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EDGE-PLASMAS AND WALL PROTECTION IN RFPs†

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ABSTRACT

The Reversed-Field Pinch (RFP) has the ability to operate as a compact, moderate-to-high beta, high-power-density system. A compact system requires careful control of the particle and heat fluxes impinging on plasma-facing components. A strongly recycling, toroidal-field open divertor combined with a highly radiating (>90% of plasma heating power) core plasma is required. An open divertor configuration locates the plate near the field null to take advantage of flux expansion and minimum poloidal asymmetries to minimize peak heat fluxes. The physics and engineering requirements are quantitatively discussed for an evolutionary sequence of impurity/ash-control schemes for ZT-40M (0.4 MA)→ ZT-P (0.08 MA) → ZTH (2-4 MA)→ FTF/RFP (10 MA)→ TITAN (18 MA).

I. INTRODUCTION

A main attribute of the Reversed-Field Pinch (RFP) is operation at moderate-to-high betas and high power density in a relatively small volume. Central to exploiting the benefits of compactness in a system that inherently combines confinement (poloidal field), heating (ohmic), and current drive (oscillating-field current drive, OFCD) functions into a single subsystem is the control of the particle and heat fluxes. Operation within engineering limits imposed by heat flux and wall erosion in steady-state, compact, high-power-density devices is possible only by a) controlling carefully magnetic field errors and b) shedding a large fraction (> 0.9) of the plasma internal energy by radiation; both approaches ensure uniformity of heat flux at the first wall. High radiation fractions may be more easily achieved in RFPs than in tokamaks because experimental evidence[1] suggests RFPs operate with a "soft" (i.e., without disruption) beta-limit which decreases plasma transport loss to compensate for increased radiation loss so as to maintain the total energy confinement time constant.

Concepts for impurity/ash control and wall protection in the RFP have developed over the last five years. Present and proposed RFP experiments at Los Alamos use carbon limiters and armor tiles. Upgrades of devices now under construction, ignition devices,[2] and high-power-density reactors,[3] however, require divertors to remove the heat and control erosion. The engineering requirements of adding poloidally symmetric toroidal-field divertors on both the ZTH/CPRF[4] reversed-field-pinch device presently under construction at Los Alamos and on the existing ZT-P[5] device, which is a prototype for ZTH. An evolutionary picture of the plasma particle and thermal energy control required to take present RFP experiments to a commercial reactor is presented herein.

Parameters for various-sized RFP concepts are given in Table I. The ZT-P and ZT-40M[1] machines exist at Los Alamos, ZTH is under construction, and FTF[2] (fusion test facility) and TITAN[3] (commercial reactor) represent conceptual design studies. From the viewpoint of wall protection, these machines can be separated in to two categories: short-pulsed machines consisting of the ZT series, which do not require active cooling of the first wall or limiters, and steady-state fusion devices which require active cooling.

II. SHORT-PULSED EXPERIMENTS

A. ZT-40M. The ZT-40M experiment has verified the concept as a near-minimum energy state configuration with robust global stability properties, dynamo action to sustain the configuration, high-beta operation ($\beta_\theta = 3\mu_o \int p dV / B_\theta^2(r_p) = 0.1\text{-}0.3$) in a strongly ohmically heated (7 MW) discharge. The wall is protected by a series of carbon ring limiters, but even though the average surface heat flux is only 0.7 MW/m^2, holes have been burned in the liner (first wall) which serves as the vacuum boundary. Plasma behavior was also found to improve when magnetic field errors were reduced. Experimental measurements[6] in the plasma edge show a small component of fast electrons carry most of the plasma energy (in the anti-current direction) and cause a peak-to-average energy load on a limiter of 14. Indeed, ZT-40M operates in the high streaming parameter regime (0.1) where runaways are expected.

