Physics Basis for the Fusion Ignition Research Experiment (FIRE) Plasma Facing Components*

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Abstract
A design study of a Fusion Ignition Research Experiment (FIRE) is underway to investigate and assess near term opportunities for advancing the scientific understanding of self-heated fusion plasmas. The emphasis for the FIRE program is on understanding the behavior of plasmas dominated by alpha heating ($Q \geq 5$). Study activities have focused on the technical evaluation of a compact, high field, highly shaped tokamak. One of the key issues for the design is to find suitable plasma facing components (PFCs). We have investigated a variety of plasma edge and divertor conditions ranging from reduced recycling high heat flux conditions (attached) to reduced heat flux detached operation. The inner divertor detaches easily while impurities must be added to the outer divertor to achieve detachment. The outer divertor and private space baffle will have to be actively cooled. The plasma-facing surface of the divertor is tungsten bonded to a CuCrZr heat sink. The remainder of the PFCs are beryllium coated copper attached to the vacuum vessel. Plasma current disruptions impose strong constraints on the design. Appreciable PFC surface melting and evaporation and onset of "plasma shielding" are expected. The forces induced on the PFC due to disruptions determine the size of the attachment of the PFC to the vacuum vessel.

1 Introduction

A design study of a Fusion Ignition Research Experiment (FIRE) [1] is underway to investigate and assess near term opportunities for advancing the scientific understanding of self-heated fusion plasmas. The emphasis is on understanding the behavior of plasmas dominated by alpha heating ($Q \geq 5$) that are sustained for a duration comparable to characteristic plasma time scales ($\geq 10\tau_E$, $\sim \tau_{skin}$). FIRE will be a stepping stone between present tokamaks and an attractive Engineering Test Reactor. The programmatic mission of FIRE is to attain, explore, understand and optimize alpha-dominated plasmas to provide knowledge for the design of attractive magnetic fusion energy systems. The program strategy is to access the alpha-dominated regimes with confidence using the present tokamak data base (e.g., Elmy-H-mode, ≤ 0.75 Greenwald density) while maintaining the flexibility for accessing and exploring advanced tokamak modes at lower fusion power for longer duration in later stages of the experimental program. A major goal is to develop a design concept that can meet these objectives with a construction cost in the range of $1B.

Activities have focused on the technical evaluation of a compact, high-field, highly-shaped tokamak with parameters: R_0 = 2m, a = 0.525m, $\kappa_{95} \approx 1.8$, $\delta_{95} \approx 0.4$, $q_{95} > 3$, double-null-divertor, $B_T(R_0)$ = 10T, and I_P = 6.44 MA and fusion-burn flat-top time ~ 20s (30 τ_E and >1 τ_{skin}). One of the key issues for the design is to find suitable plasma facing components. We have investigated a variety of plasma edge and divertor conditions ranging from reduced recycling high heat flux conditions (attached) to reduced heat flux detached operation using the UEDGE boundary and divertor modeling code. The predicted peak heat flux on the divertor is 5 to 25 MW/m^2 with the higher values corresponding to attached operation. The inner divertor easily enters detached operation while impurities have to be added to the outer divertor to achieve detachment.

The outer divertor and private flux region baffle will have to be actively cooled. The plasma-facing surface of the divertor is tungsten bonded to a CuCrZr heat sink. The remainder of the PFCs are beryllium coated copper attached to the vacuum vessel. Plasma current disruptions also impose strong constraints on the design. The energy deposition on the plasma facing components ranges from 4 to 96

* Work supported by DOE Contract # DE-AC02-76CHO3073.

MJ/m^2. Appreciable PFC surface melting and evaporation and onset of 'plasma shielding' is expected. Surface melting and evaporation have been studied using the A*THERMAL and SPLASH codes. The forces induced on the PFC due to disruptions determine the configuration and size of the attachment of the PFC to the vacuum vessel. We will discuss the impact of the variety of physics operating conditions on the PFC design.

2 Divertor Design Requirements

The FIRE device is designed for high power density and advanced physics operating modes. Accordingly, the divertor must accommodate the high elongation and high triangularity plasma needed for advanced physics modes. Owing to space constraints, the divertor geometry must be relatively open with short distances from the x-point to the plate and considerable spreading of the field lines. Connection lengths are short and the scrape-off layer (SOL) thickness is small. Without a radiative divertor the heat loads are high ~25 MW/m^2.

