Limiter Pumping System for Divertorless Tokamaks

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Abstract

A specially shaped toroidal belt limiter for high power tokamak systems is developed. The peak heat load is reasonable (~200-400 W/cm^2), and a water cooling system is presented. A choked gap permits adequate pumping of plasma fuel and ash.

I. Introduction

Tokamak reactors may operate with magnetic divertors for impurity control and plasma exhaust. Conceptual designs of both poloidal and bundle divertors have been carried out[1-4] and experiments indicate divertors work.[5,6] Engineering designs of divertors for reactors show they can add complexity to the overall design and make difficult procedures for maintenance and for the regular replacement of major internal components such as blankets. Divertorless tokamak reactors are being studied[7-8] to determine whether this approach is feasible in terms of plasma physics and whether the approach can be engineered.

The plasma physics issues of divertorless tokamak operation center on whether alpha ash can be removed from the plasma and whether impurities can be successfully controlled. This is potentially feasible[8] if a cold edge is maintained by impurity recycle and by heavy radiative power loss through the nonequilibrium stripping of recycled impurity atoms. Recycle of fuel at the plasma edge sustains a relatively flat density profile. We will not deal in detail with the plasma physics issues in this paper but concentrate on the engineering design of a limiter-pumping system.

The engineering of a limiter design to handle the heat flux and to permit limited helium ash pumping is the key to successful divertorless operation. Various designs have been proposed[9,10] but only the passive limiter design by Schmidt[10] for the Tokamak Fusion Test Reactor (TFTR) has been carried beyond the preliminary stage. That limiter is designed to handle the heat flux in a short pulsed reactor (pulse length less than a few seconds) but does not provide for ash or fuel pumping.

In this paper, we develop a conceptual design for a reactor limiter system that minimizes the peak heat flux and permits sufficient helium pumping. The limiter consists of two shaped toroidal belts tangent to the plasma at the midplane of the belt. The belt width extends some distance into the plasma scrape-off layer but a gap remains between the limiter edge and the first wall. It is important to recognize[11] that the equilibrium alpha density will be about 10% of the ion density if the alpha removal rate is 20% of the alpha birth rate. The aim is thus to pump 20-30% of the alphas, not all of them. Concomitantly, the recycling of both helium and fuel ions increases the effective "residence time" of particles in the chamber by a factor of $(1-R_p)$ beyond the characteristic particle confinement time τ_p. The effective particle confinement time is:

$$(\tau_p)_{eff} = \frac{\tau_p}{1-R_p} \tag{1}$$

where R_p is the recycling coefficient (the probability that a particle neutralized at the limiter or wall returns to the plasma). The "engineering fractional burnup" is related to $(\tau_p)_{eff}$ which in turn sets the rate of tritium flow to the tritium reprocessing system, the tritium fueling rate, and the tritium inventory. In the next sections, the principles of the limiter design are described and various engineering features such as a preliminary mechanical design, coolant passage design and heat transfer, and vacuum system analysis are discussed.

II. Principles of the Limiter-Pumping System Design

Consider for simplicity two flat belts extending toroidally around a plasma of circular cross section. The relevant geometry is shown in Fig. 1. The toroidal plasma of minor radius a and major radius R is limited by the two flat plates. (In practice, the belt is segmented toroidally.) The plasma in the region of minor radius r greater than a is referred to as the plasma scrape-off layer. Let δ represent the distances into the scrape-off region. A magnetic flux surface of radius $a+\delta$ will intercept the limiter at point c (in Fig. 1) at a distance d from the center of the plate. The angle θ between the plate limiter and the flux surface at the point of intersection is simply:

$$\theta = \cos^{-1}\left(\frac{a}{a+\delta}\right) \tag{2}$$

which for $\delta/a << 1$ is:

$$\theta \simeq \left(\frac{2\delta}{a}\right)^{1/2} . \tag{3}$$

To this order, the distance d is

$$d \simeq (2a\delta)^{1/2} . \tag{4}$$

Note that for noncircular cross section plasmas, "a" is simply replaced by a local radius of curvature, r_a, (the local plasma radius) so long as $\frac{\delta}{r_a} << 1$.

