**High Beta Predemonstration Fusion
Devices (Philosophy, Characteristics
and R&D Requirements)**

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FUSION ENERGY DIVISION

HIGH BETA PREDEMONSTRATION FUSION DEVICES
(PHILOSOPHY, CHARACTERISTICS AND R&D REQUIREMENTS)

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ACKNOWLEDGMENT

The concepts in this report summarize research and design efforts from many of the Departments and Programs in the Fusion Energy Division of the Oak Ridge National Laboratory. The work of personnel in these areas is focussed toward the development of an economically viable fusion reactor plant. Results have been used here to summarize the basis for a predemonstration fusion device which leads to such a reactor. Detailed aspects of the work will be published in the future by the individual contributors.

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ABSTRACT

A predemonstration fusion device (PDFD) must provide the physics and technology information necessary to warrant design, construction, and operation of succeeding fusion reactors. This report summarizes in three sections the bases for developing a PDFD.

In the first section the philosophy and physics bases which will result in an economically viable fusion reactor are described. The characteristics and operating parameters of a PDFD which will provide a working demonstration of the concepts are given in the second section. In the final section the major research and development activities necessary to support construction and operation of the PDFD are outlined.

This report is a compilation of the work presented by the authors at the International School of Fusion Reactor Technology, Erice (Sicily), Italy, September 20 - October 2, 1976.

PHILOSOPHY AND PHYSICS OF PREDEMONSTRATION FUSION DEVICES*

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ABSTRACT

A PDFD will operate in the 1980's and must provide the plasma and plasma support technology information necessary to warrant design, construction, and operation of succeeding experimental power reactors and then the demonstration plant. The PDFD must be prototypical of economic fusion devices to justify its cost. Therefore, development of the fusion core will be the focus of the PDFD. The physics performance, power production objectives, and characteristics of the PDFD, and their relationship to the research and development needs to achieve them are outlined. The design criteria for a PDFD which satisfied these constraints will be established.

During the past seven years a number of tokamak fusion power reactor designs have been undertaken with the goal of identifying major technological obstacles to the realization of practical fusion power [1]. In addition to these long range studies a number of shorter range projects have attempted to identify the technological requirements for the construction of an experimental power reactor which would demonstrate the basic technology necessary to produce net fusion power [2,3,4]. These latter studies have proved useful in defining relative priorities within the ongoing tokamak R&D program. However,

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the devices themselves have proved to be large complex machines with a high cost benefit ratio. In fact, the cost benefit ratio is so high and the uncertainties in the present designs so large that one is forced to conclude that the EPR's as presently conceived are not the next logical step in the fusion program.

The original objective of the EPR program was to advance the science and technology required for commercial fusion power by providing on a timely basis: (1) system operating experience and testing of components and subsystems for a larger demonstration plant to follow; (2) an intermediate focal point for research and development programs; and (3) large scale testing of plasma physics scaling, including deuterium, tritium, and alpha particle effects. The attempt to satisfy all of these objectives in one device while at the same time generating significant amounts of electrical power resulted in the complex designs referred to above. The dilemma in defining the characteristics of the next logical step in the fusion program stems from the desire to eliminate as much of this complexity as possible, while at the same time providing for a major step forward in the fusion program.

If the US fusion program is to achieve its goal of a demonstration reactor by the year 2000, a number of critical scientific and technological problems must be addressed in the mid-1980's. Some of these questions will be addressed in the TFTR device, but many others will require the construction of an additional facility which we shall call a predemonstration fusion device. We shall attempt to define the nature of this device in this paper.

The primary objective of the TFTR experiment is the attainment of $Q \approx 1$. Calculations have indicated that this objective can be reached in a device with a plasma radius on the order of 90 cm. The next step in the fusion program should be a reasonable extrapolation in performance and size beyond TFTR. On the basis of available scaling laws, to be discussed below, it follows that $Q = 5$ operation can be attained with a reasonable step in size beyond TFTR. It can also be shown that with a Q of approximately 5, ignition can be achieved. Thus, it is reasonable that the PDFD should operate with a near ignition plasma. Operation of such a plasma in hydrogen would require a supplemental heating power of over 200 MW. This prohibitive requirement can be reduced to 50-75 MW when D-T is used and advantage is taken of alpha heating. Therefore, real ignition of the PDFD plasma seems necessary.

An ignition experiment will require sufficient pulse time to investigate the consequences of ignition. Again, simple considerations of fusion reactor rates lead one to conclude that the machine should have a pulse time on the order of tens of seconds. Designs of magnets for $Q = 1$ devices [5,6] indicate that on a purely economic basis superconducting coils would seem to be an essential part of the device. Long pulse operation also implies efficient control of impurity build-up if ignition is to be sustained. Furthermore, since the ignition implies high temperatures and densities, the device must be capable of withstanding large pulses of fusion power. Thus a logical extrapolation beyond the performance of TFTR leads us to a device possessing most of the characteristics of the EPR. The device need not possess the ability to breed tritium, recover the neutron energy, operate with

a high duty cycle, or convert heat to electricity. All of these functions are peripheral to operation of the fusion reactor core and can be developed separately from this core. Thus it seems that a PDFD, which is the minimum significant extrapolation beyond TFTR, can be identified by the characteristics of a fusion reactor core.

Since the PDFD is a fusion reactor core, its characteristics should be compatible with the generation of economic fusion power, if tokamaks are to qualify as power systems. In order to define the most relevant version of the PDFD, it would be well to consider some of the factors which determine the economics of fusion power. The first point to be made is that, unlike the fission process in which energy is generated and recovered in the same physical region, fusion energy is generated in a plasma and recovered in an external blanket. Since in building a reactor one pays for materials and construction, not the vacuum region where the plasma resides, the costs associated with fusion systems are those associated with the external structure surrounding the plasma. Let us consider a general idealized tokamak power system. To simplify calculations, the elliptical system shown in Fig. 1 is assumed to include the general features of noncircular tokamaks. It is easy to show that the average power density in the blanket region is

$$\frac{P_B}{V_B} = \frac{2P_W}{\Delta} \sqrt{\frac{1 + \sigma^2}{2}} \left\{ \frac{1}{1 + \sigma + \Delta/a} \right\} \quad (1)$$

where the elongation, $\sigma = b/a$, the structure thickness, $\Delta = \Delta_S + \Delta_C$, includes the coil thickness, and P_W is the wall loading. For a fixed

ORNL-DWG 76-15421

Fig. 1. The geometry of an idealized tokamak power reactor is shown. The elliptic shape is chosen to include the effects of elongation on reactor economics.

wall loading, plasma elongation increases the structure power density by increasing the surface area; however, it also decreases the power density by increasing the volume of structural material. The thickness of the coil, Δ_C , is determined by the strength of the magnetic field. It is easy to show that

$$\Delta_C = a\sigma \left\{ \frac{B_M^2}{2\mu_0 S} \right\} \left\{ 1 - \frac{(1 + \Delta_s/a)}{A} \right\}^2 \quad (2)$$

where B_M is the maximum value of the toroidal field on the coil as indicated in Fig. 1, S is the stress in the coil, and $A = R/a$. For equal stress in the toroidal field coil, elongation is accompanied by a thickening of the coil and consequently a lowering of the average power density in the structure.

Since the capital cost of constructing the fusion power reactor is proportional to the volume of material used, the unit cost of fusion power is simply proportional to the inverse of the average power density in the reactor structure. Thus

$$C_P \propto \frac{V_B}{P_B} = \frac{\Delta}{2P_W} \sqrt{\frac{2}{1 + \sigma^2}} \left\{ 1 + \sigma + \frac{\Delta}{a} \right\} \quad (3)$$

It is clear that in order to minimize the unit cost of fusion power, one must operate with the highest possible wall loading, P_W , and the thinnest possible structure, Δ . In view of the above, any economic

reactor will operate with the highest practicable P_W . The practical limit on P_W is defined by the material damage of the wall material combined with the detailed economics of wall replacement. It is generally accepted that wall replacement times of less than five years would be economically unacceptable.

The scaling of unit power cost is independent of plasma performance if the plasma is capable of loading the wall to the material limit. From the plasma side, the wall loading depends on the plasma power density. This is conventionally written as $P_V \propto \beta^2 B^4 (\langle \sigma v \rangle_F / T)^{E_F}$ where $\beta = 4\mu_0 \bar{nT} / B^2$. Many reactor studies point out the power density advantage to be gained from high beta and magnetic field operation so that the wall loading, $P_W = V P_V / A_W$, can be increased.

Elongation of the system has an advantageous effect on the unit power cost because the increase in surface area admitting power to this structure outweighs the increased volume of the blanket structure. However, the increased coil thickness necessary to keep the coil stresses at a constant level adds to the volume of the structure. In terms of unit power cost, elongation of the plasma would be more advantageous if it were necessary to increase the plasma power output in order to raise the wall loading, P_W , to the limit imposed by the materials used in the construction of the first wall.

High magnetic fields affect the unit power cost in a deleterious way through Eq. 2 since the increased structure necessary to contain higher magnetic fields increases the structural volume. High magnetic fields would only be advantageous if they were necessary to raise the

level of wall loading to the material limit. Thus in terms of unit power cost, the ideal fusion reactor would be moderately elongated and would operate with the lowest possible magnetic field.

