour a solid breeder approach, based on lithium oxide, with helium cooling generally preferred over water on grounds of better control of temperature for tritium removal, and less risk of tritium leakage into water systems in the power circuits. The European viewpoint at the time of the U.S. and Japanese reviews (1983/84) seemed more conditioned by the large costs of investigating numerous blanket concept options, in particular as it relates to self-cooled liquid lithium, considerable investment already having been made in lead/lithium alloy (liquid) and lithium aluminate (solid) concepts. The lead/lithium alloy with water cooling in a stainless-steel structure became the reference concept. A crucial question is the attainable overall breeding ratio, depending on the achievable local breeding ratio values (which may fall in the range 1.1 to 1.7, depending on many factors, including especially the neutron multiplier material), and the attainable blanket coverage fraction (which would seem to be unlikely to be any greater than 80% in a power reactor). Much reliance is placed on the use of beryllium as a neutron multiplier, but it is by no means yet clear whether beryllium is a feasible world resource for a major power system, nor the extent to which it could be recycled.

4.3 BLANKET CONCEPTS for NET: There has evidently been little change of outlook in the formal development of the European work on blanket concepts as it relates to the more recent effort being devoted to NET. However, it is particlarly notoworthy that the self-cooled liquid-metal design nominally predicated upon flowing lead/lithium alloy in fact turns out to be substantially superior in terms of neutronic and materials-corrosion performance if pure lithium is substituted for the lead/lithium alloy. Nevertheless, although the novel design concept of electrically-insulating duct-cladding 'Flow Channel Inserts' (FCI's) would appear to have overcome the problems of electromagnetic pressure drop, the safety problems of lithium (air/water reactions) remain to be solved. The other blanket concepts so far reported in relation to NET relate to the reference concept of slow-flowing Pb/Li alloy in stainless-steel containers cooled by tubes containing forced-convection flow of pressurised water in stainless-steel tubes immersed in the alloy; and three versions of ceramic (lithium aluminate or silicate) breeder material cooled by pressurised helium.

4.3.1 BLANKET REMOVAL and REPLACEMENT: A major design innovation in NET is the arrangement for removal and replacement of the blanket sectors. *The blanket sectors* are suspended from their top cover-plate/flanges upon the edges of the upper access port of the Vacuum Vessel (VV), and are guided and located by approriate grooves in the lower part of the VV. To remove a sector the secondary vacuum containment of the reactor is opened, the coolant pipes and other services decoupled, the key plug removed, and then the central outer blanket segment withdrawn by direct oblique lifting. This unit is then followed by the central inboard segment, after which the outer segments can be moved, in turn, into the central position under the access port and removed in the same way.

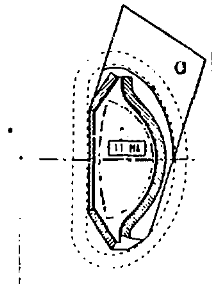


4.3.2 FIRST WALL CONCEPTS: Since these have been developed in association with their related blanket concepts they are dealt with below in the paragraphs dealing with each blanket. However, there are certain common considerations. First, it is not yet determined for NET whether an unclad stainless steel first wall will be acceptable, or whether protection by refractory tiles will be essential. Present asumptions tend to lean towards the tiles, graphite at present being the preferred option. For the case of NET, with 1MW/sq.m. neutron wall loading and 0.4 MW/sq.m. thermal load, such tiles would operate around 1400 deg.C. In some power reactor concepts the use of tungsten tiles has been proposed, and indeed tungsten tiles may be needed for divertor protection in NET. In all cases the tiles would radiate their heat to the first wall immediately behind. Their presence would reduce the breeding performance of the blanket, and add to construction and maintenance costs. The adjacent diagram shows a helium-cooled first-wall tile concept.



He gas-cooled 1st wall

4.3.3 INBOARD BLANKET SEGMENTS and DIVERTORS: In NET, which is intended to have the maximum feasible degree of flexibility, provision is made to instal non-breeding blanket segments which are considerably thinner in the radial direction than would be breeding blankets, and so enable the use of a slightly larger plasma chamber for experiments on plasma shape and current. The option is also available to have a breeding blanket outboard segment in association with a non-breeding inboard segment. Such a segment could be fitted with a beryllium liner behind the first wall to give additional neutron multiplication.



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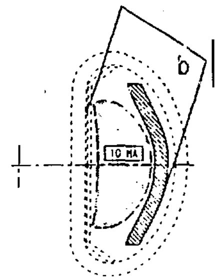

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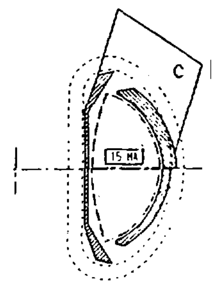


EXTENDED

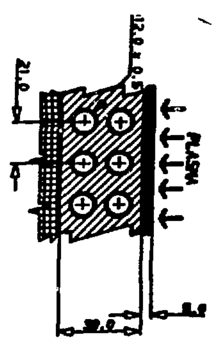
In all cases the top and bottom extremities would carry the divertor target plates. Some preliminary design work has been performed on such targets, one concept being to use pressurised-water-cooled copper blocks 39mm in radial thickness, inside which are embedded and bonded two layers of 11mm bore stainless-steel pipes, 0.5 mm wall thickness, serving as the go and return pipes of a U-tube running poloidally the full height of the segment, and manifolded in a low-radiation zone in the upper structure of the blanket sector. The copper blocks, or plates, would be supported on a stainless-steel backing-structure/radiation-shield, and the front surface of the copper would have deposited upon it, and bonded to it, a 5mm thickness of tungsten/5%rhenium-alloy.