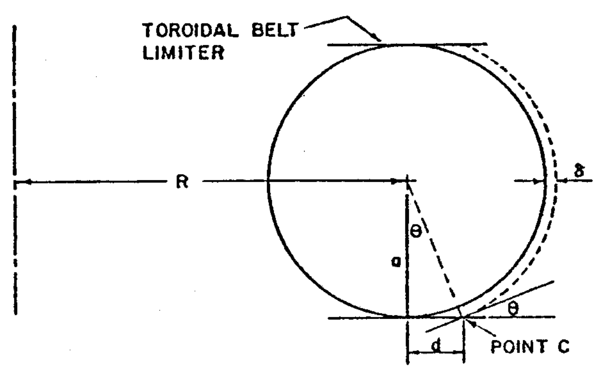

Fig. 1 Geometry for analysis of a simple toroidal belt limiter.

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The heat load on the limiter face is a function of the spatial variation of the heat flux in the plasma scrape-off. A reasonable assumption is:

$$q(\delta) = q_o e^{-\delta/\lambda_E} \qquad (5)$$

where λ_E is a characteristic e-folding distance. The length, λ_E, need not be equal to λ_p, the e-folding distance for the plasma density. Typically λ_E is less than λ_p which means particles extend further into the scrape-off layer than does heat. Along the limiter plate, the heat flux at point d is:

$$q(d) = q_o e^{-\delta/\lambda_E} \left(\frac{2\delta}{a}\right)^{1/2} . \qquad (6)$$

The maximum heat flux is

$$q_{max} = q_o \left(\frac{\lambda_E}{ea}\right)^{1/2} \qquad (7)$$

which occurs at $d \equiv d_c = (a\lambda_E)^{1/2}$ when $\delta = \delta_c = \lambda_E/2$. A plot of q/q_o versus δ, the distance into the scrape-off plasma, is shown in Fig. 2 for several likely values of λ_E in a reactor plasma edge. Note that the heat flux is theoretically zero at the point of plasma tangency with the limiter. This will be used later in the design of coolant flow paths within the limiter.

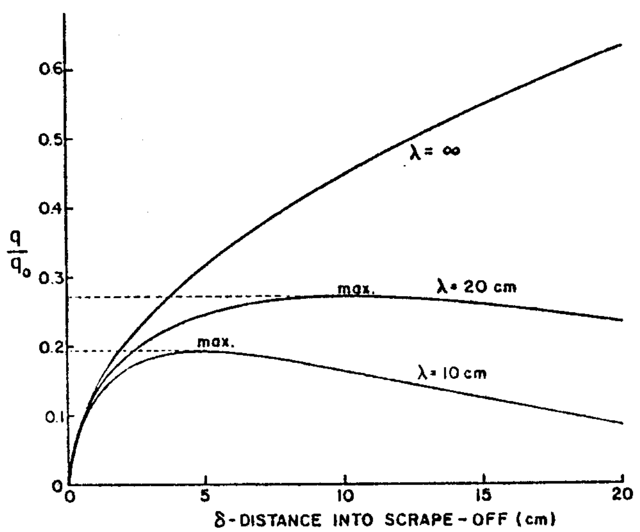

Fig. 2 Plot of q/q_o vs. δ, the distance into scrape-off for λ_E = 10 cm, 20 cm and ∞.

For $\delta > \lambda_E/2$ (or $d > d_c$), the heat flux decreases from its peak value. To maintain q(d) a constant equal to q_{max} for $d > d_c$, it is necessary to curve the limiter away from the plasma as shown in Fig. 3 to decrease the projected area. At any δ, the angle ϕ between the flux surface and the limiter such that $q = q_{max}$ for $d \geq d_c$ is:

$$\phi = \sin^{-1}\left(\left(\frac{\lambda_E}{ea}\right)^{1/2} e^{\delta/\lambda_E}\right) \simeq \left(\frac{\lambda_E}{ea}\right)^{1/2} e^{\delta/\lambda_E} \qquad (8)$$

where the latter approximate equality holds for $\lambda_E/a \ll 1$.

