DRAFT

Power and Particle Handling Solutions for the Fusion Ignition Research Experiment (FIRE)*

M.A Ulrickson[1], C. Baxi[2], J. Brooks[3], D. Driemeyer[4], A. Hassenein[3], C. E. Kessel[5], B. E. Nelson[6], T. Rognlein[7], J. C. Wesley[2], D.L. Youchison[1]

[1]Sandia National Laboratory, Albuquerque, NM 87185, USA
[2]General Atomics, San Diego, CA 92186, USA
[3]Argonne National Laboratory, Argonne, IL 60439 USA
[4]Boeing Corp., St. Louis, MO 63166 USA
[5]Princeton Plasma Physics Lab., Princeton, NJ 08543 USA
[6]Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
[7]Lawrence Livermore National Laboratory, Livermore, CA 94551, USA

Abstract

The programmatic mission of the Fusion Ignition Research Experiment (FIRE) is to attain, explore, understand and optimize alpha-dominated plasmas to provide knowledge for the design of attractive magnetic fusion energy systems. The activities have focused on the technical evaluation of a compact, high-field, highly-shaped tokamak with the parameters: $R_0 = 2$m, a = 0.525m, $\kappa_{95} \approx 1.8$, $\delta_{95} \approx 0.4$, $q_{95} > 3$, double-null-divertor, $B_T(R_0) = 10$T, and $I_P = 6.44$ MA and flat top time ~ 20s (25 τ_E and ≥1 τ_{skin}). One of the key issues for the design is to find suitable plasma facing components (PFC). We have investigated a variety of plasma edge conditions ranging from detached divertor operation to reduced recycling high heat flux conditions. We discuss the impact of this variety of physics operating conditions on the PFC design. The predicted peak heat flux on the divertor is 5 to 25 MW/m^2. The inner divertor easily enters detached operation while impurities have to be added to the outer divertor to achieve detachment. Because of concerns about retention of tritium in carbon-based materials, we have adopted an all-metal plasma facing material baseline for the device. Tungsten was chosen as the plasma facing material for the divertor components. A divided surface is used to reduce the thermal stresses in the heated surface and mitigate the effect of disruption melting. The outer divertor and baffle are actively cooled while the inner divertor is cooled between shots. Plasma current disruptions also impose strong constraints on the design. The forces induced on the PFC due to disruptions determine the attachment of the PFC to the vacuum vessel. Beryllium was chosen as the surface of the first wall because it has a low atomic number (Z) and strong oxygen gettering capability. The lifetime of the Be first-wall is determined by disruption erosion. Particle control is achieved through the use of cryo-pumps in a duct behind the outer divertor. We show that this choice of an all-metal design simplifies machine design, minimizes plasma contamination, and minimizes tritium inventory.

Introduction

A design study [1,2,3] 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 is on understanding the behavior of plasmas dominated by alpha heating (Q ≥ 5) that are sustained for a duration comparable to characteristic plasma time scales (≥ $10\tau_E$, ~ τ_{skin}). 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 activities have focused on the technical evaluation of a compact, high-field, highly-shaped tokamak with the parameters: $R_0 = 2$m, a = 0.525m, $\kappa_{95} \approx 1.8$, $\delta_{95} \approx 0.4$, $q_{95} > 3$, double-null-divertor, $B_T(R_0) = 10$T, and $I_P = 6.44$ MA and flat top time ~ 20s (25 τ_E and ≥1 τ_{skin}). One of the key issues for the design is to find suitable plasma facing components (PFC). We have investigated a variety of plasma edge conditions ranging from detached divertor operation to reduced recycling high heat flux conditions. We discuss the impact of this variety of physics operating conditions on the PFC design. The predicted peak heat flux on the divertor is 5 to 25 MW/m^2. The inner divertor easily enters detached operation while impurities have to be added to the outer divertor to achieve

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

detachment. Because of concerns about retention of tritium in carbon-based materials, we have adopted an all-metal plasma facing material baseline for the device. Tungsten was chosen as the plasma facing material for the divertor components. A divided surface is used to reduce the thermal stresses in the heated surface and mitigate the effect of disruption melting. The outer divertor and baffle are actively cooled while the inner divertor is cooled between shots. Plasma current disruptions also impose strong constraints on the design. The forces induced on the PFC due to disruptions determine the attachment of the PFC to the vacuum vessel. Beryllium was chosen as the surface of the first wall because it has a low atomic number (Z) and strong oxygen gettering capability. The lifetime of the Be first-wall is determined by disruption erosion. Particle control is achieved through the use of cryo-pumps in a duct behind the outer divertor. We show that this choice of an all-metal design simplifies machine design, minimizes plasma contamination, and minimizes tritium inventory.