Aside from minimizing unit power cost, it is also desirable to minimize the total cost of a fusion reactor system. The total cost, C , is obtained by multiplying the unit cost by the total power output of the reactor.

$$C \propto A \Delta a^2 \left\{ 1 + \sigma + \frac{\Delta}{a} \right\} \quad (4)$$

Equation 4 shows clearly that to minimize the total cost of a fusion system, one should choose the lowest possible aspect ratio, the smallest plasma radius, and the smallest elongation consistent with reactor operation. There are lower limits on each of these parameters.

One of the difficulties with the EPR designs is the fact that they all employ low aspect ratio construction. This creates problems with respect to the assembly of the reactor and its remote maintenance [2]. Thus, for ease of remote maintenance, there is a practical lower limit of $A \sim 4$.

The minimum size of the plasma radius is determined by the condition that the plasma, $n\tau$, satisfies the ignition condition. The plasma $n\tau$ is determined by the scaling law which applies to the reactor regime. This scaling law is typically a function of magnetic field, elongation, impurity content, and aspect ratio. Consequently, the precise dependence of the cost of a fusion reactor on these parameters depends on the operative scaling law determining the plasma radius.

There are presently two scaling laws which might be thought to apply to reactor operation. The first is the scaling law derived [7]

from the theory of trapped ion modes (TIM). When this theory is modified to apply to our elliptical plasma model, the plasma $n\tau$ becomes

$$(n\tau)_{\text{TIM}} = 3.1 \times 10^{13} a^4 \left(\frac{B_T}{5} \right)^6 \beta^2 Z \left(\frac{A}{3} \right)^{5/2} \frac{1}{T_e^{11/2}} \left(\frac{1 + \sigma^2}{2} \right) \quad (5)^*$$

The linear instability theory on which Eq. 5 is based should apply if the plasma collisionality is sufficiently low. Collisionality is conveniently defined as the ratio of the effective plasma particle collision frequency to the so-called bounce frequency, or the frequency with which particles complete their orbits in tokamak magnetic fields. This is measured by the collisionality parameter, $\nu^* = 3.3 \times 10^{-18} q(A/3)^{3/2} (an/T^2)$. Thus reactors can have varying degrees of collisionality depending on their aspect ratio, plasma size, density, and temperature. Figure 2 shows the range of collisionality expected for the EPR and some other existing or planned experiments.

Fig. 2. The collisionality of some existing, planned and contemplated experiments are shown for comparison. ORMAK-Upgrade with 4 MW neutral heating, to be operational in 1978, should explore the range of collisionality appropriate to the devices operating in the 1980 time period.

* The units in the practical equations in this paper are centimeters, teslas, and kiloelectron volts.

For each device a range of ν^* is shown since the collisionality is a function of radius in the plasma, reaching its minimum value at about half the plasma radius. The trapped particle modes are expected to appear clearly below $\nu^* \sim 0.1$. Thus they may not have been observed in existing experiments and will only be found in the new generation of tokamaks represented in the figure by ORMAK-Upgrade and TFTR. If instabilities do occur for collisionalities less than 0.1, they should appear in reactor plasmas. However, their effect on the transport is still not well understood and Eq. 5 represents only an approximate estimate of this effect. It will remain for experiments such as PLT, ORMAK-Upgrade, and TFTR to verify the effects of these particular modes as well as their existence.

Present day tokamak experiments obey an empirical scaling law [8], $n\tau \propto n^2 a^2$. Plasmas obeying this scaling law have collisionalities ranging from a ν^* of 0.5 to 40.0. Reactors will operate in the range of $\nu^* \approx 0.001$ to .1. Consequently, there is a gap between the lowest collisionality observed in present day experiments and the highest collisionality which we can reasonably expect for a fusion reactor. If this gap in collisionality could be kept small by operating reactors at higher collisionality, there might be some hope [9] that the empirical scaling law observed on present day experiments might describe the behavior of fusion reactor plasmas. However, it is clear from Fig. 2 that it remains for machines like ORMAK-Upgrade to bridge this gap.

Operating reactor plasmas with collisionality comparable to present experiments in order to preserve the empirical scaling

requires high density or equivalently high beta operation. Written in terms of beta, the empirical scaling law becomes

$$(n\tau)_{\text{emp}} = 2.5 \times 10^{15} a^2 \left(\frac{B}{5} \right)^4 \frac{\beta^2}{T^2} \quad (6)$$

showing a strong dependence on magnetic field strength [10] and beta.

We can use Eqs. 5 and 6 to determine the size of an ignition experiment for collisionless ($\nu^* < .1$) or collisional ($\nu^* > .1$) reactor operation. From Eq. 5

$$a_{\text{TIM}}^2 \propto \frac{1}{B^3 \beta} \sqrt{\frac{2(n\tau)_{\text{IGN}}}{1 + \sigma^2}} \quad (7)$$

and from Eq. 6

$$a_{\text{EMP}}^2 \propto \frac{\sqrt{(n\tau)_{\text{IGN}}}}{B^4 \beta^2} \quad (8)$$

In both cases, beta can be seen to be advantageous for lowering cost. The effect of elongation appears only in the case of the collisionless reactor and has a weaker effect on the plasma size. Returning to Eq. 4 we see that the total cost or size of the system is roughly independent of elongation for a collisionless reactor and rises with σ for a collisional reactor.

It is known that the minimum ignition $(n\tau)_{\text{IGN}}$ is a sensitive function of the plasma impurity content. The rapid increase of $(n\tau)_{\text{IGN}}$ with effective plasma charge Z and the direct dependence of a^2 on this increase dictates an extremely effective impurity control method in an economic reactor.

The advantage gained from high magnetic field is real, but limited. Extremely high magnetic fields can be defined as those

which cause the coil structure to dominate the blanket and shield structure. Equations 3, 4, 7, and 8 show that although the total cost can be minimized by extremely high field operation, the specific power cost will rise as B^2 for a fixed wall loading.

Let us now turn to an examination of an application of these considerations to the specification of a PDFD which satisfies the constraint of relevance to economic fusion power as well as the advancement of fusion science and technology. A consideration of the EPR designs reveals that these general rules do reflect the economic trends found in those studies.

Table I provides a comparison of the three EPR designs and a candidate for the PDFD. This candidate, the TNS, has been developed by ORNL and Westinghouse to satisfy both the research and development needs of the US fusion program in the 1980's and the need to have a demonstration of the essential components of an economically attractive fusion reactor core. The device was sized using either empirical or trapped particle mode scaling as used in the ANL and ORNL EPR studies. High beta operation allows ignition for either scaling and provides high wall loading. The aspect ratio was enlarged relative to the EPR designs to allow for easier remote maintenance. Moderate D elongation was chosen to optimize trapped particle mode scaling and unit cost. This equilibrium also has a natural divertor property for impurity control. The device shown in Fig. 3 will be described further in the following paper.

Because of the present uncertainties associated with many features of the science and technology of TNS, which will be continually

TABLE I

	EPR			TNS
	ORNL*	ANL [†]	GA [‡]	ORNL/ (W)
Major Radius (M)	6.75	6.25	4.5	5
Minor Radius (M)	2.25	2.1	1.1	1.25
Structure Δ (M)	2.25	2.25	2.25	2.25
Elongation	1	1	3	1.6
Maximum Toroidal Field (T)	11	8.0	7.9	8.0
Plasma Toroidal Field (T)	5	3.5	3.9	4.1
Plasma Beta (%)	3	4.8	10	14
Wall Loading (MW/M ²)	.55	.35	.85	3.7
Thermal Power (MW)	357	261	344	1510
Fusion Core Cost (M\$)	249	249	218	218 [#]
Unit Power Cost [§] (\$/kW)	695	954	634	144 [#]

* ORNL EPR study, ORNL/TM-5572-77, M. Roberts, Ed.

[†] ANL/CTR-76-3, August 1976, W. Stacey, Ed. The ANL-EPR reference design includes a 10 T maximum field. An alternate case which includes moderate field has been shown here for comparison.

[‡] GA-A14043, July 1976, C. Baker.

[§] Cost of fusion core divided by thermal power.

^{||} For purpose of comparison we have used the cost of similar size ANL EPR.

[#] For purpose of comparison we have used the cost of similar size GA EPR rather than an independent cost estimate.

Fig. 3. The cross section of the initial TNS design is shown to illustrate the attempt at size reduction and economical structure utilization in a high performance tokamak reactor experiment.

resolved over the next four years, the design will undoubtedly evolve from its present form in that same time period. However, the reference parameters indicated illustrate the improvement over the performance of the EPR's which can be realized in tokamak systems by a focusing on the key design elements as revealed in Eqs. 3 and 4.

It should be emphasized here that the absolute dollar estimates quoted in the table are not to be taken seriously at this time. We are concerned only with the trends in total and specific power cost as a result of the selection of different design choices. For the sake of consistency, we have estimated the ORNL EPR cost and the TNS cost from the other two EPR's which should have relatively consistent costing since they were performed concurrently. We have also used the nominal structure thickness of 2.5 M for all the devices to simplify comparison. The model in Fig. 1 assumes a compact structure. Reference to a typical EPR cross section (Fig. 4) shows that the structure is greatly extended. By comparison, the TNS design (Fig. 3) attempts to economize on the volume of material to better accord with the prescriptions of Eq. 2 and 3.