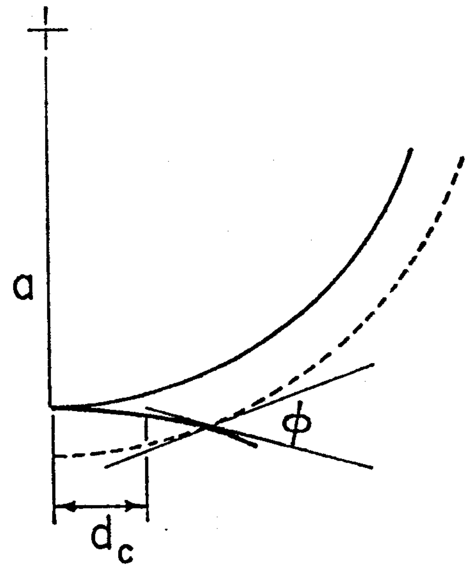

Fig. 3 Curvature of plate to increase the angle between the limiter and a flux surface such that the heat flux everywhere on the plate beyond d_c equals the heat flux at d_c.

The value of q_o is related to the total power deposition in the plasma. This is simply the alpha power, P_α, in an ignited D-T plasma and is $P_\alpha + P_{inj}$ in a driven device. P_{inj} is the external heating power. Furthermore, q_o represents heat crossing the plasma surface at "a" by conduction and convection, excluding power leaving by radiation and charge exchange neutrals. Let f be the total power leaving the plasma as radiation or charge exchange neutrals and let P_o be the power deposition in the plasma. Then q_o is:

$$q_o = \frac{P_o(1-f)}{2\pi R\lambda_E} . \qquad (9)$$

The heat flux on the limiters depends on the number of limiters N_L. The maximum heat flux is:

$$q_{max} = \frac{P_o(1-f)}{2\pi R(ea\lambda_E)^{1/2} N_L} . \qquad (10)$$

For illustration, we have estimated q_{max} for three different reactor systems: International Tokamak, INTOR [12] (nominal fusion power of 500 MW, R = 4.8 m, a = 1.2 m); the Engineering Test Reactor (ETF) [13] (nominal fusion power of 1000 MW, R = 5 m, a = 1.2 m); and STARFIRE,[14] a conceptual tokamak reactor (nominal fusion power of 3200 MW/ R = 7 m, a = 1.94 m). All three devices have noncircular plasmas with a height to width ratio of 1.6. A typical value of f for divertorless operation is 0.5 and could be as high as 0.8. Assuming λ_E = 10 cm and f = 0.5, the peak heat flux with two toroidal belt limiters at top and bottom of the plasma is 110 W/cm^2 for INTOR, 220 W/cm^2 for ETF, and 400 W/cm^2 for STARFIRE. At f = 0.8, the peak heat flux on a STARFIRE limiter drops to

160 W/cm^2. These values are high but manageable, as discussed in the sections to follow.

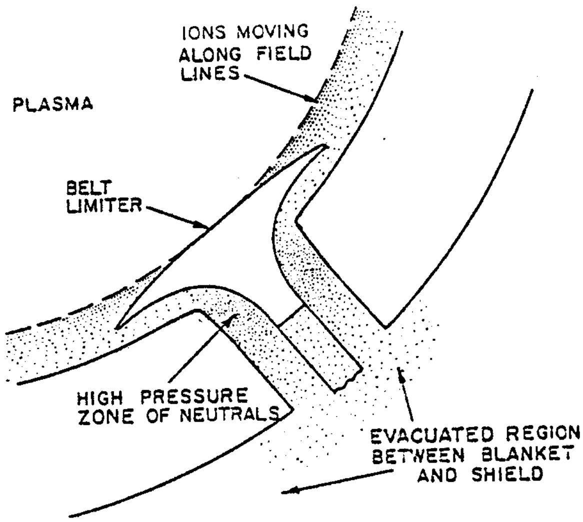

Fig. 4 Illustration of toroidal belt limiter showing the gap for plasma flow to the inner stem. Flow conductance to the pump is strongly aided by geometric considerations and by the ionizing power of plasma flowing into the gap.