The total surface thermal loading upon the inboard blanket would be about 100MW, plus another 100MW due to volumetric heating by neutrons. The cooling water would enter at 6MPa, with a pressure drop of 0.6MPa, inlet temperature 50deg.C, outlet 116deg.C. It would obviously be quite unacceptable in a power reactor to have a third of the reactor's heat output delivered at this low temperature, but for the NET experiment it would be satisfactory.

The highest power loading at the divertor plate region would be 16MW/sq.m, but this would be reduced at the plate surface by angling the plate at 18 degrees to the outer separatrix, resulting in a peak heat flux at the plate surface of 5MW/sq.m.

Temperature estimates for these divertor plates are: tungsten layer: 550 deg.C, copper heat sink 400, copper/steel interface 310. Stress analysis indicates that the interfacial thermal stresses could be made acceptable.

The part of the inboard-segment first-wall between the upper and lower divertor plates would be clad with graphite tiles, and perhaps equipped with limiters: in the proposed burn cycle the plasma toroid would initially be formed in contact with the inboard wall (as is now done in JET), which may accordingly need special protection. The heat flux to the first wall would be less than a twentieth of that to the divertor targets.



4.4 BLANKET DESIGNS, LIQUID and SOLID: There are three important criteria which should be met by any blanket system: the tritium should be able to be extracted continuosly, the inventory of tritium in the blanket materials should be minimised (to keep the partial presure of the tritium below that at which permeation through the containers into the working space and environment would become unacceptable, and to avoid a catastrophic release in event of accident), and the materials should be adequately durable and be such as to minimise accident hazards. Many of today's conceptual designs do not succeed in meeting all of these criteria simultaneously.

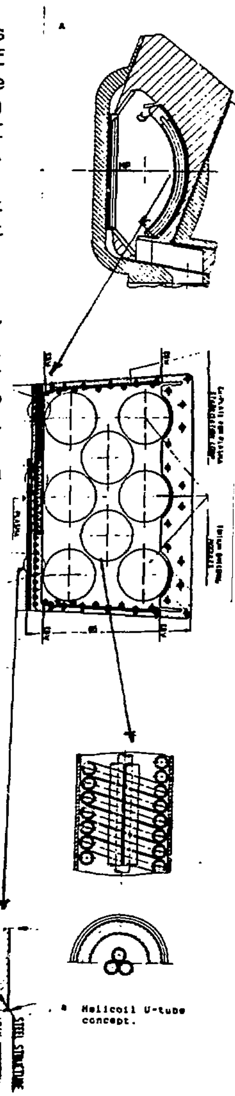
4.4.1 LIQUID BREEDER BLANKETS: The present concepts for NET are based on lead/lithium eutectic alloy (Li17Pb83, density 9.51gm/cc at 300deg.C, contained lithium density 0.066gm/cc, melting temperature 234.7deg.C). Pure liquid lithium (MP186deg.C, BP1336, density 0.5g/cc) offers certain attractions, including better breeding performance and less corrosion problems, but it has air/water reaction hazards and a much larger tritium hold-up.

4.4.1.1 PRESSURISED WATER COOLING: The reference design is a drastically modified variant of the European version of the INTOR blanket, based on slowly-circulating (somewhere between one blanket inventory per hour to one per day) lead/lithium alloy , cooled by pressurised water, with AISI 316L stainless-steel as the structural material throughout. Water pressure would be 11.5 MPa, pressure drop 0.6 MPa, pumping power around 1.5% of the gross electrical output (were electricity to be generated), and inlet and outlet temperatures would be 265 and 305 deg.C respectively.

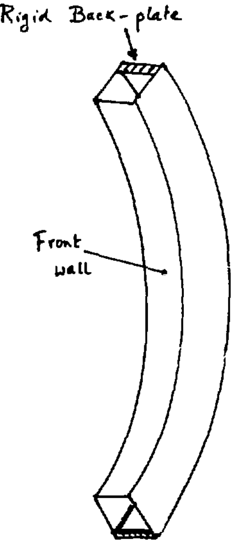
Both the inboard and outboard segments would carry breeders. No neutron multiplier is incorporated in this preliminary design, and the packing density of breeder materials is not as high as it could be made by using variable-diameter modules, but using 20% enriched lithium a net breeding ratio of 0.4 could be obtained, which would probably be sufficient for NET experimental purposes (it is expected that the greater proportion of NET tritium will be imported from Canada).