A simple model for the heat flow in the edge plasma was developed to describe this peaking. The normal heat flux to the wall is calculated for a given plasma shift, Δ, of the centroid of a core plasma of radius r_p relative to the centroid of a first wall of radius r_w (Fig. 1). Flux surfaces external to the core plasma are assumed to be concentric about the core-plasma surface. The heat flux to the wall is assumed to consist of three parts: (a) transport along field lines,

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TABLE I. DEVICE PARAMETERS FOR RFPs AND EVOLUTION OF WALL PROTECTION SCHEMES

	ZT-40M	ZT-P	ZTH	FTF	TITAN
Principal Function	concept demonstration stability/dynamo	ZT-H prototype	confinement scaling experiment	ignition/ neutron production	commercial reactor
Minor radius, $r_p(m)$	0.2	0.062	0.4	0.3	0.6
Major radius, $R_T(m)$	1.14	0.45	2.4	1.8	3.9
Plasma surface area, $A_p(m^2)$	9	1.2	38	21	92
Plasma toroidal current, $I_\phi(MA)$	0.4	0.08	2-4	10	18
Plasma thermal power	6.3	7.5	20-28	49	457
• ohmic power, $P_\Omega(MW)$	6.3	7.5	20-28	26	8
• alpha-particle power, $P_\alpha(MW)$	–	–	–	24	449
• radiative power, $P_{RAD}(MW)$					
Average surface heat flux, $q_s(MW/m^2)$	0.7	4.1	0.5-0.8	2.3	4.9
Discharge time, $\tau_D(s)$	0.03	0.0015	1	∞	∞
Wall protection scheme	limiter	limiter	carbon armor	divertor	divertor

$q_{\|}$; (b) radial diffusive transport characterized by a diffusion coefficient, D, and a radial gradient scale length, λ; and (c) a radial radiation transport characterized by a core-plasma radiation fraction, $f_{RAD} \equiv P_{RAD}/P_{TOT}$. Where, P_{TOT} is the total source of power to the plasma core, P_{RAD} is the core radiated power, and edge-plasma radiation is neglected.

The normal heat flux to the first wall is given by

$$q_w = q_{\|} \sin\alpha + q_r \begin{cases} 0, & \text{for } \alpha \geq \frac{\pi}{2} \\ \cos\alpha, & \text{for } \alpha \leq \frac{\pi}{2}, \end{cases} \qquad (1)$$

where,

$$q_{\|} = w v_{\|},\ q_r = wD/\lambda + f_{RAD}P_{TOT}/A$$

$$w = w_o e^{-\frac{r-r_p}{\lambda}},\ \lambda = (D\ell/v_{\|})^{1/2},\ A = L2\pi r\ .$$

The angle between a field line and the wall is α, the edge-plasma energy density is w, the flux surface area in the edge plasma is A, and the $L = 2\pi R_T$. The average field line length along which energy is transported, ℓ, is estimated to be $\ell = 2\pi r/n_D$ for the RFP, where $n_D = 1$ or 2 is the number of directions of energy flow. The $n_D = 1$ option is included because ZT-40M exhibits preferential energy flow in the opposite direction of the plasma current. The results presented herein used $n_D = 1$. The average heat flux to the wall is $q_w = P_{TOT}/A_w$. Thus, the heat flux peaking factor can be written:

$$\frac{q}{q_w} = \frac{2\pi r_w(1 - f_{RAD})}{n_D\lambda} e^{-\frac{r-r_p}{\lambda}} \sin\alpha + \qquad (2)$$

$$\left[\frac{r_w}{r_p}(1 - f_{RAD}) e^{-\frac{r-r_p}{\lambda}} + \frac{r_w}{r} f_{RAD}\right] \begin{cases} 0, & \text{for } \alpha \geq \frac{\pi}{2} \\ \cos\alpha, & \text{for } \alpha \leq \frac{\pi}{2}. \end{cases}$$

The model was first applied to existing ZT-40M data[8] (Table II). With the parallel flow velocity given by $V_{\|} = 2G(kT/m_i)^{1/2}/n_D$.

A standard fluid edge plasma (i.e., $G = 1$ and $D = 1\ m^2/s$) with ZT-40M input parameters is presented as Case 1 (Table II). For this case, the calculated heat flux peaking factor, q_{MAX}/q_0, is a factor of 4.5 less than the experimentally factor. Cases 2-5 varied Δ, T, D, and G until the peaking factor matched the experimental value of 14. All matching cases have a radial gradient scale length of about $0.2\,mm$, which satisfies the experimentally determined limit of ≤ 5 mm. Such small gradient scale lengths are required to explain large peaking factors. The best match to the experimental data is given by Case 2 and requires that G equals 600. Case 3 is for an edge temperature of $400\,eV$, which is the approximate energy of the runaway electron component measured in the ZT-40M plasma edge.