The equilibrium helium content in a D-T plasma is a function of the helium removal rate from the reaction chamber. From a simple global balance, the ratio of helium density to fuel ion density is:

$$\frac{n_\alpha}{n_i} = \frac{(n_i \tau_p) \langle \sigma v \rangle}{4(1-R_p)} \quad (11)$$

where τ_p is the alpha particle confinement time and R_p is the recycling coefficient. R_p is the probability that a particle reaching the plasma edge is recycled into the plasma rather than being removed from the chamber or buried in nearby surfaces. Without pumping, R_p is 1. With only partial pumping such that $R_p \sim 0.7 - 0.8$, Brooks[11] has shown that n_α/n_i can be 10% or less. Thus, alphas must be removed at a rate equal to 20-30% of their birth rate. This can be accomplished by leaving a gap for plasma flow between the end of the limiter and the first wall. A schematic is shown in Fig. 4. Plasma flowing in the gap strikes the stem of the limiter and is neutralized. The neutral atoms have an enhanced flow conductance to the pump ports relative to flow back through the gap. Furthermore, calculations by Jacobson and Boozer[15] at Princeton show that neutral gas attemping to escape through the gap has a high probability of being ionized and returned to the central limiter stem. Thus, the neutral gas pressure at the head of the pump port can be much larger than the neutral pressure around the plasma. This considerably eases the helium pumping problem as we discuss shortly.

As a concrete example, suppose λ_E = 10 cm and that the limiter extends into the scrape-off such that the edge intercepts the flux surface at δ = 10 cm. Then 63% of the heat will be incident on the limiter surface facing the plasma. If the gap between the limiter and the first wall is 10 cm, 20% of the heat will flow to the stem. The particle flux is likely to decay more slowly with δ than the heat flux, i.e., $\lambda_E < \lambda_p$. In this case, greater than 20% of the particles, including helium, strike the inner stem of the limiter and with high probability are pumped.

We are fully cognizant of plasma factors such as the limiter sheath effect,[16] sputtering, and arcing. An absolutely essential factor is maintaining a cold plasma edge. This part of the problem will be discussed in future publications.

Thermal Design of the Limiter

A water cooled thin wall tube is the most effective configuration for absorbing a large thermal flux in a magnetically confined fusion reactor environment. Laboratory experiments have demonstrated[17] that water at high velocity in forced convection is capable of absorbing 3 kW/cm^2. However, it is much more complicated to determine the maximum allowable thermal flux for safe and reliable operation over an extended period of time, subjected to a large number of pulses in a fusion reactor environment. Heat transfer, pressure drop, thermal stress, fatigue, creep, radiation damage and sputtering, each can be the flux limiting parameter under different conditions. The discussion here provides a credible limiter design from thermal hydraulic considerations.

The parameters of the limiter are listed in Table 1. The mechanical design of the limiter is shown in Fig. 6. Basically, it is a stream of high velocity heavy water flowing around the outer edge of the limiter to absorb the heat flux. Heavy water is used as the coolant to reduce the load on the isotope separation system.[14] The structure used is OFHC copper for its thermo-physical properties. The maximum design temperature is limited to 200°C to avoid excessive swelling. This is 1/3 of the melting temperature for copper. The velocity required to achieve a heat transfer coefficient of 6 w/cm^2-°C (10,000 BTU/hr-ft-°F) is 12.5 m/sec. Both values appear to be reasonable and readily obtainable.