As seen from Eq. 3 the dominant factor in determining the specific cost is the wall loading. The General Atomic EPR has the highest wall loading and the lowest specific cost. However, according to Eq. 3 the cost advantage would be 2.75 over the ANL design and 1.7 over the ORNL design. In fact, the actual advantage is 1.4 over the ANL design and the ORNL EPR achieves comparable cost. The reason is unclear. Compared to the ANL EPR, the GA design has larger β and B which should contribute to higher power density in the plasma at the same operating

Fig. 4. The cross section of the best performing EPR design, the GA EPR is shown. This design possesses many of the desirable characteristics of a PDFD and is a candidate for this function.

temperature. Thus, the average wall loading in the GA EPR,

$\bar{P}_W \propto \beta_B^2 (V/A_W)$, should be about five times larger than in the

ANL EPR. However, the wall loading advantage quoted is only 2.4. By

comparison, the wall loadings of the ORNL EPR is larger than the ANL

EPR by the amount expected on the basis of the $\beta^2 B^4$ scaling. Thus, although Eq. 3 indicates a reduction in unit power cost with elongation in the elliptical model, it does not appear that this advantage is fully realized in the GA EPR. Therefore, the nominal performance of the GA EPR does not differ significantly from the others shown.

As mentioned above, the use of higher field on the ORNL EPR compensates for the low beta plasma and high wall loading is achieved. However, the field increase is modest and does not result in significant increased structure because of a novel design utilizing force cooled Nb_3Sn superconductor within stainless steel conduits [2] which support the increased forces. This results in roughly the same coil size as the ANL and GA designs. Larger fields would require increased structure and possibly new types of superconductor. Thus the range for further performance improvement through magnetic field is limited.

The performance of the TNS as indicated in Table 2 is greatly improved over all of the EPR designs. This improvement results from the operation of the device at a high beta and a moderate plasma elongation, in a D configuration which provides for an efficient utilization of available volume within the coils. This results in the wall loading in the TNS design being much larger than that in any of the EPR designs and the power output being five times greater than these devices.

The details of the TNS design will be described in the following paper [11]. However, it is important here to consider the reason for assuming that high beta can be achieved in such a device and that the resulting high wall loadings have practical application for fusion

TABLE II

REQUIREMENT FOR PDFD	CHARACTERISTICS OF TNS
High wall load	3.7 MW/M^2
High beta	14 %
Z control	Natural divertor
Moderate elongation	1.6
Moderate magnetic field	4.1 T
Minimum structure Δ	2.5 M
High performance	1500 MW _{TH}
Moderate size	a) Coils half EPR size b) $a_{\text{TNS}}/a_{\text{TFTR}} = 1.4$

power systems. The possibility of high beta operation is based on the flux conserving tokamak concept [12,13]. This concept utilizes the fact that heating of large reactor plasmas from their initial low beta state to the desired high beta ignition state must occur on time scales which are much faster than magnetic diffusion times. Consequently this heating will occur through a sequence of equilibria containing the same amount of magnetic flux. As a result the low beta q profile is preserved during the heating process and a unique final equilibrium is obtained. This final equilibrium has been shown to possess many interesting stability properties such as good shear and an absolute magnetic well [14]. The ORMAK-Upgrade experiment which will be operational in 1978 is aimed at verifying the existence of such equilibria. The belt pinch [15] experiment seems to possess such an equilibrium at betas of 30 to 70% and to exhibit stable

operation for 20 MHD growth times.

Operation of tokamak plasmas at high beta with efficient use of the plasma volume inevitably results in high wall loading such as shown in Table II. Equation 3 indicates that this high wall loading has economic advantages with regard to reducing the unit cost of fusion power. However, if this high wall loading results in such severe wall damage that the reactor wall must be replaced on short time intervals, this advantage would be more than offset.

Recent results on the simulation of CTR radiation damage in stainless steel [16] has shown that wall lifetimes in excess of 20 megawatt years/m² could be expected from a reasonable materials development program. Consequently, the wall loading indicated for the TNS design would result in a wall lifetime in excess of five years.

An examination of the costs of TNS components resulting from our initial design shows that the fusion core of TNS has about the same cost per unit thermal output as the fission core of the Clinch River Breeder Reactor. Thus, we have some indication that a device with TNS parameters has characteristics of an economic power system. Therefore, the TNS parameters satisfy the major requirement of a PDFD. Elements of these requirements are indicated in Table II. We conclude that the plasma size is a reasonable step beyond the TFTR and the ignition condition seems to be attainable in a reactor core which is directly on the path leading to an economic fusion system. The key assumptions about the attainability of the beta and confinement necessary for the performance indicated must be verified by experiment over the next few years; however, the logic for TNS is clear. Results and plans

from the ongoing fusion program lead naturally to a DT fueled ignition device following the TFTR. The objective of ignition in turn sets definitive criteria for numerous of the key physics and technological systems. With these criteria set, the goals of the research and development program necessary to insure that the criteria can be met on a time scale which will support TNS construction can be established. In the following paper the criteria for the key physics and technological systems will be established. In the last paper of this series, the research and development programs for some of the necessary technology supportive of the TNS design will be described.

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CHARACTERISTICS OF A PREDEMONSTRATION
FUSION DEVICE*

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ABSTRACT

Low beta reactor system studies have shown that criteria necessitated by the physics lead to difficult engineering problems. The low power density produced in these systems also results in large reactors and makes the question of economic viability a serious one. It is clear that if higher pressure plasmas could be confined and operated, then many of these problems could be alleviated. The incentives for high plasma density as a means of achieving higher pressures are discussed. The possible methods of achieving high density are outlined with particular attention to the high beta approaches. The requirements which are imposed on the reactor by using the flux-conserving tokamak concept are described.

The characteristics of a predemonstration fusion device which can test the necessary aspects of such a system are developed. The plasma size required to attain ignition is found to be relatively small provided other criteria can be satisfied. These criteria are described and the technology developments and operating procedures required by them are given. The dynamic behavior and parameters of the reference system during the operating cycle are also outlined.

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[†]Exxon Nuclear Co., Inc.

I. INTRODUCTION

Several of the features which are desirable for a tokamak reactor are discussed in this lecture. The considerations which lead to the parameters of a predemonstration fusion device (PDFD) which can test the necessary aspects of such a system are then developed.

The power density produced in a reactor plasma depends on the density and temperature of the fuel ions burned. Tokamak reactors will have to operate under ignition conditions to attain economic viability. This requirement imposes parametric constraints on the containment system since ignition depends on density, temperature, and energy confinement time. The energy confinement time in turn depends on other variables, but at present this dependence is not known. There are indications, however, that confinement time will scale favorably with an increase in density. This result is suggested both by experiments and by theoretical analyses. Thus from several points of view there are incentives to operate the reactor plasma at high density.

Several approaches for achieving high density are envisioned. A method which requires a moderate magnetic field (or high beta) is desirable for economic reasons. It may be attainable if, as expected, flux is conserved during heating and if the resulting magnetic configuration can be maintained during the burn phase. This concept is that of a flux-conserving tokamak.⁽¹⁾ Noncircular plasmas may offer a means to high density.⁽²⁾ Finally, high magnetic fields (or low beta) may be necessary.⁽³⁾ In this paper a reference system which includes a moderate magnetic field, a moderately elongated plasma, and high beta will be developed. Since the operating temperature of the plasma will be in the range of reactor

interest, i.e., $\sim 10 - 15$ keV, high beta implies high density and vice versa in this discussion.

High beta (density) imposes constraints on the design and operation of the system. For example, stable high beta plasma equilibria must be created and maintained and a compatible poloidal field system must be developed. High power density means that the first wall will be subjected to high neutron fluxes ($> 1 \text{ MW/m}^2$) as well as a high thermal flux. The breakdown and current rise phases of start-up may require higher voltages when increased filling pressures are used. Supplemental heating using neutral beams may be difficult because of the beam energies required to achieve adequate beam particle penetration. Finally, impurity and burn product effects in a dense plasma must be examined. In some cases technological developments may be needed to meet design criteria, while in other cases it may be possible to develop operating procedures which provide the desired solutions. Specific features of this type are discussed below.

II. SYSTEM MODELING

There are many ways to simulate the detailed energy balance processes in a tokamak. In the present work the same model used previously in the ORNL Experimental Power Reactor Design Study⁽⁴⁾ has been adopted, and the impact of high density has been assessed. The time dependent model includes pseudoclassical, neoclassical, and dissipative trapped particle mode scaling laws as well as radiation effects and cold fueling. Conduction and convection times are functions of plasma collisionality.