Unique Divertor Features of a Compact High Field Tokamak

Plasma characteristics needed to reach high performance

The activities have focused on the technical evaluation of a compact, high field, highly shaped tokamak with the parameters shown in Table I. The plasma performance of FIRE was estimated using a zero-dimension analysis with energy confinement given by ITER IPB98(y) [4]for the Elmy H-mode, $\beta_N \leq 2.5$, density $\leq$ 0.75 Greenwald density, $P_{threshold} \geq (0.9/A_i)n^{0.75}BR^2$, 3% Be impurities and alpha ash accumulating self consistently with $\tau_{He} = 5\ \tau_E$. For the baseline FIRE parameters (6.44 MA/10T), the alpha heating fraction, f_α, rises from ≈ 40% at the low end of the present database to ≈ 80% at the high end. A range of advanced tokamak modes could then be studied on FIRE. If the H-mode multiplier (HH98(y,1)) is about 1.2 and $\beta_N \approx 3$ can be attained and sustained at 6.5 MA, then inductively-driven plasmas with Q >> 10 could be achieved at full field for durations of ~ 20 s(~ $1\tau_{skin}$).

Table I. Design Goals for FIRE

R (m), a (m)	2.0, 0.525	flattop time (s)	~20(12)*
κ_{95} , δ_{95}	≈1.8, ≈ 0.4	alpha heating fraction	>0.5
q_{95}	> 3	τ_E, τ_{skin} (s)	~ 0.6, ~ 13
$B_t(R_o)$ (T)	10(12)*	Z_{eff} (3% Be + He (5 τ_E))	1.4
Wmag TF (GJ)	3.7	Fusion Power (MW)	~ 200
I_p (MA)	6.44(7.7)*	ICRF Power (MW)	30

*** Upgrade capability**

Implications for divertor design

The parameters of the machine that most strongly impact the divertor design are the high elongation, high triangularity, and high power density. High triangularity forces the inner divertor leg to be very short while the outer divertor target is nearly vertical, which prevents a very shallow angle of incidence for the heat flux. High elongation limits the depth of the outer divertor leg because of increased stresses in the TF coils as the vacuum vessel is made taller. The high power density forces active cooling of the outer divertor leg and baffle even for the modest pulse lengths of FIRE. High-density operation offers the advantages of a high recycling divertor and higher neutral pressure in the divertor, which makes particle and ash control easier. The design of the FIRE divertor is shown in Figure 1.