The thermal and hoop stresses on the first wall can be calculated by the following two equations:

$$\sigma_t = \frac{\alpha}{2} \frac{E}{(1-\nu)} \Delta T \quad (12)$$

$$\sigma_h = \frac{DP}{2t} \quad (13)$$

where D is the diameter of the tube, P the coolant pressure, t the tube wall thickness, α the coefficient of expansion, E the modulus of elasticity, ν Poisson's ratio and ΔT the temperature difference across the limiter front surface. The properties of OFHC copper[18] at 200°C are the following:

$$\alpha = 18 \times 10^{-6} \ {}^{o}C^{-1}$$
$$\nu = 0.3$$
$$E = 1.3 \times 10^{5} \text{ MPa} .$$

The sum of the thermal and hoop stress is the maximum stress on the limiter surface and is calculated to be 36 MPa, which is well within the allowable stress limits for OFHC copper.

Further investigation of the limiter design is obviously required. The most difficult problem is that of sputtering. A low Z coating on the first wall which, after being sputtered away from the first wall, will be deposited on and protect the limiter is a possible solution.[14] High temperature operation of the limiter should also be investigated to allow the use of this energy in the power cycle.

Table 1 Thermal Hydraulic Parameters for NUWMAK Limiter

Power to the limiter	150 MW
Maximum heat flux	4 MW/m^2
Limiter structure	OFHC copper
Limiter coolant	D_2O
Coolant velocity	12.5 m/sec
Coolant heat transfer coefficient	6 w/cm^2-°C
Thermal load per tube	10 kW
Coolant flow rate per tube	980 g/sec
Coolant inlet temperature	100°C
Coolant temperature rise	5°C
Coolant pressure drop	0.5 MPa
Maximum coolant pressure	1 MPa
First wall thickness	2 mm
Maximum first wall temperature	192°C
Maximum stress	36 MPa

CROSS SECTION OF NUWMAK BETWEEN TF COILS

(with inner RF cavity)

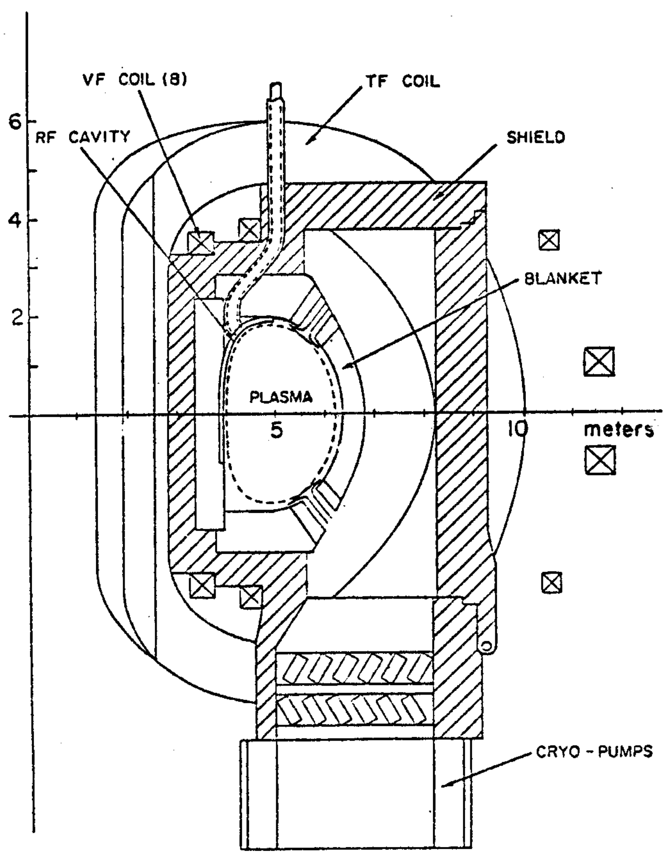

Fig. 5 Illustration showing location of limiters in NUWMAK.

Vacuum Pumping

The pumping system in this case is based on NUWMAK, a conceptual 660 MW_e(net) tokamak power reactor designed by the University of Wisconsin. The α particle production is 6.8 x 10^{20} particles/sec which corresponds to a helium gas throughput of ~ 50 torr-liters/sec. If we assume that 30% of the helium has to be pumped away, then the vacuum system must be capable of handling 15 torr-liters/sec. The needed effective pumping speed depends on the helium partial pressure in the region immediately below the limiter. For a helium partial pressure of 1 x 10^{-4} torr, the effective pump capacity needed is 1.5 x 10^5 liters/sec and for 1 x 10^{-5} torr, it is 1.5 x 10^6 liters/sec.