B. ZT-P. The small ZT-P device presently serves primarily as a prototype for the ZTH machine, as well as a test of shell-less operation. The ZT-P experiment is extremely interesting

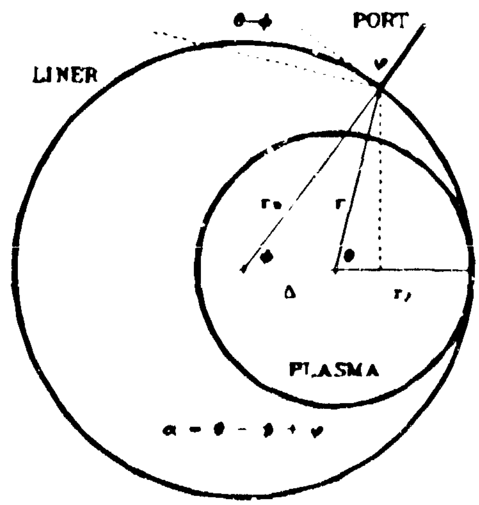

Figure 1. Geometry of the edge-plasma model used for ZT-40M and ZTH

TABLE II. WALL HEAT FLUX MODEL FOR ZT-40M

CASE:	ZT-40M[a]	1	2	3	4	5
Input:[a]						
G	--	1	600	185	6000	140
$\Delta(mm)$	3	3	3	3	3	6
$T(r_p)(eV)$	~40	40	40	400	40	40
$D^{(b)}$	--	1	1	1	10	1
Output:						
$\lambda(mm)$	≤ 5	3.76	0.15	0.16	0.15	0.32
$n(10^{19}m^{-3})$	1.0	25.4	1.0	0.1	0.1	2.2
ϕ(degrees)	--	50	11	11	11	11
q_{MAX}/q_0	14	3.1	14.1	14.0	14.1	13.9

(a) Other input: $r_w = 0.2\ m$, $R_T = 1.14\ m$, $f_{RAD} = 0.2$, $P_{TOT} = 7.2\ MW$.

(b) Tokamak edge-plasma models usually assume $D = 1\ m^2/s$.

from the viewpoint of plasma-wall interaction since $q_p = 1.1\ MW/m^2$, which is nearly the same flux as the TITAN reactor. The ZT-P device has 20% coverage of the liner with carbon armor tiles, but (without divertors) protection is insufficient to prevent strong plasma-wall interactions. Line radiation causes collapse of the plasma temperature and limits discharge lengths to such short times that testing shell-less operation is made difficult. The ability of the TITAN design to accommodate such high heat fluxes, suggests the divertor is one way of reducing the impurity problem, and divertor designs are presently being evaluated for ZT-P.

2. **ZTH.** The main goal of ZTH is to extend RFP currents and plasma temperatures sufficiently far to extend significantly the energy-confinement database. The limitations of ZT-40M associated with plasma-wall interactions are avoided by minimizing field errors, controlling the equilibrium position, and protecting the liner with 100% coverage with carbon tiles are important features of the ZTH design. Additionally, the average surface heat flux in the ZTH design is held to the ZT-40M level.

The model was applied[7] to ZTH parameters for a range of plasma shifts and for the two cases of G equal to 1 and 600. The $G = 1$ case corresponds to an ideal (i.e., no runaway electron) plasma and, therefore, determines a minimum peaking. The $G = 600$ case corresponds to a ZT-40M-like edge plasma and results in much higher heat flux peaking.

Values of the maximum peaking factor are calculated as a function of the plasma shift for the two values of G and are shown in Fig. 2. Small plasma shifts result in small peaking at large wall angles, while large shifts cause large peaking at small wall angles. For a 5-mm plasma shift the wall peaking factor is expected to be between 4 and 15, depending upon the level of anomalous parallel heat transport. Small plasma shifts, along with high density, low temperature, and low voltages to reduce electron runaways, minimize the wall heat flux.