The divertor plate geometry is shown in Figure 1. The outer divertor plate is at an angle of 30° with respect to the flux lines. The inclination is driven by the flux surface spreading close to the X-point. The inner divertor plate is nearly normal to the field lines. The slot between the outer divertor plate and the baffle provides for pumping plasma exhaust particles.

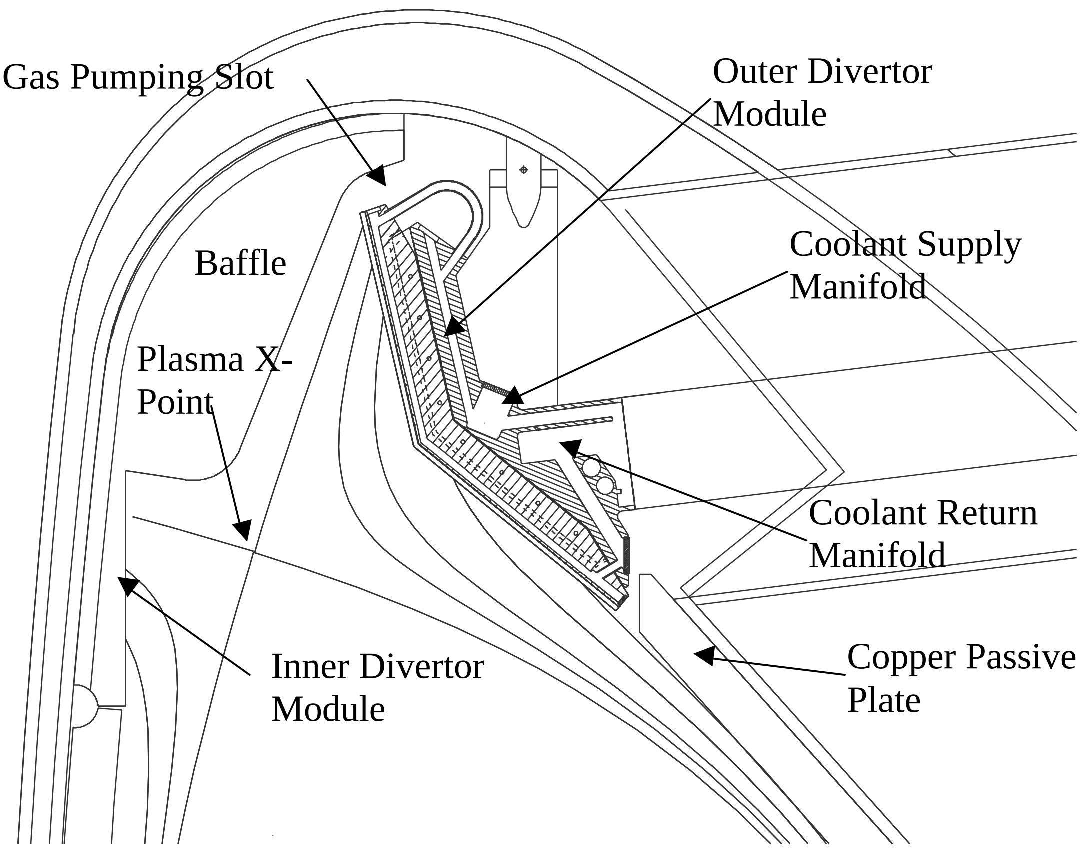

Figure 1. FIRE Divertor showing the passive plates, inner divertor plate, baffle and outer divertor plate.

The FIRE device is a very compact machine with high current and high magnetic field. The connection length along the field lines from the outer mid-plane to the divertor plate determines how much time is available for plasma energy to diffuse across the field while being transported to the divertor. The shorter the connection length, the narrower the scrape-off length. Using the magnetic field data from the equilibrium code, the connection length is 13.1 m for FIRE. Of that length, 7.4 m is from the mid-plane to the divertor throat and 5.7 m in the divertor. These distances are to be compared to 125 m total length in ITER. Accordingly, we expect a narrower scrape-off-layer in FIRE.