Figure 1 shows the plasma radius required by this model for ignition as a function of temperature; the axis magnetic field for three cases is

shown, where the density is $2 \times 10^{20} \text{ m}^{-3}$ and the plasma elongation (height to width ratio) is 1.6. The minimum size for ignition occurs at about $T = 13 \text{ keV}$; below this temperature, the $N\tau$ required for ignition increases sharply and an increase in size is necessary to attain it. Above 13 keV , trapped ion mode transport ($N\tau \propto \beta^2 T^{-11/2}$) dominates the scaling and an increase in plasma size again is required. The neutron wall loadings associated with various operating temperatures are also given. Note that at $N = 2 \times 10^{20} \text{ m}^{-3}$, $T = 13 \text{ keV}$, and $B_T = 4.5 \text{ T}$, the wall loading exceeds 3 MW/m^2 , and the plasma radius necessary for ignition is 1.25 m . With a field strength of 6.5 T , the size required is reduced to $\sim 1 \text{ m}$ and the wall loading is $\sim 2.5 \text{ MW/m}^2$.

FIGURE 1. Plasma radius for ignition (A_{ig}) and corresponding neutron wall loading as a function of temperature.

Figure 2 shows the plasma radius required for ignition as a function of density. In these cases $T = 13 \text{ keV}$. At lower densities the collisionality of the plasma is such that transport is dominated by the trapped ion mode which requires an increased plasma size. As density is

increased the size required for ignition decreases, since the impact of the trapped ion mode becomes less pronounced.

FIGURE 2. Plasma radius for ignition and β as a function of density.

There is no quantitative answer to the question of what maximum density is acceptable. Also, theoretical work⁽⁵⁾ suggests that $\beta \sim 20\%$ or higher may be attainable. For this analysis an operating density of about $2 \times 10^{20} \text{ m}^{-3}$ is assumed, which results in a $\beta \sim 0.14$ at 13 keV, a wall loading less than 4 MW/m^2 , and an ignition size of 1.25 m. Table 1 summarizes the reference system parameters computed from a time dependent, self-consistent simulation model.

TABLE 1. Reference Parameters

Major radius, R_0 (m)	5
Minor radius, a (m)	1.25
Elongation, b/a (-)	1.6
Toroidal field at winding, B_{max} (T)	8
Toroidal field on axis, B_T (T)	4.32

TABLE 1 (cont'd)

Plasma current, I(MA)	4
Electron density, $N_e(m^{-3})$	2.2×10^{20}
Ion density, $N_i(m^{-3})$	2.0×10^{20}
Ion temperature, $T_i(keV)$	13.5
Energy confinement time, $\tau_E(s)$	1.2
Toroidal beta, $\beta(-)$	0.14
Average neutron wall loading, $L_w(MW/m^2)$	3.7
Thermal fusion power, $P_{th}(MW)$	1510

III. PARAMETER SENSITIVITIES AND SYSTEM DYNAMICS

The operating parameters shown in Table 1 represent steady state conditions. Since the model used includes the trapped ion mode where $N\tau_E \propto \beta^2 T^{-11/2}$, the temperature increases to the point where the energy loss rate equals the heating rate due to thermonuclear alpha particles and a steady state is achieved. Figure 3 shows $N\tau_E$ vs T_e for three different values of magnetic field. The system size is that given in the table and $Z_{EFF} = 1$ is assumed. As the temperature increases, $N\tau_E$ is dominated first by pseudoclassical, then by trapped electron mode, and finally by trapped ion mode diffusion losses.⁽⁶⁾ $N\tau_E$ required for ignition is also shown. For example, when $B_T = 4.3$ T, ignition is attained at 10 keV. The temperature continues to rise after ignition since $P(\alpha)$ exceeds $P(\text{loss})$ and steady state conditions are reached at ~ 15 keV, i.e., $N\tau(\text{operation}) = N\tau(\text{ignition})$. There is a minimum field for this size device which can provide ignition; it is about 4.0 T and results in an $N\tau(\text{operation})$ curve which is tangent to the $N\tau(\text{ignition})$ curve. Note that this would occur

at about 13 keV and $2 \times 10^{20} \text{ m}^{-3}$, which in turn implies a fusion power output of 1200 MW. This is an important point since it means (assuming this scaling model) that the minimum power output associated with an ignited device of this size is greater than 1000 MW. A heat removal capability compatible with this power level would have to be installed in such a device.

FIGURE 3. $N\tau$ required for ignition and system $N\tau$ operating characteristics as a function of temperature and various field strengths.

The system could be operated at a higher field. In the 6.0 T case, ignition occurs at 7 keV, but the equilibrium temperature is greater than 20 keV. This is the result of improved confinement at the higher field. Under these conditions the system could achieve ignition at a density less than $2 \times 10^{20} \text{ m}^{-3}$. The power produced is proportional to N^2 so that ignition conditions at the lower density would be compatible with a power output less than 500 MW. This, of course, represents a low beta system and is less attractive economically. Alternatively, a smaller size system could be used to achieve ignition under high field conditions. Figure 1 showed

that a 1 m plasma radius may be sufficient when $B_T = 6.5$ T. In this case the power output would be 630 MW when $N = 2 \times 10^{20} \text{ m}^{-3}$ and $T_i = 12$ keV.

The ignition calculations above assumed a pure ($Z_{\text{EFF}} = 1$) plasma. When impurities are included, both the ignition requirement and the system operating characteristics change. Figure 4 shows this for the reference (Table 1) system. Iron is assumed to be the impurity. Due to enhanced radiation losses, ignition cannot be achieved above $Z_{\text{EFF}} \sim 1.5$. This establishes the impurity control criteria which will have to be met during the burn cycle of the device. Note also that in the second trapped electron mode $N\tau \propto 1/Z_{\text{EFF}}$ and in the trapped ion mode $N\tau \propto Z_{\text{EFF}}$ so that the confinement characteristics shown in Fig. 4 cross over after the transition into the trapped ion mode.

FIGURE 4. $N\tau$ required for ignition and system $N\tau$ operating characteristics as a function of temperature and various impurity (iron) levels.

The desired reactor plasma performance can be attained at $B_T = 4.3$ T, $N = 2 \times 10^{20} \text{ m}^{-3}$, and $T_i = 13$ keV provided that a plasma with a beta of about 0.15 can be stably confined. The equilibria of such plasmas are discussed in the next section.

IV. MHD EQUILIBRIUM AND STABILITY

The maximum attainable value of beta in a stable toroidal plasma is influenced by the plasma cross sectional shape,^(7,8) safety factor profile, and toroidal current profile. The usual approach to and results of MHD equilibria calculations are discussed first and compared to those of the high beta flux-conserving tokamak (FCT) equilibria.^(1,5)

Under the assumptions of large aspect ratio and nearly circular flux surfaces, it has been shown that $\beta_p = A$ is an upper limit in an axisymmetric equilibrium.⁽⁹⁻¹¹⁾ If β_p exceeds A , a poloidal field separatrix would intersect the plasma boundary. For nearly circular flux surfaces, the local values of poloidal field strength can be used to relate q , A , β_p and β as follows:

$$q = \frac{B_T}{AB_p}, \quad \beta = \left(\frac{B_p}{B_T} \right)^2, \quad \beta_p = \frac{\beta}{q^2 A^2}.$$

This then implies that beta is limited to $1/q^2 A$, which yields $\beta \sim 1 - 5\%$, using typical values for q and A .

It has been shown numerically that when these assumptions are not made, equilibria with $\beta_p \sim A^2$ can exist in an ideal, circular shell.⁽¹²⁾ These equilibria were computed in the usual way, i.e., the pressure function, $p(\psi)$, and the toroidal diamagnetism function, $F(\psi)$, were chosen arbitrarily so that reasonable, bell shaped toroidal current distributions resulted when $\beta_p \sim 1$. These conditions are consistent with those observed in present experiments.

The high beta profiles obtained in this way for $\beta_p \sim A$, typically include reversed currents at small values of R , that is, on the inside of

the plasma cross section. Also, to prevent $q < 1$ at the plasma center so that local (Mercier¹³) stability criteria are satisfied, as β_p is increased, it is necessary to increase q at the plasma boundary. This is accomplished by increasing B_T or decreasing the total plasma current. Because of this, as β_p is increased, β first increases but ultimately decreases. Figure 5 shows this behavior for an equilibria in an optimized D-shaped cross section. The maximum β (2.5 – 5.6%) depends on the shapes of the current and pressure profiles assumed, but it occurs at $\beta_p \sim 2.4$.

FIGURE 5. Local stability limits for non-FCT MHD equilibria in a tokamak with an optimum D-shaped cross section. ($A = 3$); (a) behavior of $q(\text{limiter})$ and $q(\text{center})$ and (b) maximum β as a function of β_p . Flat, rounded, and peaked refer to the pressure profile.

In the same beta optimization calculations,⁽⁴⁾ it was found that with bell shaped profiles there is a limit to increasing beta by elongating the plasma. As the elongation is increased, successively more peaked current profiles are required to satisfy local stability criteria. The increase in beta obtained from the slight noncircularity is nullified as the current profile becomes more peaked. The peaked profile enhances the importance of the plasma center where the flux surfaces remain approximately circular. Thus, in the highly elongated cases the peripheral regions of the plasma are found to contribute little to the average beta. A moderate elongation of ~ 1.6 is found to provide the maximum benefit with respect to beta within the assumptions used here.