Numerous other improvements are conceivable (eg. martensitic in place of austenitic steel, thicker blanket, possible elimination of passive electrical loops giving better coverage, reduction of amount of tile protection etc.) which could greatly improve the breeding performance of this type of blanket were it decided to develop it intensively.

Two types of immersed water-tubes were considered: straight or helical. The sketch shows the 'helicoil' concept, which offers the advantage of less header connections, and those needed can be made outside the radiation zone, in the top of the blanket module. Each helicoil unit would consist of two or three parallel nested helices, each contiguous with its downcomer (3-tube version shown). Development tests have demonstrated the feasibility of fabricating these units.

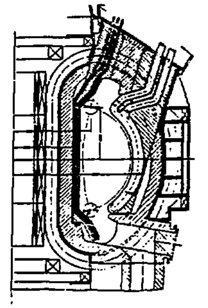


Helicoil U-tube concept.



Several first wall concepts have been examined. These would be free-standing, using the curved box concept, and cooled by a separate coolant supply. The front and side walls would be made from stainless-steel plate about 1cm. thick, to the rear side of which would be brazed tubular water-cooling pipes, running in either the toroidal or the poloidal direction. Helium might be the preferred coolant for an eventual reactor development. Graphite tile protection would probably be employed. Within this curved box the blanket unit would have electrical conductors associated with the side-walls for the plasma stabilisation loops.

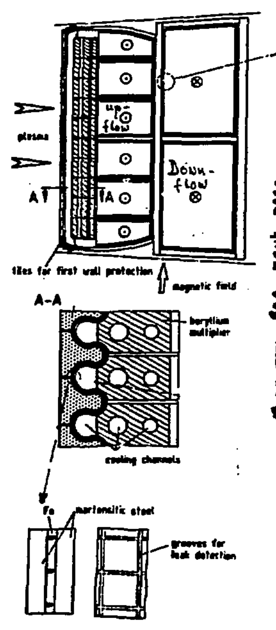
4.4.1.2 SELF-COOLED LIQUID METAL BLANKET: This blanket concept uses lead/lithium alloy, with the lithium enriched to 90% Li6, (or conceivably liquid lithium alone, natural or enriched) as both tritium breeder and sole heat removal agent for both breeder and first wall; a beryllium metal neutron multiplier; and a martensitic steel structure. It can produce a sufficiently high breeding performance (more than 1.1% overall, assuming 100% blanket coverage of the outboard region) that, for NET purposes, adequate tritium production could be achieved using an outboard breeder only, the inboard segments being either for VV shielding only, or possibly, with the addition of an appropriately cooled beryllium layer, also serving as an additional neutron multiplier zone.



The key feature of this concept is the reduction of magnetohydrodynamic (MHD) pressure drop to an acceptably low value through the adoption of two important techniques:

* arranging for the high-speed (several m/s) liquid-metal flows essential for adequate cooling of the first wall to be in the toroidal direction, minimising the cutting of field-lines by the flowing liquid metal conductor

* for the poloidal-direction flows, which cut the field lines almost at right-angles, insulating the liquid metal from electrical contact with the structural/duct walls through the use of Flow Channel Inserts (FCI's), and only using flows in the poloidal direction for heat removal duties requiring but relatively low flow speeds (below 1 m/s).



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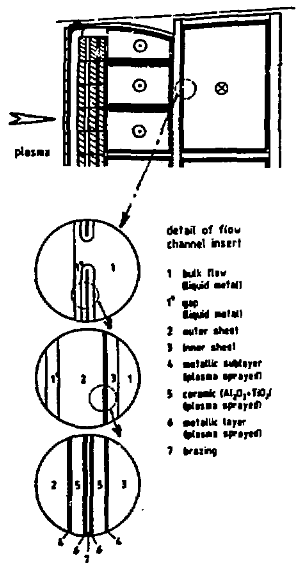
A significant contribution to the reduction of MHD pressure losses comes about because, since the breeding performance is so good, it is only neccessary to use outboard segments for the breeder units, in which location the magnetic field is only about 60% of that at the inboard segments.

The limit* to the pumping of a liquid metal through an electrically-conducting pipe in the presence of a transverse magnetic field is the limiting stress in the pipe wall; but high pressure drop also consumes additional pumping power and may become unacceptably large even if the stress limit is avoided. Assuming an average magnetic field in the blanket region of 4T, the calculated pressure-drops for the various flow zones are:

service tubes:	inlet 0.17 + outlet 0.17	= 0.34	MPa
poloidal channels:	downflow 0.13 + upflow 0.47	= 0.6	"
180 degree bend:		= 0.1	"
toroidal front channels:		= 0.3	"
	Total:	1.34	"

The FCI's fit loosely into the flow ducts, and comprise a sandwich consisting of a thin layer of ceramic insulation (the "meat") firmly bonded to two sheets of steel (the "bread") of unequal thickness. The thin steel sheet on the side of the FCI in contact with the main flow is made as thin as possible to maximise its electrical resistance, but sufficient to provide an adequate corrosion allowance: its thickness might be 0.5mm. The sheet on the duct-wall side would be several times thicker to provide adequate structural strength to the assemblage. There would be a stagnant layer of coolant between the FCI and the wall for pressure-balance.