3 Operating Modes

Divertor component power flows were considered for four cases: (1) the baseline D-T operating mode (10 T, 6.6 MA, 18 s) with a plasma heating power of 70 MW; (2) an advanced physics D-D operating mode (4 T, 2 MA, 215 s) with a plasma heating power of 16 MW; (3) a long-burn D-T mode (8 T, 5.5 MA, 31 s) with a plasma heating of 45 MW; and (4) high field operation (12T, 8MA, 12s) with plasma heating of 75 MW. Assumptions concerning the distribution of these total exhaust powers are: 20% is radiated from the main plasma, 20% is radiated from the scrape off layer with all being deposited on the baffle and divertor throat, 20% is deposited on the inner divertor plate, and the remainder goes to the outer divertor plate.

4 Edge Plasma Modeling

The UEDGE code [2] was used to calculate the expected edge conditions in FIRE. Four Cases having various combinations of thermal and particle diffusivity were examined. For all cases, the power into the scrape-off layer was 28 MW and the separatrix density was 1.5 x 10^{20} /m^3 with a wall recycling coefficient of 1.0. Three different values of the particle and heat diffusivity were considered. The parameters in Case C duplicate edge plasma data from existing machines the best and were the conditions used for the ITER design. The divertor plate was kept perpendicular to the field lines for most cases. Case D is the same as Case C with the divertor plate tilted as in the baseline design and with 10^{21} particles/sec pumping. The conditions for the various Cases are shown in Table 1.

Table 1. Transport parameters used for modeling the FIRE edge plasma.

Case	Description	Thermal diffusivity (m^2/s)	Particle diffusivity (m^2/s)
A	High Conductivity	1.5	1.0
B	ITER Conductivity	0.5	1.0
C	Bohm like diffusivity	0.5	Dbohm + 0.1
D	Tilted plates and pumping	0.5	Dbohm + 0.1

NOTE: Dbohm = Te/16 eB

The results are shown in Table 2. The peak heat flux is less than 25 MW/m^2 for all cases. The outer divertor is not detached under any of the conditions considered. Additional gas will have to be added to get the outer divertor to detach.

Table 2. Results of UEDGE modeling of the FIRE edge plasma

Case	Te_m (eV)	λ_m (cm)	Te_p (eV)	ne_p ($10^{21}/m^3$)	Q_p (MW/m^2)	λ_p (cm)
A	106	0.8	1.5	61	5.7	6.5
B	152	0.6	15	44	25	1.8
C	138	0.7	14	43	23	2.3
D	138	0.7	13	52	19	2.5

The UEDGE Code has been used to study the effect of adding beryllium only or beryllium and neon to the edge plasma to stimulate detachment of the plasma in the outer divertor channel. The divertor plates were placed at the proper angle relative to the field lines for these calculations. The particle diffusivity and thermal conductivity had to be reduced on the small radius side of the plasma to achieve a single solution. One expects that the transport will be reduced on the small radius side of the plasma because of the good curvature in that region (this is consistent with the observations of less power transport to the inner divertor in a double null configuration).

The inner divertor is easily detached. With no impurity addition to the inner divertor the heat flux to the plate is about 1 MW/m^2 from particle transport and 1.8 MW/m^2 from hydrogen radiation. We find that the maximum heat flux on the inner divertor is 3 MW/m^2.

The results for the outer divertor with the angled plates are very similar to the results with the plate normal to the field lines (26 MW/m^2). When Be is added to the divertor region, the peak heat flux is reduced to 20 MW/m^2 with about 5 MW/m^2 of radiated power that peaks at a different location from the peak particle heat flux. There was no detachment with the addition of Be alone. With Ne injection, the plasma could be detached from the divertor plate. For 4.1 Pa m^3/s (31 Torr l/s) Ne injection there was no detachment but the peak heat flux was reduced to 15 MW/m^2. With 4.7 Pa m^3/s (35 Torr l/s) Ne injection, the plasma did detach from the divertor plate but the solution evolved toward an x-point MARFE. It is clear that the amount of Ne injected into the divertor needs to be controlled. The peak heat load on the outer divertor or baffle is 6 MW/m^2 for the radiative solution.