As shown in Fig. 5 the limiter slots exhaust to the zone bounded by the blanket and the shield. In NUWMAK, there were 8 pump stations situated below grade and shielded against neutron streaming by chevron shields. The system had 160 m^2 of cryosorption pump surface available.[8] If we conservatively assume a pumping speed for helium of 1.5 liters/sec-cm^2, then the pump capacity will be 2.4 x 10^6 liters/sec. The conductance of the limiter slots for helium at 473 K (the max. temperature of the limiter) is also conservatively estimated at 1 x 10^7 liters/sec if the slots are 35 cm wide and 40 cm deep. The effective pumping speed of the system is then 1.9 x 10^6 liters/sec. It appears that the NUWMAK pumping system can keep the helium partial pressure at the base of the limiter at 1 x 10^{-5} torr.

If a higher helium partial pressure can be tolerated at the base of the limiter, as indicated by Jacobson and Boozer,[15] then the pumping system can be reduced. In that case, it is very likely that the system capacity will be dictated by the requirements for pumping the hydrogen (D_2, T_2 & DT) which in turn depends on the fractional burnup of the reactor.

Among the disadvantages of having the reactor gases exhausting into the space between the blanket and the shield are:

1 - Tritium contamination of the large inner shield space.
2 - The shield has to be vacuum tight.
3 - Increased volume for initial pumpout.

Although some of these problems are serious, the benefits of such a design are tremendous. They are:

1 - No seals needed between blanket modules.
2 - Shield closures can have elastomer seals.
3 - No vacuum ducts to remove during limiter changeout.

Mechanical Design

One of the difficult problems in implementing such an impurity control mechanism is the actual construction of the limiter. The relatively high heat load limits the choice of materials and unfortunately materials that are easily fabricable, in particular with respect to welding, such as SS, are unsuitable because of the high thermal stress due to poor thermophysical properties. Sputtering away of the material is perhaps even more serious and will be neglected for the present.

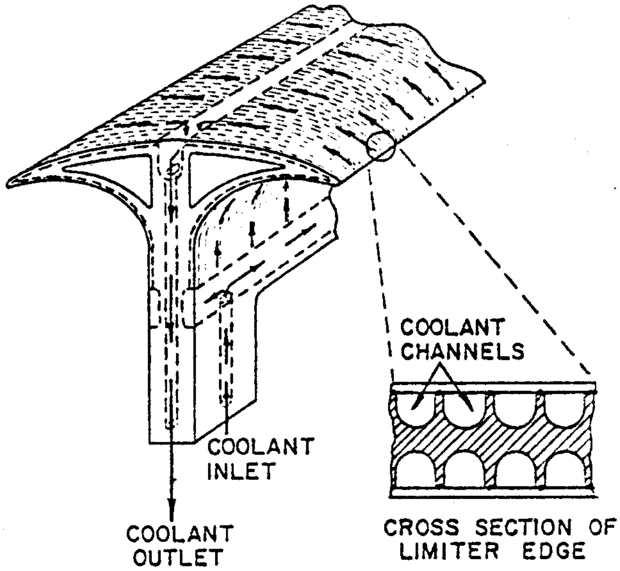

Fig. 6 Isometric drawing of toroidal belt limiter for a tokamak.

In the base case we will assume the limiter is made of water cooled OFHC copper. Figure 6 shows the general construction features of the limiter. The limiter is assembled from two components, the core, which has the cooling channels, manifolds and supply and return connection already machined into it, and an outer skin which essentially covers the outer surface of the limiter. It is the outer component which has to withstand the heat load and the high sputtering rate attendant on the limiter. The lifetime and reliability of the limiter will depend on the design of the outer layer. Although the core is assumed to be OFHC copper, machining such a complex shape will be challenging. It may be possible to make the core from yellow brass which could be cast and subsequently machined only to true up the dimensions. Although yellow brass has 1/3 of the thermal conductivity of copper, its coefficient of thermal expansion is the same can, therefore, be brazed with a high temperature brazing alloy. In any case, a totally copper limiter will also have to be brazed. The implications of having a brazed structure in such a harsh environment have to be carefully examined.