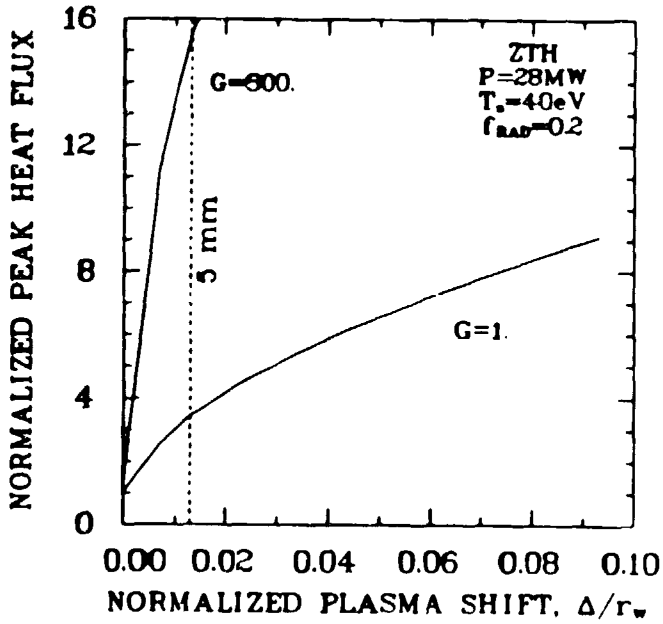

Figure 2. Maximum wall peaking factor versus plasma shift for ZTH.

In addition to demonstrating confinement scaling, ZTH with modification can address two other reactor-relevant issues. Liner and shell modifications to allow room for the addition of toroidal field divertor coils and a divertor plate would permit divertor operation which is essential to capitalize on the potential for high power density (Sec. III). Secondly, OFCD could be demonstrated with minimal wall interaction if the separatrix was fixed in space by appropriate feedback control of divertor coils.

III. STEADY-STATE DEVICES

The two steady-state, DT-burning devices considered here are the FTF[2] which is an ignition-class RFP, and the TITAN commercial reactor design.[3] Both devices capitalize on the high-power-density capability of the RFP to minimize the size and cost. The FTF while smaller than ZTH (Table I) has three times the heat flux, at $q_p = 2.3\ MW/m^2$. The TITAN reactor has an even higher flux of $4.9\ MW/m^2$. More detailed plasma modeling is necessary to describe the energy and particle flows from these DT plasmas, which at this higher flux will require considerably reduced peaking factors.

A. Plasma-Transport Models. The RFP plasma geometry is approximated by (a) a cylindrical plasma core with closed flux surfaces, and (b) a cylindrical-annulus plasma edge with open field lines that intersect a collector plate functioning as either a limiter or a divertor plate. Together, the edge plasma and the divertor plasma comprise the scrape-off layer (SOL). Because of the rapid parallel transport, the core-plasma transport reduces to a purely radial problem. The SOL, however, is inherently two-dimensional since both radial and ırallel transport determine the plate flux.

he core-plasma model RFPBURN[8,9] provides a time-de-ındent, two-fluid magnetohydrodynamic, radial description RFP transport based upon the formulation of Braginskii.[10] he temporal evolution of ion and alpha-particle densities, n and electron temperatures, and poloidal and toroidal agnetic fields is modelled. Core plasma transport and diation crucially affect SOL conditions. The radiation odel in RFPBURN, as well as the coupled 2-D edge-asma model (SOLAR) assumes a coronal equilibrium and specified constant impurity density. The transport model[2] ısumes a "soft" beta limit, wherein the losses adjust to aintain the poloidal beta, β_θ, near a critical value.

he SOLAR SOL-plasma model[15] couples two-fluid, 1-D ːscriptions of radial transport in the edge and parallel-ːld transport into a divertor to provide an approximation the 2-D geometry of interest. Parallel transport is ·iven by classical, collisional processes with free streaming mits on the energy flow and electric sheath boundary onditions. Particle and thermal radial diffusivities are given ıe following constant values: $D = 1$, $\chi_i = 0.2$, and $\chi_e =$ m^2/s, as inferred from tokamak divertor experiments. A 1- radial neutral-atom transport description (SPUDNUT)[12] ıtimates plasma/first-wall recycling, gas-puff refueling and ıarge exchange.