4.1 Disruption Heat Loads

The database on tokamak disruptions [3] has been used to estimate the energy deposition on the PFCs. Two phases have been identified for disruptions; the thermal quench phase when the plasma stored energy is lost to the divertor and the current quench phase when the plasma current decays and the magnetic stored energy is lost to the first wall. We have assumed the plasma-thermal energy is 33 MJ and that all of this energy is deposited on the divertor plates. There is a wide range of possible parameters describing disruption energy deposition, so the energy deposition is specified as a range of possible values. The wide range arises because of incomplete understanding of disruption deposition on existing devices, variation in the deposition observed, and uncertainties in the extrapolation to FIRE conditions. The values specified for the disruption analysis are shown in Table 3. The low-end and high-end columns represent the extreme of all the uncertainties. The column labeled most likely is what we believe to be the expected upper limit on energy deposition. The reference column is what we expect for a typical full power disruption.

Table 3. Disruption energy deposition on the divertor plates

	Low End	Most Likely	Reference	High End
Inner Divertor	8 MJ/m^2	31 MJ/m^2	13.4 MJ/m^2	96 MJ/m^2
Outer Divertor	4MJ/m^2	16 MJ/m^2	6.8 MJ/m^2	48 MJ/m^2

The magnetic stored energy is radiated to the first wall during the current decay. The stored magnetic energy in the FIRE reference plasma is 35 MJ. The expected minimum current decay time is 2-6 ms. The average energy deposition on the first wall is 0.5 MJ/m^2. For a toroidal peaking factor of 2:1, the peak energy deposition is 0.67 MJ/m^2. This is enough energy to melt 0.12 mm of Be if all the energy goes into melting. Thermal conduction and radiation will reduce the amount of melting. This reduction should give an adequate lifetime for the first wall but further modeling is required to confirm the details of the melt depth and possible loss mechanisms.

4.2 Disruption Forces

The 2 ms duration of the current disruption was used to estimate the eddy currents induced in the divertor structures for the case of a stationary plasma disruption. The maximum current decay rate is assumed to be 3 MA/ms. The L/R time for the divertor plates is longer than the disruption time. Purely inductive solutions for the modules yield forces of 1.9 MN (outer) and 2.8 MN (inner). These forces determine the size of the divertor supports and backplate.

5 Analysis of Erosion

The erosion of the W plasma facing material due to normal plasma operation has been assessed using the REDEP/WBC erosion code package [4]. The plasma temperature and density profiles, and charged and neutral particle fluxes to the divertor plates from UEDGE/DEGAS2 [5, 6] were used as input to REDEP/WBC to calculate the sputtered tungsten transport. The codes include the sputtered atom velocity distribution, electron impact ionization, Lorentz force motion, magnetic and Debye dual- structure sheath, impurity-plasma charge changing and velocity changing collisions.

The results (for attached plasmas) predict essentially zero net erosion of the divertor plate and no plasma contamination. This is due to the short ionization mean-free path for W (~ 25 µm) and high sheath acceleration and/or collisional induced flow back to the plate. The gross sputtering (~ 3 nm/s) is mostly due to impurity sputtering (due to 0.1% O impurity) and self-sputtering. Analysis is underway to compute erosion of the Be wall surfaces and the subsequent mixed Be/W divertor surface, and for other plasma cases.

5.1 Disruption Erosion

The HEIGHTS computer code package [7] was used to model the erosion of plasma facing components owing to disruption energy deposition. The code package includes the effect of plasma-target interactions, plasma-debris interactions, photon radiation and transport, and plasma-melt layer interaction. A typical result for $10 MJ/m^2$ deposition in 1 ms shows that melting starts about 10 µs after the disruption thermal quench starts, and vaporization starts about 20 µs later. Once vaporization starts there is a strong reduction in the heat flux because of interaction between the plasma and the atoms in the vapor (vapor shielding). Because of vapor shielding, the amount of melted and eroded material is only weakly dependent on the energy deposited. The amount of vaporized material increases by about a factor of two due to a ten-fold increase in energy deposition. The melt layer is predicted to be 150 to 200 µm thick and 2-4 µm is predicted to evaporate due to a disruption. The loss of material due to disruptions determines the lifetime of the divertor plates.

6 Conclusions

The challenge of designing plasma-facing components for a compact, high-power density tokamak with high elongation and triangularity is the same problem that must be solved for a tokamak fusion reactor. We have shown that the requirements can be met by using tungsten and beryllium as plasma facing materials. The outer divertor and baffle must be actively cooled, but the inner divertor and first wall only need to be cooled by conduction to the vacuum vessel liner. The lifetime of the divertor plates will be determined by disruption erosion. Further study is required to quantify the likely lifetime.

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