The performance of such an FCI is indicated by an example of a 400mm diameter duct with 12mm wall thickness (presumably in a 4T field):

without FCI 3.9 MPa/metre duct length;
with FCI 0.16 MPa/m.

The convoluted first wall provides both for anchorage for the graphite tiles and for double-containment combined with leak and crack detection.

4.4.2 SOLID-BREEDER BLANKETS: Preferred lithium-bearing ceramic materials in the various European blanket concepts are lithium aluminates and silicates, but there is some interest in the material favoured in the USA and Japan, lithium oxide, which has the highest available lithium density. These materials are usually pressed and sintered to around 85% of theoretical density to provides sufficient inter-connected porosity to enable the bred tritium to escape into the extraction, or purge, lines. The sintered bodies may be in many possible forms, but the three designs at present being considered are rectangular bricks, circular discs and 0.5mm diameter pebbles. In two cases the coolant flows inside tubes, the ceramic being outside.

Helium is the reference coolant in all cases, but possibilities for other coolants, especially water, are being kept in mind. Water-cooled first walls have the disadvantage of reducing the attainable breeding performance due to neutron moderation.

* L (max, m.) = Stress(max. allowable, Pa) / B^2 (T) Velocity (m/s) wall conductivity (A/Vm)

The close-fitting bricks or plates have to have holes, spacers or surface grooving to facilitate the movement of the purge gas. The pebble bed has its inherent gross porosity to accomodated the purge flow.

The purge gas is at present specified as 0.1% hydrogen in helium. The hydrogen is believed to improve the release of tritium and also to ensure that the partial pressure of HT is limited to 0.2Pa, resulting in the total leak-rate of T to the coolant being held down to 0.1g/d.

All the solid breeder designs rely on beryllium metal for neutron multiplication. In two designs this is all concentrated at the plasma end of the blanket, but in the disc design it is dispersed, heterogeneously, throughout the blanket volume, and serves as the principal conductor of heat from the ceramic discs to the coolant. Direct contact between ceramic and beryllium is prevented, in order to avoid corrosive reactions, by interposing thin steel barriers. It is also essential to make provision for the growth of the beryllium, which may amount to as much as 10% by volume, on account of helium formation due to the neutron reactions.

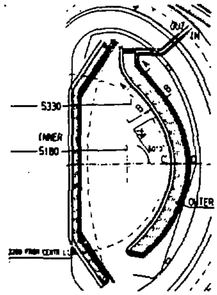
An important condition for the solid breeder materials is to maintain their temperatures within the limits of around 400 and 800 degrees centigrade to ensure that the material is hot enough (above 300 deg.C) to release the bred tritium, but cool enough (below 900) to avoid sintering and the closing of porosities. An advantage claimed for the solid breeders is that their hold-up of tritium is very low, possibly only 100g or so for NET.

In all cases the designs provide double cotainment of the blanket materials and coolant(s), in some cases by the free-standing curved first-wall box system described earlier, in others by a more closely integrated arrangement.

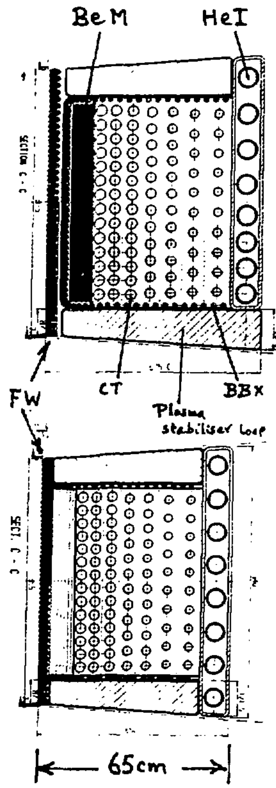
Since the breeder material does not conduct electricity, as it does in the case of the liquid metals, it is necessary to surround each breeder segment with an electrically conducting loop to aid in the stabilisation of the plasma. This could be of copper or aluminium, the latter being favoured. Since these conductors have to occupy the full radial width of the blanket (around 60cm), and must be some 10cm thick, they represent a considerable inroad into blanket coverage fraction and available breeder volume. It is conceivable that developments in plasma control may enable them to be dispensed with.

There is as yet no clear preference for separate or integrated first walls, or for a common or a different coolant. In all cases provision is made for graphite tile protection of the first wall, and the reduction in breeding performance consequent on the neutron moderating effects is indicated.

4.1.2.1 SINTERED CERAMIC BRICK: This concept is based upon gamma lithium aluminate ($LiAlO_2$), beryllium metal multiplier, helium coolant, and AISI316 stainless steel structure. The assemblage of 7 rows of poloidal-direction coolant tubes (CT) in the outboard blanket segment (3 rows in the inboard segment), the breeder bricks (BB) and the beryllium multiplier (BeM), is contained in a helium-cooled blanket box (BBX). This box has a constant cross-section over its whole poloidal length. The first wall (FW) is either integral with, or slightly separated from, the blanket box.