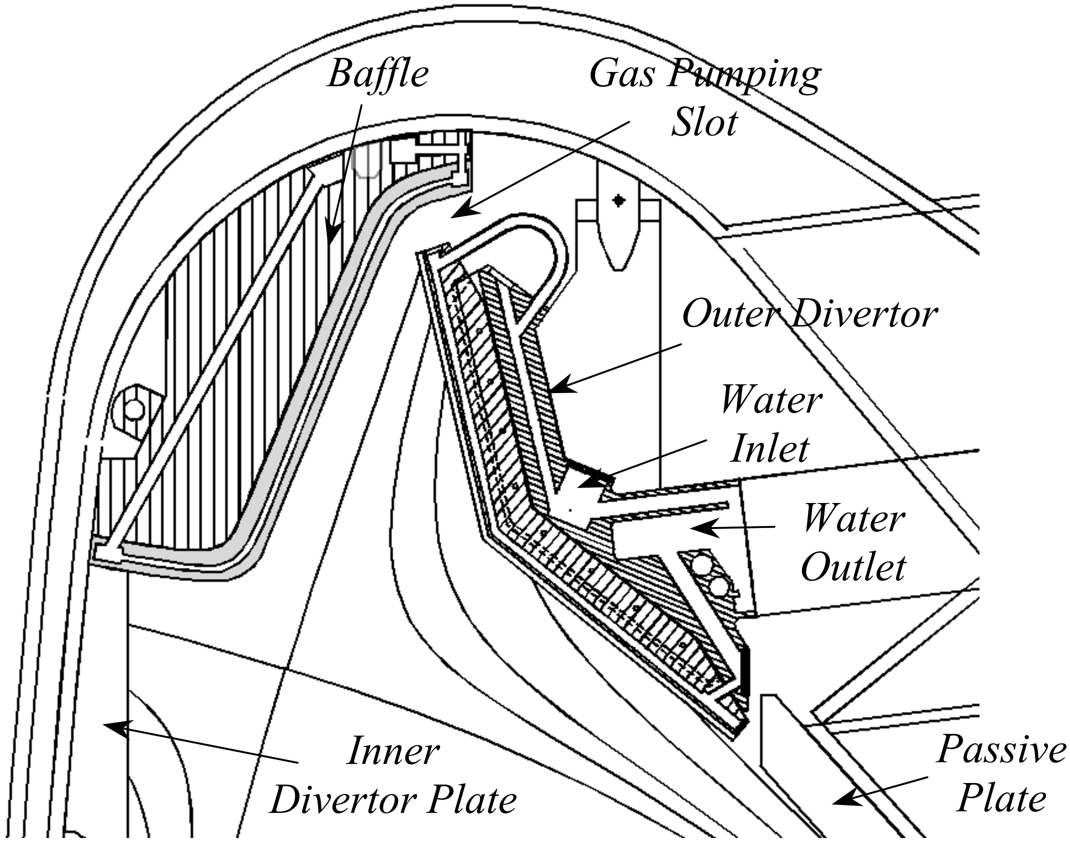

Figure 1. The design of the FIRE divertor plates is shown in the vacuum vessel. The plasma flux surfaces are shown for reference.

Modeling of the Plasma Edge

High recycling conditions

The UEDGE code [5] was used to calculate the expected edge conditions in FIRE. For all cases considered 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. The particle diffusivity (D) was varied between 1.0 m^2/s and D_{bohm} +0.1 m^2/s (D_{bohm} = Te/16 eB). The thermal diffusivity (χ_e) was varied between 0.5 and 1.0 m^2/s. The edge plasma data from existing machines are best matched with χ_e = 0.5 and D = D_{bohm} +0.1 (these conditions were used for the ITER design). The modest tilt of the divertor plates in FIRE did not significantly change the results from runs with the plates perpendicular to the field.

Radiative divertor conditions

The UEDGE Code has been used to study the effect of adding 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 from the plate. 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 used 3 MW/m^2 for the heat flux on the inner divertor.

When Be (3%) is added to the divertor region, the peak heat flux on the outer divertor is reduced by about 20% with about 5 MW/m^2 of radiated power located at a different location from the peak particle heat flux. There was no detachment with just Be addition. With Neon 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 [6]. The radiated power is 80 MW/m^3 in the MARFE region. It is clear that the amount of Ne injected into the divertor needs to be controlled, but we do know the range of injection that is needed. A scheme for feedback control of the Ne injection will have to be developed. We can control the amount of Ne to limit the heat flux on the outer divertor. Reduction of the heat load by added radiation will increase the engineering margin in the divertor.

Predicted heat loads and distribution

The peak heat flux was between 6 and 25 MW/m^2 with most of the results between 20 and 25 MW/m^2. The scrape-off length for power at the divertor plate was 18 to 25 mm (the scrape-off length for power at the mid-plane was 6-8 mm). Figure 3 shows the distribution of heat flux on the outer divertor plate. Divertor plasma temperature was about 15 eV and the density was 4.3 to 5.2 x $10^{22}/m^3$. The outer divertor is not detached under any of the conditions considered. Additional gas had to be added to the model to get the outer divertor to detach (see above).