Placement of the limiter within the plasma chamber will have to be very exact. Since it appears that the sputtering problem will necessitate limiter changeout more frequently than the first wall, the design must be simple enough to accommodate that. In Fig. 7 we show a possible solution. The limiter is attached to a section of the blanket which is a part of the limiter module. Removal of a limiter section (same fraction of the total belt) will entail removing that part of the blanket attached to it as shown at the top of Fig. 7. Accurately located guides and stops will be needed for proper limiter location. A method for securing the limiter module to the rest of the blanket will also have to be devised. Obviously, coolant connections to both limiter and the extracted part of the blanket have to be made. This problem is one that

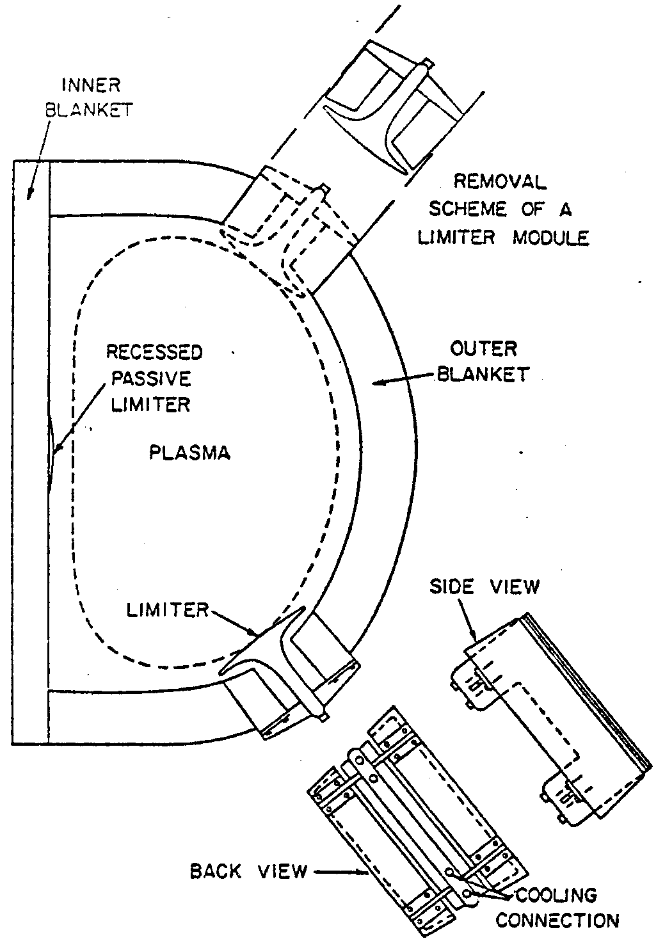

Fig. 7 Illustration showing method for attachment and extraction of limiter modules.

has puzzled designers for a long time, especially since these operations have to be made by remote control.

At the present, there don't appear to be easy solutions to the problems of removal of limiters from the reactor. In NUWMAK, it may be possible to design remotely operated machinery which can operate in the space between the back of the blanket and shield. In that case, the shield may not have to be disturbed for limiter changeout. However, the space available shown in Fig. 5 will most likely be occupied by other equipment (blanket cooling connections, etc.) which will make such an operation very difficult. The obvious solution is to design a limiter which will have the same lifetime as the first wall and can then be changed out during routine first wall maintenance. This presents a challenge to both material specialists and design engineers.

Conclusions

It appears that a specially shaped toroidal belt limiter for high power tokamaks which will allow 20-30% alpha particle removal can be designed using water cooling. The peak heat loads will be on the order of 200-400 W/cm^2. Plasma flowing into the limiter gap ionizes neutrals attempting to leave thus creating a high density neutral zone which mitigates the pumping requirements. Material sputtering is a serious problem which will require further investigation.

Acknowledgement

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