/hen the three 1-D plasma codes (core, radial edge, ıial edge) are coupled through continuity conditions, the ısultant model provides a self-consistent, time-dependent, ıdial and parallel profiles of the density, thermal energy and ıeir fluxes. The vacuum magnetics code, TORSIDO,[13] is ıed to compute the magnetic topology in the equatorial ane for use by the SOLAR code.

. **Results.**

FTF Poloidally symmetric pump limiters were examined ır FTF. Six limiters, providing 15% wall coverage, were ːsigned[2] to achieve a uniform surface heat flux of MW/m^2. The tolerances associated with machining the ːeal limiter surface and aligning the limiters, magnetic ›ils and plasma, as well as the unknowns associated with ːge plasm transport make it unlikely that the goal of MW/m^2 uniform heat flux could be achieved. In addition, ıort connection lengths, low edge/limiter recycle, and the ose proximity of the core plasma on closed field lines › the limiter make it difficult to obtain sufficiently low ːmperatures to keep sputtering rates at acceptable levels. he high edge density associated with high-recycle divertors otentially resolves this issue.

toroid·l cross-sectional view of a divertor is given in Fig. The field lines and SOL plasma in a closed divertor re directed between a narrow constriction in the vacuum hamber walls and into a large divertor chamber physically ıolated from the core plasma. The diverted plasma is then

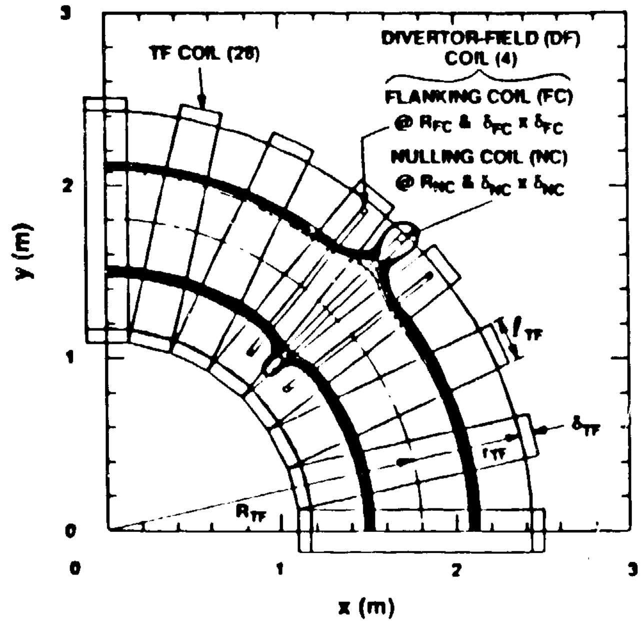

Figure 3. Magnetics geometry for toroidal-field divertors.

energy loss, and, therefore, degrades energy confinement. The RFP, then, permits large impurity fractions in the core and edge plasma, and entrainment of impurities in the divertor chamber, therefore, is not necessary; the divertor need only control impurities in the core plasma.

While an open divertor configuration does not entrain collected (neutralized) on a collector plate located far from both the core and the edge plasmas. The neutral particles are retained in the divertor chamber by a narrow throat at the divertor entrance until ionized. Plasma impurities are hopefully held in the chamber by the force of the plasma flow. The closed divertor, however, concentrated the heat flux through poloidal asymmetries and flux surface compression such that no practical steady-state design could be found for the FTF conditions.[2]

The open divertor configuration shown in Fig. 4 avoids the drawbacks of the closed divertor by moving the collector plate near the divertor throat. In this configuration the plate collects the heat and particle fluxes where magnetic flux surfaces are expanded, just prior (in minor radius) to the onset of the poloidally-asymmetric concentration of field lines. The open divertor geometry also allows room for a larger collector plate. The FTF open-divertor plate heat flux is lower than the closed divertor by a factor of 60. At $15.2 MW/m^2$, the plate flux is within a factor of ˙ of the target heat flux, this final reduction is achievable by increasing radiation losses.