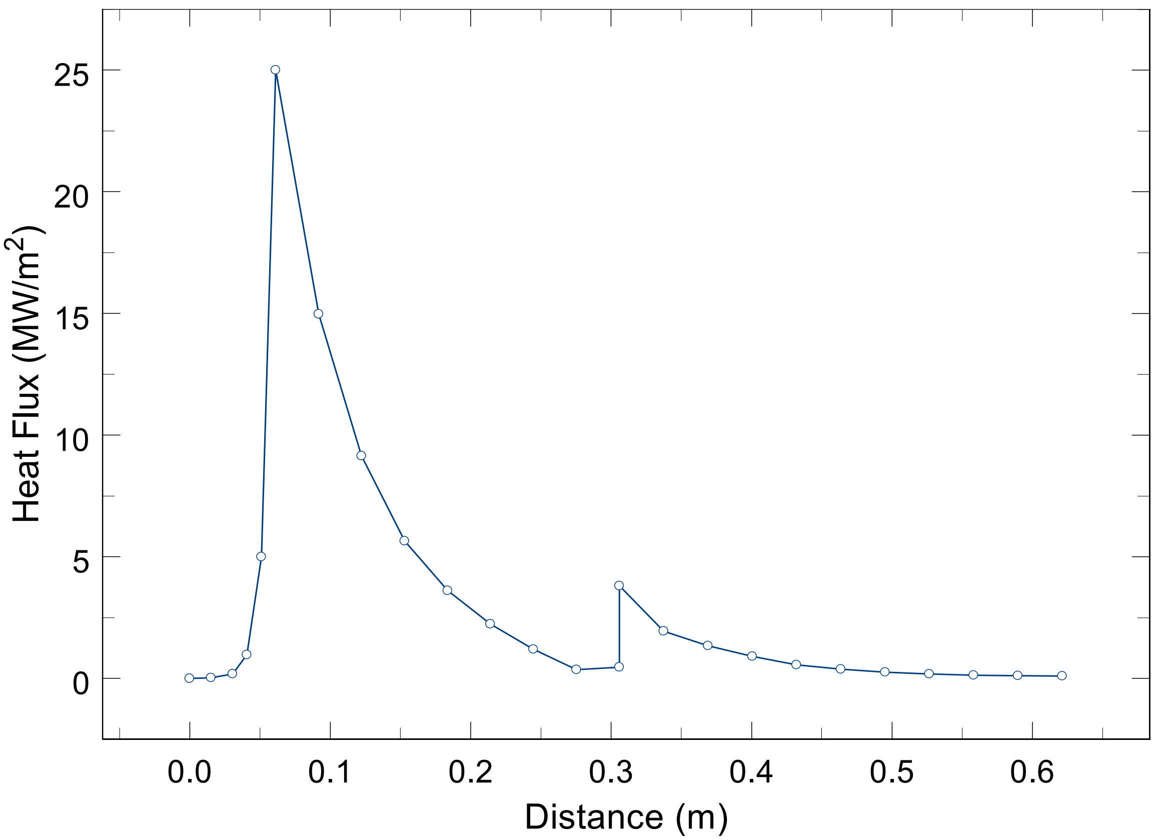

Figure 3. The heat flux on the outer divertor plate is shown as a function of the distance along the plate. The jump at 0.3 m is where the bend in the divertor plate is located.

Disruption effects

Analysis of the thermal response of the divertor to disruption heat loads has been reported previously [6,7]. The lifetime of the divertor plates and the first wall Be tiles are determined by disruption erosion and the fraction of the melted layers removed during the disruption. The 7 mm thickness chosen for the plasma facing material is a compromise between thinner layers for good heat removal during normal operation and sufficient thickness for disruption erosion.

Plasma current disruptions were modeled using the Tokamak Simulation Code (TSC) [8]. Three disruption cases have been considered. A condition where the plasma does not move during the disruption (SD case) and the current decay rate is 3 MA/ms (this implies that a 6.5 MA plasma will decay in 2.2 ms). If the plasma maintains vertical stability during the disruption, the plasma will be pushed into the inner wall because of the sudden loss of plasma pressure (RD case). Finally, a vertical disruption event was considered where the plasma drifts vertically with no loss of plasma current then disrupts while near one of the divertors (VDE case). The eddy currents induced in the divertor plates are most severe for the VDE case.