Next the concept of flux conservation and its impact on attaining high beta equilibria will be outlined. As described above, high beta equilibrium calculations previously have used relatively arbitrary profiles for $p(\psi)$ and $F(\psi)$. When heating leads to a pressure excursion on a time scale which is short compared to the magnetic diffusion time, however, the toroidal and poloidal fluxes are frozen into the plasma. The $p(\psi)$ and $F(\psi)$ functions which describe FCT equilibria are then not arbitrary. Rather $p(\psi)$ is determined by the heating and transport processes and $F(\psi)$ by the requirement that flux be conserved. Also, since

$$q(\psi) \propto \delta\phi/\delta\psi ,$$

the q profile established at low beta will persist during the pressure excursion due to flux conservation. The beta limitations observed in Fig. 5 due to the necessity to increase $q(\text{limiter})$ to prevent $q < 1$ at the plasma center no longer occur. Numerical procedures have been developed⁽⁵⁾ to preserve $q(\psi)$ as $p(\psi)$ is increased.

A typical FCT equilibrium sequence where $q(0) = 1$, $q(a) = 4.8$, and $\beta \leq 0.18$ is shown in Fig. 6. In these equilibria, it is not necessary to decrease the total plasma current or increase B_T to maintain local stability as β_p is increased. Thus, unlike the previously calculated high beta equilibria, the FCT approach permits β to increase monotonically with β_p .

The FCT equilibrium configurations do not show reversed currents; however, the total plasma current does increase. It has been estimated analytically⁽¹⁴⁾ that the total plasma current scales as $p^{1/3}$. Using the same analytical procedure but performing the integrals numerically, it is found that the current scales as $p^{1/4}$ in the range of pressure increases of

interest.⁽¹⁵⁾ This has been observed numerically⁽⁵⁾ as well. The increased current in a high beta FCT equilibrium is produced by diamagnetic effects and does not have to be driven by the ohmic system primary. However, the poloidal field system must be designed specifically to accommodate the changes in the plasma current distribution and magnitude during the heating phase. A low impedance poloidal coil configuration which can provide automatically most of the needed current change during the FCT pulse is desirable.

FIGURE 6. Character of (a) flux surfaces, (b) mid-plane pressure profiles, and (c) current profiles as they evolve while β increases. $q(\text{limiter}) = 4.8$.

Only local stability criteria which are satisfied by $q(0) \sim 1$ have been addressed. There is experimental evidence from the belt-pinch⁽¹⁶⁾ and SPICA⁽¹⁷⁾ that high beta plasmas can operate in a stable way. There is, nonetheless, considerable effort required to resolve the nonlocal and more general stability questions associated with FCT equilibria. This work is under way.

V. POLOIDAL FIELD SYSTEM

The high beta equilibria computed in the previous section were obtained using a fixed boundary MHD code. In this model the eddy currents induced in the shell which surrounds the

plasma respond to changes in the plasma. The eddy currents provide the proper equilibrium shaping field, cancel the poloidal field which otherwise would exist outside of the shell, and have a sum equal to the plasma current but flow in the opposite direction.⁽¹⁸⁾ The shell can be replaced by a set of discrete coils in the vicinity of the plasma which produce the ideal equilibrium field pattern. When the coils are properly located they will respond in the same way as the ideal shell.

The shape of the current induced in the shell on either side of the "tip" of the D-shaped plasma remains unchanged for large changes in β and I_p . Thus, the coils on either side of the tip can be located⁽¹⁸⁾ to carry equal currents, which reduces the number of power supplies and the degree of control required. Also, since the coils must be placed a finite distance from the plasma boundary, it is found that a divertor-like pair of coils must be located near the tips of the D to reproduce the proper field pattern.⁽¹⁹⁾ Such a poloidal field system retains the features of the STATIC system^(18,20) and in addition it is compatible with the equilibrium requirements of the rapid changes of the D-shaped plasma during heating.

The divertor-like coils described above carry currents which flow in the same direction as the plasma current. The separatrix formed as a result of this can be shifted to the edge of the plasma by making modest changes in the poloidal field system. This divertor configuration is "natural" for D-shaped plasmas. The coil locations in this case are determined as described above except that the shell used is D-shaped and has a sharp tip similar to the geometry of a separatrix. The resulting coil system is shown in Fig. 7. Using these coil locations, a typical FCT equilibrium ($\bar{\beta} = 0.15$) which includes the natural divertor has been

computed using the free boundary MHD code; it also is shown in Fig. 7. The current required in a divertor coil is $1.54 I_p$ compared with $1.14 I_p$ for an equivalent equilibrium without the natural divertor feature. The current level is small compared with conventional divertor coil requirements. Since the coils in this system simultaneously maintain the equilibrium and the separatrix configuration, the location of the separatrix is fixed when the plasma is centered and the difficulty of controlling the separatrix location should be minimized.

FIGURE 7. (a) Optimized coil locations for the D-shaped FCT with natural divertor, and (b) a high β (~ 0.15), free boundary, FCT equilibrium confined by the coils.

equilibrium evolves in a way dictated by flux conservation. Ignition occurs and the temperature continues to rise until the burn phase is established ($10 \text{ s} \leq t \leq 30 \text{ s}$). Finally, the shutdown phase ($30 \text{ s} \leq t \leq 40 \text{ s}$)

Finally, the current wave forms in the coils during the pulse are of interest. The pulse is composed of start-up, ohmic heating, FCT heating, ignition excursion, burn, and shutdown phases. Start-up ($0 < t \leq 1 \text{ s}$) is defined as the time required for breakdown and for the plasma current to reach about 0.5 MA. During the ohmic heating phase ($1 \text{ s} \leq t \leq 2 \text{ s}$) the plasma current rises to its final value and a low beta discharge is established. The FCT heating phase ($2 \text{ s} \leq t \leq 6 \text{ s}$) is when rapid heating occurs and the

is entered and results in termination of the discharge. Figure 8 shows the current time histories in the plasma and the equilibrium field and ohmic coil systems which are required to maintain plasma equilibrium during the cycle. The average beta, which is also consistent with the cycle for the predemonstration device (see Table 1), is also shown as a function of time. This typical case does not include a natural divertor. Table 2 summarizes the results.

FIGURE 8. β and current time histories compatible with the operating cycle of a high density system.

TABLE 2. Parameters for PDFD Reference Cycle

$\bar{\beta}(\%)$	$\beta_{axis}(\%)$	$\bar{\beta}_p(\%)$	$I_p, 7 \times I_{EF-D}(\text{MA})$	$I_{EF-0}(\text{MA})$	$I_{EF-I}(\text{MA})$
0.8	2.7	0.5	4.1	2.43	1.94
1.8	6.4	1.1	4.2	2.68	1.85
5.7	17.3	2.6	4.8	3.60	1.54
11.7	33.6	4.0	5.7	5.02	1.06
14.7	41.2	4.4	6.1	5.75	0.82

The conclusion of these analyses is that the high density desired may well be accessible at moderate magnetic field strengths (high beta) by

operating in the flux-conserving mode. The poloidal field system required to permit such operation is reasonable and, in fact, may facilitate impurity control as a result of its natural divertor configuration.

VI. PLASMA START-UP

The gas breakdown and current rise phase in a large tokamak will establish certain design criteria for the system. The parameters of special interest are voltage and current wave forms, maximum filling pressure and plasma density, volt-second consumption, tolerable impurity level and line radiation, wall interactions, and runaway electron flux. Innovative start-up schemes may be required for large tokamaks ($R_0 \approx 5$ meters) because high density operation may need extremely large voltages to ionize the gas, high runaway electron fluxes may occur, and large tokamaks are very sensitive to impurities.

Previous research in this area⁽²¹⁻²³⁾ has been concerned with both gas discharge physics, and atomic and molecular physical processes. The model used in this analysis⁽⁴⁾ includes these processes as well as wall interactions, runaway electrons, and time dependent impurities. These are incorporated into a set of coupled rate equations. The equations are used to compute the time dependent average values of the density and temperature of electrons, H_1^+ , H_2^+ , H_1^0 , H_2^0 , the non-equilibrium ionization states of oxygen, plasma current from a circuit equation, and the runaway electron flux.

The model has been applied to the PDFD reference design (cf. Table 1). For an applied voltage of 50V/turn, which is typical of present tokamaks, and an initial H_2^0 density of $0.2 \times 10^{20} \text{ m}^{-3}$ which is five times

less than is desirable for high density operation, the density, temperature, and plasma current stagnate at low levels. This indicates that much higher voltages are required in larger tokamaks to achieve start-up, even at relatively low filling pressures. With $N_{20} = 1.0 \times 10^{20} \text{ m}^{-3}$ and 450 V/turn the plasma is fully ionized after 14 msec. However, even a small amount of impurities would cause excessive line radiation at these very high densities; thus the plasma must be kept extremely pure.

The ionization and plasma current initiation in large tokamaks requires very large applied voltages and low impurity levels. Runaway electron creation may be an important effect. Research is required on small radius and low density start-up methods.