The helium coolant inlet tubes (HeI) are situated behind the breeder region and convey the gas to the breeder-inlet manifold at the bottom of the segment.



Helium pressure is 4MPa, inlet temperature 100deg.C, outlet 300, pumping power about 0.2% of the heat output. The breeder power density varies from about 12watts/cc (MW/cu.m) at the plasma end to 1watt/cc at the VV end of the breeding region. The maximum ceramic temperature is maintained below 700deg.C. However, in the design as developed so far no special provision has been made in regard to setting a lower limit to ensure tritium release, dependence being placed on heating the materials after shut-down.

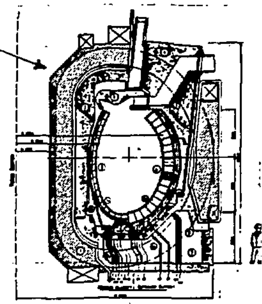
There is a purge flow, presumably of helium with 0.1% hydrogen, to remove the bred tritium. This flow takes place along radial channels in the brickwork formed by flats on the edges of the bricks.

The poloidal arcs of the aluminium conductors (forced convection cooled: coolant arrangements not shown in sketches) for the plasma stabiliser loops (PSL), some 10cm thick at the equator, are atached externally to the sides of the blanket box and extend upwards and downwards poloidally until their width is reduced to around 2cm, at which point the loop is completed by toroidal-direction connections.

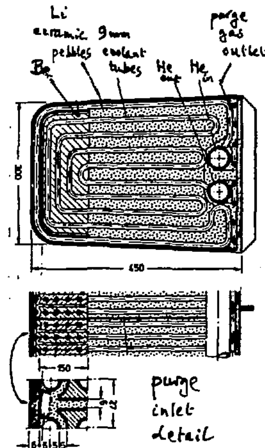
The lithium is enriched to 30%Li6. The overall breeeding ratio of the entire blanket would be just over unity, but the occupation of some 20% of the wall area by he plasma stabilisation loops reduces this to about 0.9.

No special designs for the divertor targets have been described.

Scheme for a single-null NET variant

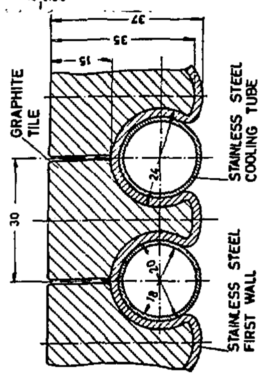


4.1.2.2 *CERAMIC PEBBLE BED: In this concept the* breeding material is lithium silicate (Li4SO4) formed into sintered 0.5mm pebbles cooled by pressurised helium (60 atmospheres) flowing in radially-directed tubes. The pebble bed is contained in numerous austenitic stainless-steel cannisters, of width (almost) that of the blanket segment, and height in the poloidal direction about 30cm, some 18 cannisters being stacked in the poloidal direction, attached to the structural back-plate. The helium supply and return pipes run poloidally just behind the cannisters, and from these the connections are made to the toroidal-direction manifolds supplying the first-wall and cannisters.



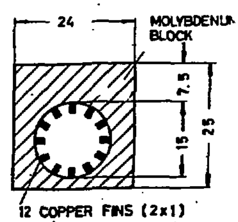
The cannisters for the outboard blanket are 45cm long in the radial direction: the first 15cm. is occupied by the beryllium metal multiplier, the remaining 30cm. by the pebble bed. The same cooling tubes that remove heat from the pebble bed flow in the poloidal direction through the beryllium to cool the multiplier.

The first wall is also helium cooled, at a pressure of 6MPa, inlet temperature being 200deg.C, the outlet from this flow (at 238deg.C) being collected in toroidal manifolds and directed to the cannister cooling system, from which it emerges at 450deg.C. The cooling system design is such as to maintain the breeder temperature within the range 270 to 900deg.C.



The helium purge flow (containing 0.1% hydrogen) is supplied at 1bar (0.1MPa) from porous pipes alternated between the coolant tubes brazed to the front walls of the cannisters, and collected at porous collector grids at the cannister back wall. Internal stiffening is provide to ensure that the cannisters can withstand an accidental overpressure to 0.6MPa. The tritium inventory of the whole blanket is estimated to be 120g, with total plant losses of less than 10Ci/d.

The plasma stabilisation loops are of copper, in the form of 13mm plates attached to the side walls of the outboard blanket first-wall box, inter-connected by copper sheets 50cm. in (poloidal) width, 2.2mm thick, placed just behind the first wall.



The divertors are formed from 24 square-section 24mmx25mm molybdenum rods placed side-by-side to form a plate 210cmx60cm. Each rod is internally cooled by helium flowing in a 15mm diameter channel, internally finned with copper.

By providing an inboard blanket (cannisters only 12cm. long) the breeding ratio can reach 1.31: with only beryllium in the inboard blanket this woud be reduced to below unity.