The TSC output consisted of the temporal variation of the current in over 900 filaments representing the plasma. The PC-Opera code (a product of VectorFields) was used to compute the eddy currents induced in the divertor plates, passive plates and the vacuum vessel shells. The model included all of the poloidal field coils. A one sixteenth segment of the vessel and other nearly axisymmetric components were used, but the actual size and shape of the divertor plates and support structure were modeled. The results showed that the passive plates carry substantial toroidal current during the disruption. Those currents reduce the currents induced in the divertor plates by about a factor of three compared to a model where the divertor plates are modeled without any surrounding conductors. Figure 4 shows the distribution of currents due to a VDE. The details of the circulating current in the outer divertor plate are more clearly seen in Figure 5. Currents in the baffle are circulating through the thickness of the structure rather than in the face as in the outer divertor (see Figure 6). We are investigating techniques to reduce the currents in the baffle by breaking the loop.

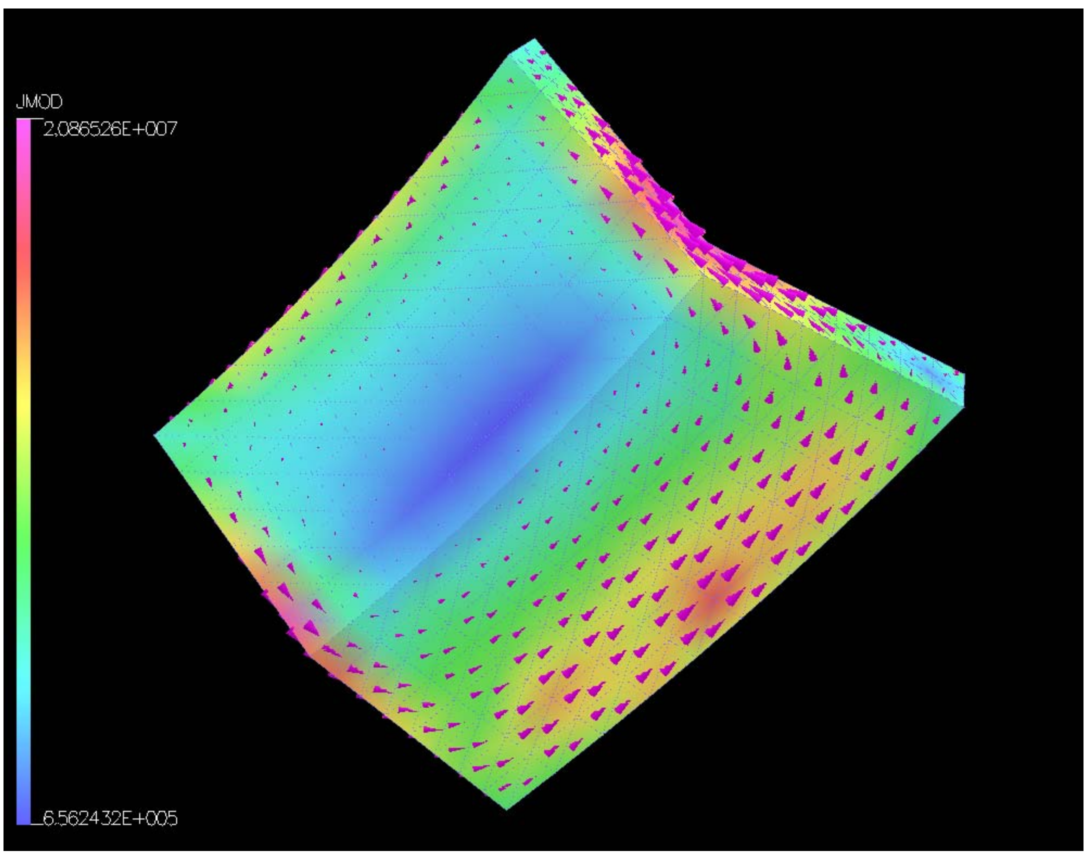

Figure 4. The distribution of eddy currents is shown in the outer divertor plate during a vertical disruption event. Note that most of the current flows in the outer edges of the