VII. PLASMA HEATING

The fundamental aspects of neutral beam heating during the plasma start-up are discussed in this section. The supplemental heating power level necessary to achieve ignition in the reference system in about 6 s is found to be 75 MW. This calculation is based on energy balance considerations using the final density ($\sim 2 \times 10^{20} \text{ m}^{-3}$) and does not include the problems of beam penetration into a dense plasma. High density and high impurity content make penetration into a large plasma increasingly difficult. Beam focusing, perpendicular injection, and high beam energies may be necessary to overcome the penetration problem. Small radius start-up and low density start-up may provide other means to solve the problem. Here the concept of low density start-up, which has previously been suggested,^(24,25) is outlined.

Beam deposition can be characterized by the function $H(r)$ which was first introduced by Rome et al.⁽²⁶⁾ $H(r)$ is a geometric measure of the deposited fast ion number density profile which results from injection. It is desirable to have the beam deposition peaked in the central region of the plasma for maximum heating efficiency. In these calculations a deuteron beam, 30 cm in diameter, is assumed to be injected into a $Z_{\text{EFF}} = 1$ plasma.

Figure 9 shows the deposition profiles for several different beam energies. In these cases injection is nearly perpendicular (16° from perpendicular to prevent ripple trapping of the fast ions), and the density profile is assumed to be $N = 3 \times 10^{20} (1 - (r/a)^2)^{-3}$. Adequate penetration can be achieved in this way if a beam energy of ~ 250 keV can be made available. Impurity effects dictate an even higher energy.

FIGURE 9. Neutral injection fast ion deposition profiles for various injection energies (perpendicular injection).

High density and temperature are required at the ignition point where the beams are no longer needed to sustain the plasma energetically. It may be possible to establish the temperature conditions at low density, increase the fueling rate to build up the necessary density while continuing injection, and tolerate the less favorable injection profiles which will occur near the end of the heating phase. Figure 10 shows $H(r)$ for 150 keV tangential injection into average densities of 7.5, 10,

and $20 \times 10^{19} \text{ m}^{-3}$. The injection deposition profile is favorably peaked when $N = 7.5 \times 10^{19} \text{ cm}^{-3}$, is relatively flat for $N = 1 \times 10^{20} \text{ m}^{-3}$, and becomes peaked at $r/a = 0.75$ when $N = 2 \times 10^{20} \text{ m}^{-3}$. Therefore, heating efficiency at this beam energy will decrease when $N > 1 \times 10^{20} \text{ m}^{-3}$, but if this part of the heating cycle occurs over a time on the order of an energy confinement time, the less efficient heating process may well be tolerable.

FIGURE 10. Neutral injection fast ion deposition profiles for various plasma densities and a beam energy of 150 keV (tangential injection).

Figures 11 and 12 show the time dependent density and power histories during heating. Initially $N = 7.5 \times 10^{19} \text{ m}^{-3}$, $T = 100 \text{ eV}$, and $P_{\text{beam}} = 75 \text{ MW}$. At $t = 2 \text{ s}$, $T_i = 12 \text{ keV}$, and the fueling rate is increased to begin the density buildup phase. The plasma ignites and beam injection is terminated after $t = 4.4 \text{ s}$. Density buildup occurs over a period of 2.4 s , and the average confinement time during this phase is $\tau_E \approx 1.5 \text{ s}$. Energetically this scenario is feasible since 75 MW is sufficient to heat the plasma to ignition at full density.

In fact, more detailed studies of the low density approach using a coupled beam deposition-spatially dependent transport calculation are necessary to optimize the start-up procedure. Also, as the flux surfaces shift when high beta is attained, beam penetration should improve.

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FIGURE 11. Plasma density buildup during low density start-up procedure.

FIGURE 12. Fusion power output during low density start-up and burn cycle.

VIII. CONCLUSIONS

There are incentives for high density operation in a tokamak reactor. Several features of a predemonstration fusion device (PDFD) which will demonstrate the feasibility of attaining such operation in flux-conserving tokamaks were described. Other key aspects such as fueling, impurity control, and the effects of magnetic field ripple were not included here.

Suitable high beta equilibria were found and a poloidal field system design compatible with the equilibria was developed. The natural divertor configuration inherent in the system led to the inclusion of a poloidal divertor in the PDFD. The start-up analyses showed that further innovations are required unless very large voltages are applied to achieve breakdown and to establish the plasma current. Neutral beam energies of ~ 300 keV were found to be necessary when the plasma is heated under full bore and full density conditions. A low density start-up procedure was outlined which, if feasible, suggests that injectors with an energy capability similar to that being developed for TFTR may be adequate to achieve ignition in the PDFD. Successful operation of a PDFD of this type means that fusion reactors can be designed with high power density and high beta. These features lead to reduced size and enhance the possibility of attaining economic viability in the tokamak.

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PREDEMONSTRATION FUSION DEVICES
RESEARCH AND DEVELOPMENT NEEDS*

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ABSTRACT

In this concluding lecture, the criteria and reference design information for the PDFD will be examined with respect to the presently planned R&D programs, and the major research and development needs will be described. Emphasis will remain on the fusion core rather than the nuclear shell. Scaling, heating, fueling and other needs will be outlined. In the area of technology, magnet systems, and tritium handling compatible with PDFD operation will be discussed. Presently planned research and development programs will provide the information needed to support the TNS version of the PDFD by 1980.

Since the design of the PDFD must be based on our knowledge of present experiments and theoretical predictions, it is not surprising that the tokamak experimental program planned on the basis of this same knowledge and theory should be addressing itself to the key uncertainties in the physical requirements of the PDFD plasma. Table I outlines the key physical problems and the experiments which will provide answers to these questions. These experiments utilize neutral beams for plasma heating and with the exception of TFTR are hydrogen fueled by gas recycle. The exception, TFTR, has a DT handling system which

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TABLE I

US EXPERIMENTS ADDRESSING KEY PDFD QUESTIONS

Problem	Experiment	Start
Impurity evolution and control	ISX	1977
	PDX	1978
Scaling	PLT	1976
	PLT + beams	1977
	ORMAK-Upgrade	1978
	Doublet-III	1978
	TFTR	1981
High beta equilibrium and stability	ORMAK-Upgrade	1978
Non-circular plasma operation	Doublet-III	1978
	PDX	1978
Neutral heating	PLT	1977
	ORMAK-Upgrade	1978
High field effects	Alcator	1978
D-T operation	TFTR	1981

is not designed to be extrapolated to a fusion reactor system.

None of these devices are long pulse or operate with superconducting magnet systems. Since PDFD will have to address the ignition of a DT plasma and the maintenance of the ignited plasma for tens of seconds, it will require additional development in two areas not now receiving much support—fueling and tritium handling, and one area now being heavily supported because of its near-term impact—neutral injection.

In the latter area, the emphasis in development beyond that required for TFTR must be on efficiency. There may also be a need to raise the maximum beam energy, possibly to as high as 300 keV which would require development beyond the TFTR beams. In this paper we will discuss these three technologies in the context of the requirements imposed by the TNS design.

Neutral Beams in TNS

Penetration of the injected fast neutrals must be adequate to heat the plasma core under all imaginable experimental operating conditions. If we examine the problem of adequate penetration in the high-density mode of operation ($\bar{n} = 2.10^{14} \text{ cm}^{-3}$) and allow some impurity accumulation during startup ($Z_{\text{eff}} = 1.5$) when charge exchange induced wall sputtering may be expected, a fast neutral energy of 300 keV results in slightly center-peaked deposition profiles for $\pm 20^\circ$ off-radial injection [1] into a 125 cm radius plasma. This injection scheme minimizes perturbation of the "normal" tokamak startup and operation physics, but requires a difficult technology to be developed.

Three alternative startup schemes which are presently being investigated theoretically, and which might lower the beam energy especially during the startup phase, are low density startup at full plasma radius, low radius startup at full density by programming the divertor separatrix, and injection of fast ions into ripple "windows" [2]. The possibly reduced complexity of neutral beam technology makes these schemes attractive, but further analysis of their effect on the tokamak physics is needed.

The requirement of 75 MW of power delivered to the plasma at energies up to 300 keV makes the problem of beam efficiency one of critical importance. Possible efficiencies range from about 12% in a $D^+ \rightarrow D^0$ "brute force" system to about 70% for a $D^- \rightarrow D^0$ system with an energy recovery system. The corresponding electric power required for the beam injectors would be about 625 MW and 107 MW, respectively. If we take a startup time for TNS of 4 seconds, after

which the α heating dominates and the beams may be turned off, the energy required for each TNS pulse in the two extremes above is 2500 MJ and 428 MJ, respectively. The former case raises large problems with respect to pulsed energy storage. The existing schemes for generating intense beams of negative ions, 1) direct extraction and acceleration of D^- ions from a D^- plasma generator and, 2) double charge exchange conversion of low-energy D^+ into a D^- beam followed by post-acceleration, may both have their inherently high neutralization efficiency further enhanced by the direct recovery of the un-neutralized ions' energy. This may result in a net efficiency approaching 70%.

All of these three high-efficiency schemes have their own sets of difficult physics and technology problems, the solution of which requires an aggressive beam research and development program. Table II shows the possible schemes being explored to enhance high energy efficiency. All of these options will be examined in the next three years.