4.1.2.3 CERAMIC-DISC-STACK-IN-TUBE: In this concept the multiplier and breeder materials (beryllium metal and gamma lithium aluminate, $LiAlO_2$, 60%Li6) are fabricated as discs (just under 6 or 8cm diameter) and incorporated into 6 or 8cm diameter tubular breeder elements, helical-wire-wrapped, contained within tubular cannisters (9cm front row, 8cm discs; and 19cm, three rear rows, 6cm discs), arranged poloidally inside the curved first-wall-and-containment box. The 6cm tubular breeder elements in the cannisters in each of the rear three rows are arranged in a close-packed array of seven : the first row unit, being more highly rated, contains only one 8cm tubular breeder element.

The helium coolant (6MPa) enters at one end of a cannister, flows over the external surface of the tubular breeder element, leaves the cannister at the other end, and passes on to the next cannister, in a series-flow arrangement. The pressure drop through the whole system is around 0.06MPa, the pumping power being about 1% of the heat removed. The coolant inlet temperature is 250 deg.C, outlet 500. The ceramic material is maintained in a favourable temperature range, 440 to 500 deg.C for 80% of its volume.

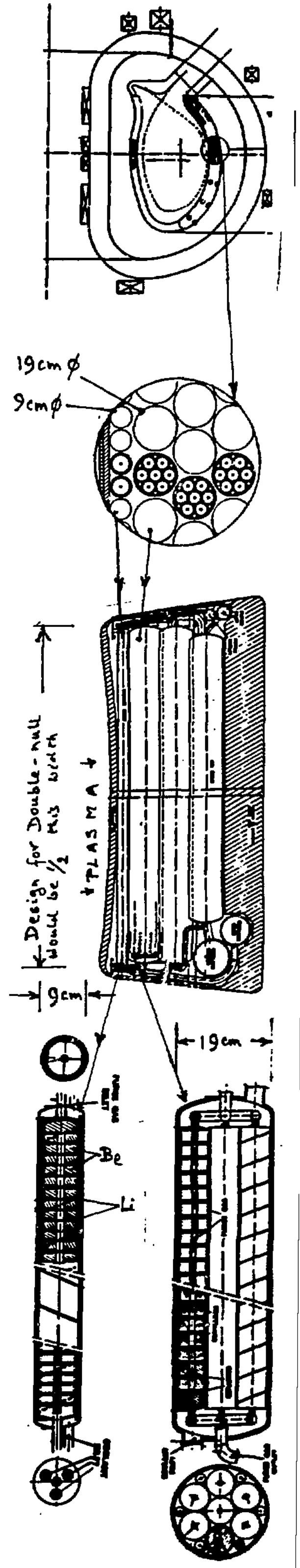
The tubular breeder element consists of a thin-walled steel cladding tube containing of alternating thick (circular) discs of beryllium metal and thin discs of sintered lithium aluminate (85% by volume beryllium, 15% enriched lithium ceramic), ceramic and metal being separated from each other by thin steel discs. Spring-washer end-loading arrangements accomodate beryllium swelling and aid thermal contact between the discs. Dependence is placed on the good thermal conductivity of the beryllium to conduct the heat from the ceramic to the breeder element cladding.

The purge gas enters the tubular breeder element at one end and leaves at the other (flowing in the opposite direction to that of the coolant flow). A system of grooves and slots in the discs causes the purge gas to flow radially over the ceramic dics. As with the coolant, the purge flow from one element flows on to the next. The exit partial pressure of the tritium is estimated as 0.2Pa of HT, from which the permeation of the coolant is assessed to be less than 0.1g/d.

The overall tritium breeding ratio is estimated to exceed unity, but depending on many variables eg. amount of graphite in tiles, of steel in structure.

There is concern that this system involves a very large number of pipe-joint welds in the irradiated zone, and, in common with all the ceramic/beryllium schemes, there are as yet unresolved questions on corrosion.

Scheme for SINGLE-NULL variant



5. EXTERNAL SYSTEMS and ENVIRONMENTAL IMPACT

A very considerable amount of auxilliary plant is essential to the operation of a fusion reactor, including electrical, mechanical, chemical and nuclear. This note is confined to the chemical and nuclear aspects.

5.1 RECOVERY of *HYDROGEN* ISOTOPES *from* the REACTOR SYSTEM: There are three major sources of hydrogen isotopes: first, the plasma exhaust containing unburnt fuel, fusion-produced helium, and an unknown range of impurities; second, the tritium bred in the blanket system; and thirdly, tritium which has leaked into the coolant flows due to permeation through duct walls. All three of these sources may in turn leak routinely in small quantities, or accidentally in large quantities, into the reactor and plant containments. These leakages must be dealt with by "Air Detritiation Plants".

Tritium is both valuable and hazardous. The present purchase price is around $10million/kg, or $10,000/gram. It is most hazardous in its oxidised form as T2O or HDO, in which state the Maximum Permissible Concentration (industrial level) is 5microcuries/cu.metre, which coresponds to only 5 micrograms of tritium as T2O or THO in a 5000cu.m. reactor containment volume: this has to be compared with an inventory inside the reactor of a few hundred grams of tritium. Evidently the requirements to minimise tritium leakage are extremely stringent. There is a great premium on avoiding all processes which may oxidise leaked or leaking tritium, since the tolerable level for the unoxidised (gaseous T2) form is 4E05 times that of the oxidised (water vapour) condition.