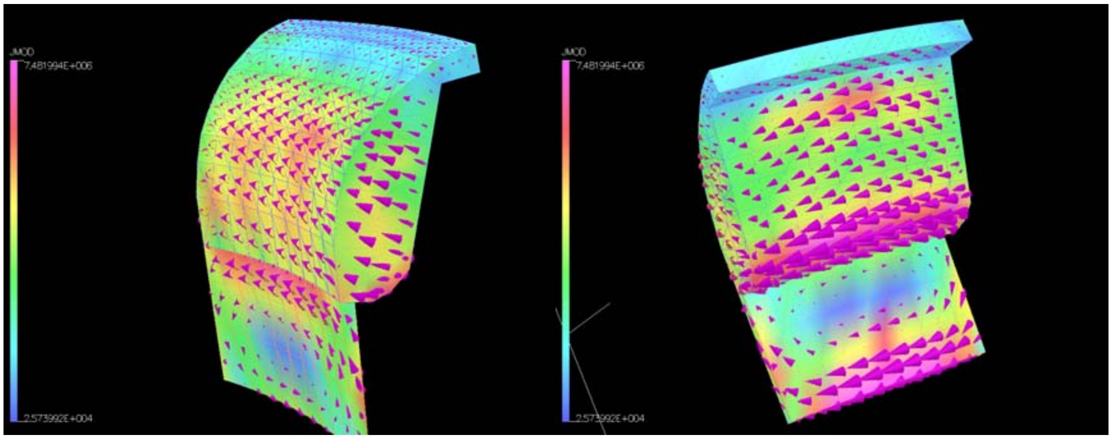

Figure 5. The distribution of eddy currents is shown in the inner divertor and baffle during a vertical disruption. Note the circulation loop in the lower part of the baffle.

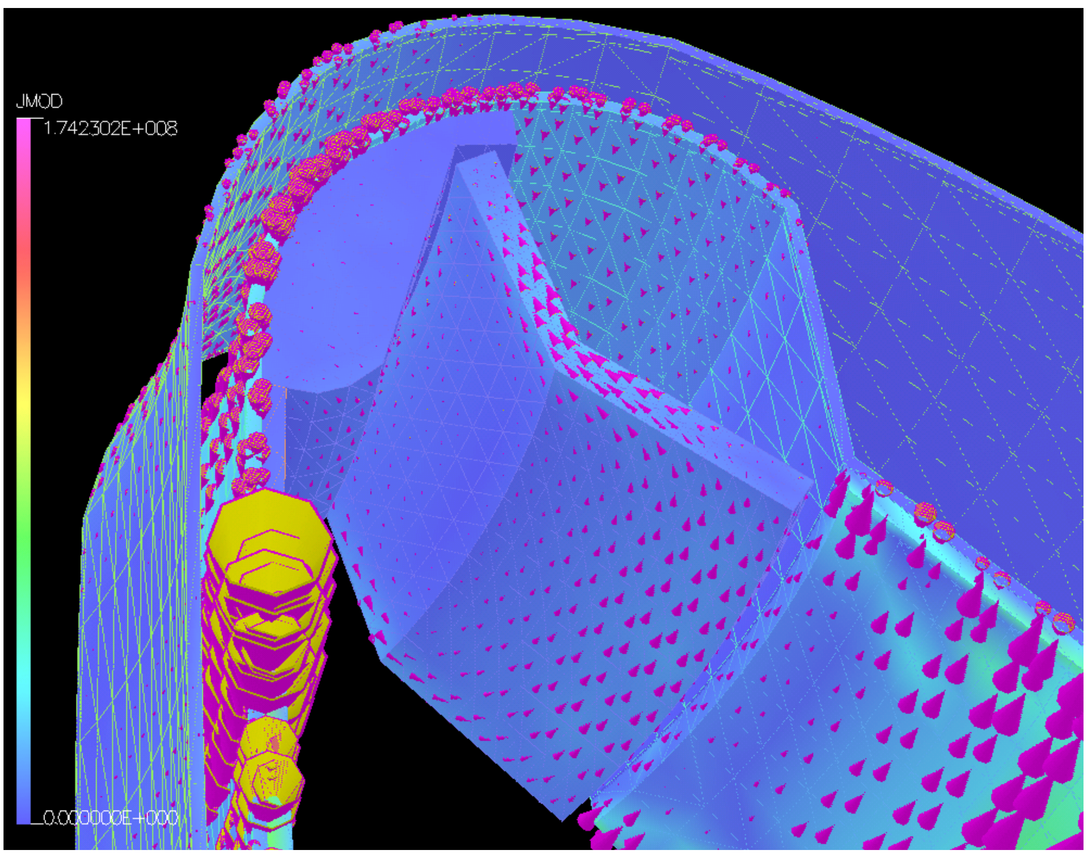

Figure 6. The distribution of eddy currents in the vacuum vessel and passive plates are shown during a vertical disruption event. Note the large currents in the inner passive plates.