As a result of the soft beta-limit assumption, to first order, RFP plasma parameters and the energy confinement are unaffected by the addition of impurities to the plasma. This behavior is in marked contrast to tokamaks in which injecting high-Z impurities increases both the radiation and the total

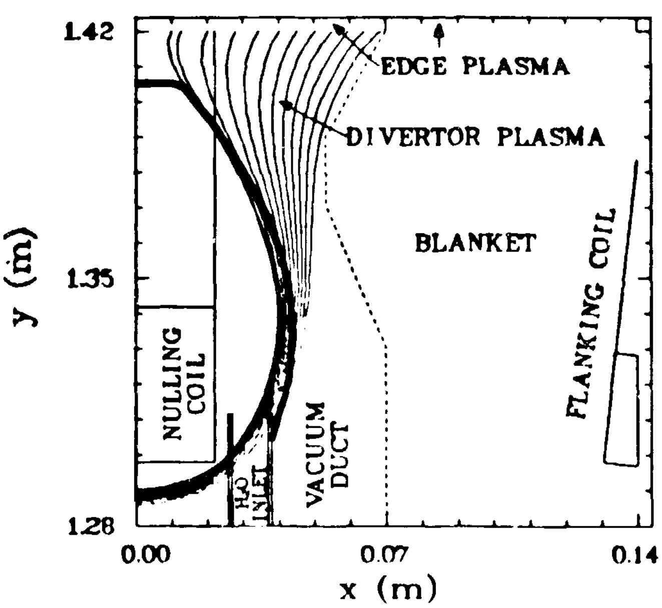

Figure. 4. Open divertor geometry in meters.

impurities, it must physically isolate the hot core plasma from the collector plate to protect both the plasma from neutral atoms and a possibly uncontrolled source of impurities, as well as to protect the plate from strong erosion. The minimum separation distance needed to isolate the collector plate from the core plasma is about four neutral-atom ionization mean-free-paths. Typical parameters for the plasma in front of the FTF collector plate are $T_e \sim T_n \sim 10eV$ and $n_e \sim 10^{21}m^{-3}$, which give a mean free path of $0.2mm$. The FTF divertor design locates the plate $15mm$ from the core plasma, which is 18 mean free paths. This large margin is sufficient to protect the plate from hot neutral-atoms which form by charge-exchange reaction either with the high-energy tail of the Maxwellian SOL ion distribution or with the core-plasma ions.

A precise recycling description, including flow reversal, requires a 2-D neutral-atom transport model applied to the particular geometry and plasma conditions of interest. The edge-plasma/plate recycling coefficient was chosen to describe a high-recycling "flow-reversal" regime[7] in which in the presence of a net SOL flow into the divertor, some plasma returns from the divertor to the edge plasma along field lines near the separatrix.

Radial edge-plasma and axial edge-plasma profiles for the open divertor are presented in Fig. 5. Summarizing the key parameters for the open-divertor design, peak edge-plasma density and temperatures are $n \simeq 1.2 \times 10^{20}m^{-3}$, $T_e \sim 100eV$, and $T_i \sim 250eV$, with a core- and edge-plasma radiation fraction of $f_{RAD}^{c} \sim 0.816$ and $f_{RAD}^{tot} \sim 0.925$, respectively. Plasma density and temperatures near a tungsten-coated plate are about $10^{21}m^{-3}$ and 10 eV, which result in negligible erosion and a transported heat flux normal to the plate of only about $1MW/m^2$.