TABLE II

POSSIBLE APPROACHES TO EFFICIENT NEUTRAL BEAMS AT 300 keV

Approach	Efficiency
1. D^+ (300 keV) $\rightarrow$ D^0 (300 keV)	12%
2. D^+ (300 keV) $\rightarrow$ D^0 (300 keV) + direct recovery	35%
3. D^+ (1 keV) $\rightarrow$ D^- (1 keV) $\rightarrow$ D^- (300 keV) $\rightarrow$ D^0 (300 keV)	50%
4. D^- (300 keV) $\rightarrow$ D^0 (300 keV)	50%
5. Approach 3 or 4 + direct recovery	70%

Superconducting Magnets in TNS

The TNS machine is primarily aimed at achieving a deuterium-tritium fueled ignition. Plasma engineering studies on energy containment and ignition conditions indicate the need for a plasma major radius of about 5 m and minor radius of about 1 m with moderate elongation. This plus the requirement of the space for plasma shaping coils, magnetic ripple criteria [3], a divertor and shield dictates a TF coil bore of about 4 x 6 m.

The maximum field required for the machine is not completely fixed yet. A maximum field of 8 T will give an ignition plasma density at $\beta \approx 10\%$. Higher fields will allow operation at a lower value of β and improve the confinement. Therefore, there is still a good incentive for increased field. A 12 T or even higher field might be required as a fall-back position if high β cannot be achieved.

The ignited plasma is expected to deliver about 1500 MW of thermal power during the burn cycle. If a radiation shield (about 60 cm) capable of three-decade attenuation is used and we assume 1/3 of the radiation hits the TF coils, the peak radiation load on the TF coils would be about 500 kW. This is 30 times higher than the EPR design point. However, it is expected that the TNS machine will not have a duty cycle any higher than 10% so this load can be removed by the refrigeration system.

The poloidal field coil system of the TNS will have similar features to that used in the ORNL-EPR conceptual design study [4]. The EF coil inside the TF coils can serve to reduce the pulse field

on the TF coils. However, in the flux-conserving mode operation, the plasma current is raised from 4 MA to 8 MA in about 4 sec [5]. In this time period the shielding function of the EF coils may be degraded. Therefore, the average $\dot{B}$ on TF coil may be somewhat higher than 0.05 T/sec (the value used in the EPR study), but possibly lower than 0.5 T/sec (the value to be employed in the coil fabricating program).

The poloidal field coils required for the TNS consists of two types of coils: the Ohmic Heating (OH) coils and the Equilibrium Field (EF) coils. The main part of the OH coil will be inside the center column of about 4 m in diameter. It serves to supply a flux swing of about 80 V-sec for the plasma current start-up and burn period. The EF coils should be as close to the plasma surface as possible in order to shape the plasma. On the other hand, if the EF coils are placed inside the shield, the radiation load would be very high. If we assume 10% interception of the peak load, then the value is about 150 MW. This is probably too high even for a water-cooled copper coil and consequently optimization of the position and shielding of these coils is an important area of development.

Toroidal Field Coils

Superconducting, resistive, and hybrid magnet designs were all considered for the TF magnet system. At first thought, one might assume that a smaller TF coil would suffice if it is resistive. One needs less shielding for high temperature resistive coils operating at 350-600 K than for superconducting coils at 4 K. However, because of

the need for space for the EF coils and some shield and the consideration of field ripple, the size of resistive coils would not be less than superconducting ones. The minimum space between the plasma edge and the TF coil necessary to meet the field ripple and access requirements might just be enough for shielding the superconducting coils. Furthermore, a resistive coil should be operated in pulsed mode. The capital cost of a resistive system and the operating costs (power consumption) are both probably higher than a comparable superconducting coil and the latter allows dc operation. The pulse nature also introduces a serious mechanical fatigue problem. An all-resistive TF magnet is most probably not scaleable to the larger devices such as an EPR. The technique of using resistive insert coils in a superconducting coil (hybrid design) for the size and environment required in TNS is still uncertain. A much closer look at this approach both from a mechanical and thermal point of view is necessary. The TNS design employs superconducting TF coils because they are feasible and can be applicable to larger machines. However, their design is not fixed and the options are many. A direct conclusion of the design project to date is the requirement of fields in excess of 80 kG at the coil to provide a margin for ignition in the presence of uncertain plasma behavior.

To select the best set of design choices, the US has a research, development, and demonstration program for superconducting toroidal magnets centered at ORNL. The program includes both in-house research, industrial design and fabrication and international cooperation. The industrial and international cooperation is centered

around the Large Coil Project (LCP). This project will let industrial subcontracts for building 6 coils with 2.5 x 3.5 m bore covering the range of viable possibilities above. These coils will be assembled in a compact torus which will allow full field testing by 1979. Cooperation with the International Energy Agency (IEA) on LCP will result in 1 to 3 coils being contributed to the compact torus demonstration by the IEA members.

Fueling Requirements for TNS

The principal problem to be addressed in fueling a TNS-like plasma is that of injecting D-T material deeply into the central portion of a hot, dense plasma against the confining effects of the magnetic fields. It is generally acknowledged that fueling by neutral beams would require prohibitively large power expenditures and that the most practical method of introducing fuel into such a device is by injection of solid D-T pellets accelerated to hypervelocities. The work of Gralnick [6] suggests that millimeter size pellets moving into the plasma at speeds of the order of 10^4 m/sec would be required to refuel a reactor grade plasma. Velocities of this magnitude have been reached in the laboratory using ballistic techniques, but the technology for accelerating hydrogen ice to hypervelocities has not been developed. In recent experiments, small frozen hydrogen pellets (about 0.1 mm) were successfully formed and injected into ORMAK [7] but only at velocities of 100 m/sec. Acceleration was accomplished by gas dynamic drag and it is believed that this technique can be extended with relative ease to reach velocities of about 1000 m/sec. The magnitude of the problem associated with attaining higher speed is

made clear by recognizing that at 10^4 m/sec each molecule in the pellet acquires about 1 eV of directed kinetic energy. Since the binding energy of the solid hydrogen phase is only about 0.01 eV, it is apparent that any acceleration scheme must be highly efficient in the sense that only a very small fraction of the energy involved could be permitted to appear as internal energy or heat. This criterion alone would ostensibly eliminate many of the known hypervelocity techniques that rely on strong impulsive loading of the pellet for acceleration.

Pellet Ablation Scaling Law

In addition to the work of Gralnick, a recent pellet ablation theory was proposed by Parks, Foster and Turnbull [8] to explain the observed behavior of pellets injected into the research tokamak, ORMAK [7]. The model accurately predicts the observed pellet lifetimes under the conditions investigated (ORMAK edge plasma parameters). The essential difference between this latest theory and that proposed by Gralnick is in the treatment of the interaction of the hot plasma with the pellet surface. Gralnick assumes that the incident plasma electrons evaporate ion pairs from the surface and that the resultant warm plasma cloud partially shields the pellet from the hot plasma by Coulomb collisions. The energy input required for the solid-to-plasma phase transition at the surface is the 32 eV corresponding to dissociative ionization of molecular hydrogen. In the model of Parks, et al., the assumption is made that a cold neutral gas is created at the surface and that the expanding neutral cloud shields the pellet by providing a dense medium for inelastic collisions with incoming

electrons. The energy requirement for this neutral ablation process is the 0.01 eV corresponding to the latent heat of evaporation of hydrogen. The key elements of the two theories differ widely but the predictions agree surprisingly well for a plasma representative of TNS. The neutral ablation model lends itself more easily to numerical calculations and a pellet lifetime scaling law can be extracted from it easily.

Larger pellet radii contribute to longer lifetimes and this effect tends to relieve the requirements for high injection velocities. However, pellets cannot be made arbitrarily large, but must be limited to sizes that contain no more than a small fraction of the total plasma ion content. In terms of this fractional value f and the plasma volume V , the expression for the pellet lifetime τ reduces to

$$\tau \propto 7.1 \times 10^{-3} (fV)^{5/9} n_e^{2/9} T_e^{-11/6} \quad (1)$$

One important result is that pellet lifetimes are increased with plasma density and volume. This is due to the fact that large dense plasmas will accommodate bigger pellets which provide a more effective gas dynamic shield. High β operation affords a slight but definite advantage in this respect.

The most important parameter in this scaling law is the plasma energy and it is this fact that may make hotter plasmas difficult to fuel. To demonstrate this we take as typical average parameters for TNS the values $n_c = 2 \times 10^{20}/m^3$, $T_e = 13$ keV, $V = 250$ m³. Assuming that pellets contain 10% of the plasma ion content ($f = 0.1$) a characteristic lifetime of 40 μ sec results. If the pellet must

penetrate 0.60 m to the center of the TNS shifted plasma, characteristic velocities in the 10^4 m/sec range are required. Non-circular shifted plasmas might offer significant advantages with respect to injection velocity requirements since pellets can be injected along the plasma minor axis from the outer wall which is closest to the plasma center. Because of the projected high velocity requirements, perpendicular injection seems wise.

Calculated fueling parameters for various experimental tokamaks are presented in Table III based upon the complete neutral ablation model and accounting for the variation of plasma parameters along the pellet trajectory. Pellet radii were limited to values that would contribute only 10% of the plasma ion content. It is clear that velocity requirements become more demanding as plasma parameters progress toward breakeven values. The 6000 m/sec velocity projected for TNS is high but as will be shown in the following section, at least one acceleration scheme appears to be capable of providing this performance.