5.2 TRITIUM RECOVERY from BLANKET MATERIALS: Two cases have to be considered: liquid blankets and solid blankets. All the tritium plants are likely to require total remote operation except during start-up and shut-down.

5.2.1 *LIQUID METAL BLANKETS:*

5.2.1.1 WATER-COOLED LM BLANKETS: Inthe case of water-cooled liquid-metal blankets, first there will be some degree of permeation of tritium into the coolant water. This tritium must be recovered, which calls for one specialised type of plant. Secondly, the blanket LiPb alloy must be recirculated through another type of plant, possibly based on a permeable metal (eg. Zr, Nb, V) membrane through which the tritium could permeate (and be simultaneously oxidised by a catalyst layer) into a flow of helium, from which in turn it would be extracted by cold-trap or molecular sieve, then electrolysed to release the hydrogen isotopes in gaseous form. All the flows of recovered hydrogen isotopes from the numerous types of recovery plants would then be combined and passed into a cryogenic distillation plant for isotopic separation and onward flow to the fuel preparation plant.

5.2.1.2 SELF-COOLED LM BLANKETS: In the case of self-cooled blankets, in which the LiPb flow transports the entire heat output of the reactor, there is the possibility that it would be necessary to interpose a non-radioactive secondary liquid sodium loop between the LiPb circuit and the steam circuit. In this event the tritium would permeate through the structure of the LiPb/Na heat exchanger to form NaT, which could be removed by cold trapping, the tritium later being recovered by heating. There are many other possibilities.

5.2.2 SOLID CERAMIC BLANKETS: In this case the tritium is present as THO in the helium purge flow and would possibly be recoverd by a molecular sieve technique.

5.3 FUEL PELLET and GAS PUFFING INJECTION: It seems probable that fuel will be injected as an equimolar mixture of T and D, either as gas or as hydrogen ice pellets or both. Numerous other injection technologies are under consideration.

5.4 TRITIUM LEAKAGE and ENVIRONMENTAL HAZARD: It is believed that a large accidental tritium leakage constitutes the main radiological hazard from DT fusion plants. Recent developments have indicated that the kilogram inventories of tritium earlier thought necessary for fusion reactors may be able to be reduced by a factor of ten or more. Special care would be taken to isolate from each other plant units containing the largest inventories, that of any one possibly not exceeding 100 grams. If plant reliability could be made sufficiently high to demonstrate that accidental releases of this sort of quantity from an isolated plant unit were unlikely to occur more frequently than once per 100 years, and that off-site doses from such an accident could not exceed 10rem, then it may be that this would win public acceptance and licencing approval.

5.5 RADIOACTIVATION of STRUCTURAL MATERIALS: A favoured objective is that no neutron-activated material or component from a fusion reactor should be such that it would not be suited to shallow land burial for ultimate disposal, if necessary after a reasonable delay-and-decay period. To this end much thought is being given to the selection, or special formulation, of structural materials have low activation products. For example, an analysis has been made of the activation of stainless steels, in which nickel and molybdenum produce a high level of long-lived radioactivity (>100yr). One suggestion is to substitute manganese for these elements to perform a similar metallurgical function. It has even been contemplated that especially harmful isotopes of desirable alloying metals could be separated and rejected before the alloy additions are made.

5.6 GENERAL ASPECTS: Because there are so many and such varied possibilities in regard to auxilliary plant their concept and design cannot be taken to a high degree of detail until the precise needs of the fusion reactor plant become more narrowly specified. However, it is most important that tentative schemes should be developed so that those responsible for reactor design may have some idea of what support processes and facilities are likely to be feasible.

6. CONCLUSIONS

Mankind's growing numbers and rising expectations demand the large-scale development of electric and other power sources. In the long term, only nuclear energy sources are fit to meet this need as the predominant provider. For large-scale, widespread use, fossil fuels and renewables will be environmentally unacceptable or insufficient or both. Fusion of light elements may well provide a nuclear power source offering more control in design in regard to the radioactive wastes than is possible in a fission plant, both in relation to nuclear characteristics and to quantity. There is no shortage of fuels: fossil fuels will last for centuries, fission fuels for millenia, fusion fuels for aeons.

The technology for developing fusion power appears at present to be immensely complex, and there is no indication at present that it will become simple, desirable though this may be. Therefore, it is most unlikely that commercially efficient fusion plants can be developed in one step. The urgent matter now is to develop a practical system that works at all and can be demonstrated to be able to deliver potentially useful power. It appears that NET could fulfil this important role. The testing-out of the whole range of technologies essential to the production of useful power, and the capability to compare several alternative schemes, especially in the breeding and heat removal areas, are most important attributes of the NET concept.