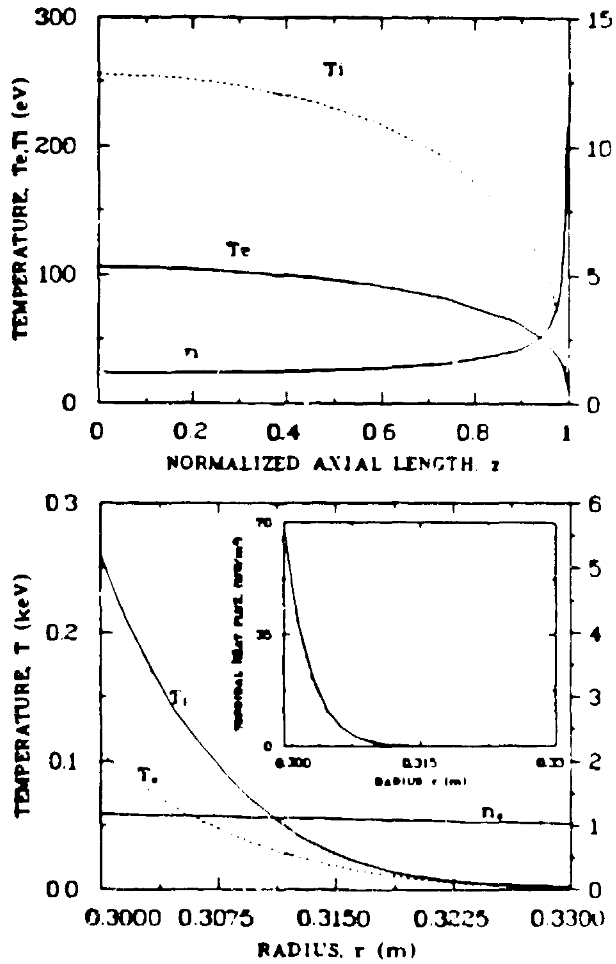

Figure 5. Radial and axial edge-plasma profiles for the open divertor for FTF.

Plasma density and temperatures at the first wall are $10^{20}m^{-3}$ and $1eV$, with a negligible transported heat flux and a radiation heat flux of about $2MW/m^2$. Wall erosion by plasma particles is negligible, erosion by charge-exchange neutral atoms, however, is $0.44mm/yr$. Increasing the SOL thickness from 30 to $40mm$ decreases the erosion rate by 30%. Changes which lower the separatrix temperature (e.g., lower heat flux or higher density) have a larger effect. A high-Z coating on the first wall would eliminate erosion concerns by raising sputtering thresholds.

A geometry calculation determines the shape of a plate (Fig. 4) which is located at some specified angle (8°) relative to a field line. This 8° angle is about the smallest that allows the plate to fit within the divertor chamber. The design was carried out on the inboard side of the torus (highest heat flux and the minimum space). Details of the locator algorithm and a characterization of the thermal-hydraulic properties is presented in Ref. 2.

C. TITAN. The TITAN commercial reactor study similarly concluded the need for an open divertor and a highly radiating plasma to maximize the reactor power density at acceptable heat fluxes. The actual design is similar to the FTF design, with a higher total radiation fraction ($f_{RAD} \sim 0.95$) and higher plate heat fluxes (8-10 MW/m^2) resulting from the higher average surface heat flux.

IV. SUMMARY

Wall protection schemes for existing, proposed, and conceptual RFPs have been reviewed. The ZT-40M and the proposed ZTH experiments are short-pulsed, devices with surface heat fluxes that allow operation with limiters or wall armor. The ZT-40M experiment has runaway electrons in the edge plasma which carry most of the energy and which combine with a small equilibrium shift to cause an estimated wall peaking factor of 14. Such peaking infers a radial gradient scale length for power of only 0.2 mm. Runaway electrons cannot be allowed in high-power-density devices like the FTF or TITAN, which will operate at higher density and lower drift parameter. The ZT-P device indicates that limiter operation does not protect the plasma in a device with high average heat flux since impurity line radiation quenches the discharge.

The reactor studies exploit the compact, high-power-density capability of RFPs to reduce size and cost. A toroidal-field, highly recycling "open" divertor for RFPs provides the optimal "conventional" wall protection scheme in that it provides the maximum power and particle handling capabilities. A toroidal-field divertor modifies only the minority field thereby leaving the main confining field unperturbed and also minimizing magnet ohmic power consumption. High recycle is necessary to maximize the edge density and minimize the edge temperature in order to reduce physical sputtering. An "open" configuration locates the divertor plate near the field null to take advantage of flux expansion and poloidal symmetries to minimize plate peak heat fluxes.

These recent findings of the importance of RFP divertors for high flux systems, including reactors, has prompted new studies of the potential for modifying ZT-P and ZTH to accommodate divertors within existing space limitations. The ZT-P device may need divertors to extend the discharge pulse length and permit study of shell-less operation. The ZTH device could be used to demonstrate the feasibility and to study RFP divertors as well as to isolate the plasma from the first wall during OFCD experiments.

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