It should be noted that none of the various pellet ablation models have had sufficient experimental verification and, in view of this, the results presented here should be viewed only as approximate guidelines for future hardware development programs.

Pellet Acceleration Methods

Some of the methods that may be applicable to acceleration of the needed millimeter size D-T pellets to hypervelocities are listed in Table IV, along with estimates of velocity capabilities. Other schemes will undoubtedly be proposed in the future and so this

TABLE III

ESTIMATES OF EXPERIMENTAL PELLET FUELING REQUIREMENTS
For Pellets Containing 10% of Device Total Ion Content and for Perpendicular Injection. Parameters are Plasma Center Values.

Device	Device Characteristics			Pellet Radius	Velocity
	a(m)	Te(keV)	$n_e(m^{-3})$	($10^{-6}m$)	(m/sec)
ORMAK	0.23	1.0	7×10^{19}	220	2000
ISX	0.25	1.0	7×10^{19}	240	2000
ORMAK Upgrade	0.30	0.9	1×10^{20}	300	1500
		3.0 ⁺	1×10^{20}	300	5000
PDX	0.47/0.57	1.0	3×10^{19}	300	1500
		3.0 ⁺	1×10^{20}	475	3000
PLT	0.45	1.7	5×10^{19}	360	3000
		4.0 ⁺	1×10^{20}	450	6000
TNS ⁺	1.25/2.0	20.0 ⁺	3×10^{20}	3000	6000

⁺with neutral beams.

⁺parameters chosen to preserve line average values.

TABLE IV

CATALOGUE OF ACCELERATING DEVICES FOR 1 mm RADIUS D-T PELLETS

Device	Speed (m/sec)	Comments
Gas dynamic	1000	Requires moderate working pressures and temperatures.
Light gas gun		
Electrostatic	600	Charging limit requires acceleration voltage = 13×10^6 V.
Laser rocket	500	Requires higher performance.
Electron beam rocket	2000	90% of pellet mass consumed as propellant.
Mechanical centrifuge	2500	Aluminum rotor.
	5400	Fiber composite rotor.

discussion is not intended to exhaust all possibilities.

No distinction is made between low pressure gas dynamic acceleration and the light gas gun because hydrogen pellets are not expected to survive the extremely high pressure atmosphere at which present design gas guns operate. The limit of 10^3 m/sec corresponds to the approximate speed of sound of hydrogen gas at ordinary working pressures and temperatures. Because of the proven performance of light gas guns, this device should not be excluded from consideration. Its exact capabilities with hydrogen pellets should be determined experimentally.

Electrostatic acceleration techniques are limited by the inability to attain sufficiently large charge-to-mass ratios on pellets above $1\ \mu$ size. The low tensile strength of solid hydrogen limits the amount of charge that a 1 mm pellet can accommodate (surface space charge fields $< 2.5 \times 10^8$ V/m).

It has been suggested that a giant laser pulse could be used to ablate a small amount of material from one side of a pellet to provide a large impulse for acceleration. The strong shock wave that is transmitted to the pellet bulk imparts half of its energy as directed kinetic energy with the remaining half appearing as internal heat. The low binding energy of the condensed phase limits the strength of the shock wave and hence the velocity. A limit of about 500 m/sec has been estimated for this concept; but it is recognized that it needs to be explored. The use of a focused electron beam to provide an ablation driven acceleration is also possible. In a properly designed device, the velocity of the expanding material would not be

as high as that for the laser ablation concept and the impulse would be less severe.

The final acceleration concept considered is a purely mechanical device that would utilize centrifugal forces to accelerate pellets outward along the arms of a rotating arbor designed for optimum peripheral, or tip, speeds. In the present concept, pellets would be introduced at the arbor hub after being cut from a continuously extruded filament of solid hydrogen and would be guided by channels formed in the arbor arms to the tip where they leave with equal components of radial and tangential velocity. Present designs attain tip speeds of 1800 m/sec using high strength aluminum alloys as the material of construction in the arbor. This should provide pellet velocities of the order of 2500 m/sec. For this device, performance varies as the square root of the strength-to-weight ratio of the material used. Several fiber composite materials exhibit strength-to-weight ratios several times larger than that of aluminum. Highest performance would result from a choice of either Kevlar 49 or Thornel 300 fibers embedded in an epoxy resin matrix. Pellet velocities as high as 5400 m/sec are indicated with 1 m diameter rotors operating at 1000 Hz rotational speed. Development of this device might lead to a slight increase in performance, but velocities higher than 10,000 m/sec do not seem possible.

Of the concepts considered here, the centrifugal accelerator appears to be the most practical solution for a TNS fueling device with 6000 m/sec capabilities. The gas gun is a likely back-up device although it has not been proven with hydrogen pellets. If the

theoretical predictions prove to be correct, it is clear that the most advanced type of accelerator will need to be developed.

Tritium Recycle in TNS

The tritium recycle system for TNS will remove unburned fuel, ash (helium), and impurities from the divertor chamber; purify the unburned fuel by removing helium and other impurities; separate high purity deuterium for recycle to the neutral beam injectors; prepare a small hydrogen purge stream for discharge; supply suitable fuel surge and storage capacity; and provide containment sufficient to hold tritium exposures and releases to acceptable levels. The ORNL design for a TNS tritium recycle system is expected to resemble that proposed for EPR. This would be a system with essentially the same process components expected to be used in EPR, and they could be easily applied to even larger devices.

Removal of fuel from the divertor chamber will require a vacuum pump with high pumping speeds, complete containment of tritium, complete recovery of fuel, minimum probability for adding contaminants to the plasma, and ability to operate under the space restrictions, radiation fields, and magnetic fields around the torus. A compound cryosorption pump is recommended with separate sorption panels for pumping the unburned D-T fuel and the helium ash. The fuel can be pumped by a molecular sieve panel cooled to approximately 15°K; helium can be pumped by a similar panel cooled to 4.2°K (liquid helium temperature). The 4.2°K panel would be located "behind" the 15°K panel so only a minimum amount of fuel would reach the helium pump. The most likely alternative to this would be substitution of

a 4.2°K cryocondensation panel for the 15°K cryosorption panel in the fuel pump. All other pump systems considered failed to meet one or more of the TNS requirements.

In the compound pump system just proposed, considerable separation of helium from the fuel will be accomplished in the vacuum pump itself. The final purification step would take place in beds of hydride-forming materials, probably uranium. These beds can be operated to remove all likely impurities (He, oxygen, carbon, nitrogen, etc.), and they will serve other purposes as well. They will provide necessary surge capacity at low pressures and fuel accumulation capacity in the event that the pumps or other portions of the fuel recycle system loses refrigeration. Other purification systems have been considered, particularly permeable Pd membranes, but these require compressors, or at least transfer pumps, which are undesirable and do not appear to be necessary.

To supply a high purity deuterium stream for the injectors, an isotope separation system will be required. The most promising isotope separation technique for TNS (and later fusion devices) is cryogenic distillation. This is a well established technique, and it appears to be clearly the most economic technique, principally because of the low tritium inventory and energy costs. One column would produce the high purity deuterium stream, and a second smaller column would process a portion of the deuterium to produce a hydrogen stream for discharge.

Tritium containment is usually associated with the fuel recycle system partly because most of the TNS tritium inventory (hazard) will

be associated with the recycle system. Since tens of millions of curies of tritium will be used and stored in TNS, a most effective and reliable containment system will be required. We suggest a multi-level containment system with active atmosphere process systems to remove tritium at each level. The fuel recycle system will be constructed as compactly as possible once allowances are made for convenient maintenance of all components. All portions containing large quantities of tritium or having significant probabilities for tritium release will be contained in a tight enclosure with continuous processing of the atmosphere to hold the tritium concentration low enough to prevent unacceptable releases of tritium from diffusion and leaks through the enclosure and from the necessary exhaust to the atmosphere. In addition, an emergency tritium cleanup system is required for the much larger reactor room.

Although the tritium recycle system may not be either the most expensive system to build or the most difficult system to develop for TNS, it certainly will be one of the most critical systems because of the crucial role it plays in containing the relatively hazardous tritium containing fuel and preventing excessive contamination of either the TNS equipment or the environment. The TNS can serve as a test stand for many components of tokamak fusion reactors, but it will be necessary to have a high reliability tritium system for TNS. Most of the development and testing of the tritium system should precede TNS, and we have proposed a relatively rapid development and testing program to assure that the components we have selected can fit into a working tritium recycle system. This assurance will be

needed by the time the design for TNS is finalized, perhaps as early as 1980. This requires detailed design of a major tritium test facility to begin almost immediately, and construction should begin within a year.

Conclusion

The physical uncertainties in the TNS version of the PDFD will be addressed by an array of experiments over the next five years. The beams and superconducting magnets are also under development on a large scale. These programs expect to have minimally acceptable products for TNS by 1980. The pellet fueling and the tritium handling systems require the largest increase in effort in the immediate future but promising techniques exist in both areas. We conclude that with a reasonably aggressive development program, firm answers to the questions existing in the TNS design can be obtained by 1980. This would allow this PDFD to be in operation by 1987.

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