It is very evident from studying the published material on NET that the plasma engineering has been developed to a far more detailed and practically-implementable extent than have the complementary materials, nuclear and thermohydraulics technologies for tritium breeding and the extraction of heat essential to the development of useful power. A measure of this situation can be gathered from the claim that, world-wide, less than 1% of the fusion budget has been expended on reactor design studies, and, recognising that this effort has mostly been of a design and assessment nature, it is evident that very little indeed has been done in regard to the practical development of these aspects. Until there is a clearly defined program for the development of an actual power-developing reactor, as opposed to another physics experiment, these technology aspects will remain as untested ideas on paper, with no incentive to develop realistic, operable systems. It seems clear already that the mere hope that NET might become a real entity in the foreseeable future has already gone some way to stimulating designers towards developing possibly-feasible concepts, but there is no doubt a long way yet to go to catch up with the plasma engineering development.

Were it to be decided simply to build a next-stage physics experiment solely to demonstrate break-even and ignition, using low-temperature short-pulse technology just sufficient to meet the needs of physics tests, then, on the successful completion of such physics demonstrations, there would need to commence many years of technological development in the complementary fields before a practical power system could be demonstrated. Such a long delay could be very damaging to public confidence in, and hence to the funding of, the entire fusion program.

The super-conducting-coil, low-beta, low-power-density, pulsed-output tokamak may appear to be a conservative system never likely to be economic as a power producer. Nevertheless, it does seem very likely actually to work in the near term, whereas the seemingly more-likely-to-be-commercially-viable, high-power-density, high-beta, and possibly all-copper-coil devices, for example the Reversed Field Pinch, or some Compact Torus schemes, perhaps able to burn one of the alternative fuels (eg.D-D), need many years more physics investigation before one or other of them could have a chance of being developed into a reliable pre-demo test-bed for systems, components and materials. Should the superiority of one of the improved schemes eventually come to be confirmed then much, if not most, of the development work which could by then have been well under way for or in NET, could be incorporated into the improved designs and the new system then be developed far more rapidly than would be possible without the development experience from the tokamak.

Meanwhile, the low power density of the low-beta tokamak may offer some very positive advantages, including the large heat capacity of the shielding and magnet systems as a storage for the shut-down heat from radioactive materials in event of coolant system failure, and a sufficiently low wall loading to permit a protracted period of experimentataion using conventional materials. Until a fusion reactor has been built large-volume materials testing with 14MeV neutrons will be impossible and effective materials improvement impracticable.

It is probably important to make clear that the blanket technology, although superficially resembling a thermal fission reactor, and able to be analysed using generally similar methods, is in fact very different indeed from any of the fission reactors which have now become conventional. The liquid-metal self-cooled blankets have some resemblance to some of the early liquid fuel fission reactor schemes, but also a great many differences; however, not only were the liquid fuel systems never fully developed, but the personal expertise of those who worked on those schemes have long since been dispersed. The solid-fuel blanket concepts may seem to have resemblance to particle-fuel HTR's: but, again, the differences are quite profound, the desire in the particle fuel being to retain the fission products to a very high level of leak-tightness, whereas in the solid blanket for a fusion reactor the objective is for the bred tritium to escape as readily as possible from the ceramic particles. These examples should serve to indicate that there will probably need to be as much development work to produce a successful, reliable blanket system as was required to develop a complete fission reactor system: this is by no means a minor undertaking, such as could satisfactorily be deferred until all the plasma physics has been resolved. The objective of nuclear fusion research is to develop a power plant, not to enhance the knowledge of plasma physics for its own sake: the whole fusion energy concept must therefore be developed as an integrated whole.

There is great scope for international collaboration in the development of the tokamak power reactor concept, both in the plasma and in the nuclear aspects of the enterprise. It would seem most desirable for Australia to have a share in this important segment of technological change in <u>both</u> the major areas of development.

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MAJOR PARAMETERS of NET-DN

PLASMA MAJOR RADIUS	R	(m)	5.18
PLASMA MINOR RADIUS	a	(m)	1.35
PLASMA ELONGATION	b/a		2.18
PLASMA TRIANGULARITY	c/a		0.65
ASPECT RATIO	A = R/a		3.8
PLASMA VOLUME	V	(cu.m.)	390
FIELD on AXIS	B	(T)	5.0
TF RIPPLE, plasma edge, outboard	delta	(%)	+/- 1.2
PLASMA CURRENT	I	(MA)	10.8
SAFETY FACTOR	q(a)		2.1
BURN TEMPERATURE (average)	T	(keV)	10
DT DENSITY (average)	n(DT)	(E20/cu.m.)	1.20
ELECTRON DENSITY (average)	n(e)	(E20/cu.m.)	1.56
MURAKAMI PARAMETER	M	(E19/sq.m.)	16
TOTAL BETA	beta(tot)	(%)	5.6
POLOIDAL BETA	beta(pol)	(%)	1.42
FUSION POWER	P(fus)	(MW)	600
ALPHA-PARTICLE POWER	P(alpha)	(MW)	120
FUSION POWER DENSITY	P(vol)	(MW/cu.m.)	1.53
NEUTRON WALL LOADING	Q(n)	(MW/sq.m.)	1
CONFINEMENT TIME during burn	tau(E)	(seconds)	2
BURN TIME (minimum)	tau(b)	(seconds)	200
ADDITIONAL HEATING POWER ABSORBED	P(h)	(MW)	50