Divertor Design

High-density operation in a compact high field device implies very high-density plasma in the closed divertor chosen for FIRE. Particle pumping is simplified because of the high neutral pressure (see the next section). The closed configuration also helps to confine the impurities generated from the divertor plates and gas injected to increase radiation in the divertor. The closed geometry with increased radiation does require cooling of the baffle opposite the outer divertor plate, but this complication does not offset the advantages.

The high heat flux surfaces are constructed from tungsten rods 3 mm in diameter. The rods are embedded in a copper heat sink. The copper has water-cooling channels containing swirl tapes to enhance the heat removal. Results of high heat flux testing have shown that this design can be used up to 25 MW/m^2 [9, 10, 11]. The use of a double null divertor configuration and high-density operation reduces the power load on the inner divertor to the point where the plasma there detaches from the plate with hydrogen radiation alone. Since the outer divertor plate must be actively cooled, we will use the power in the cooling water to balance the power split between the two divertors.

Pins on the vacuum vessel shell are used to secure one end of the outer divertor plate. A latch consisting of a sliding pin and lugs on the vessel is actuated through the vacuum pumping duct and secures the opposite end of the divertor. The baffle is attached in a similar fashion but the latching pin is accessed from inside the vessel. We plan to remove each segment of the divertor (1/16th) separately. A wedge shaped bar and bolts are used to hold the inner divertor and first wall tiles to the water-cooled copper liner in the vessel. Remote handling tools will be developed to perform these operations. The operations required are similar to other remote handling operations and no unusual problems are anticipated.

Particle Control

Requirements for He ash control

The plasma must be fueled at a rate consistent with the particle confinement time and the line average density. The total particle content of the FIRE plasma is about 10^{22} particles. The energy confinement time required to reach the performance goals is 0.5 to 0.8 sec. We have assumed that the particle containment time is twice the energy containment time (note $\tau_p = 5\tau_E$ in the modeling of the plasma performance). Taking the worst case, we estimate the fueling rate needs to be 2 x 10^{22} DT/sec (38 Pa m^3/sec). At 200 MW of fusion power, the DT consumption rate is 10^{20}/sec. To avoid the effects of He ash buildup, the He concentration should be kept below about 2%. The particle-pumping rate is given by

$$PR = \frac{1}{W_R}\left(\frac{DT}{f_{He}}\right) + FR \tag{1.1}$$

where DT is the burn-up rate, f_{He} is the He fraction, W_R is the wall recycling coefficient, and FR is the fueling rate discussed above. The worst case pumping speed estimate is 3 x 10^{22}/sec (56 Pa m^3/sec). The pumping and fueling systems have been designed to meet these requirements.

Particle pumping features

Neutral particles created near the strike point on the outer divertor are allowed to escape behind the divertor component and enter the angle ports on the top and bottom of the machine. The calculated neutral pressure is 10 to 50 mTorr (1.3-6.7 Pa). Viscous drag effects increase the pumping speed of the duct. We are able to use a combination of cryo-condensation and viscous drag turbo-pumps to remove both hydrogen isotopes and helium ash at the required rate while using every other pumping duct [12]. The remainder can be used for diagnostics.

Summary

The divertor of a high power density compact burning plasma device like FIRE presents many challenges. The high triangularity required for access to advanced tokamak modes of operation force the inner divertor leg to be very short and the outer divertor depth is limited by toroidal field coil height limitations. The use of a double null divertor configuration and high-density operation reduces the power load on the inner divertor to the point where the plasma there detaches from the plate with hydrogen radiation alone. Since the outer divertor plate must be actively cooled, we will use the power in the cooling water to balance the power split between the two divertors. The outer divertor plasma can be detached from the plate with the addition of Ne in the divertor. We can control the amount of Ne to limit the heat flux on the outer divertor.

The use of a tungsten rod surface on the divertor will allow operation with intrinsic impurities and a high recycling plasma. Reduction of the heat load by added radiation will increase the engineering margin in the divertor. Details of the coolant connections to the baffle remain to be determined. The inner divertor and first wall are cooled between shots through mechanical contact with the cooled copper shell inside the vacuum vessel.

High-density plasma operation typical of a high field device like FIRE simplifies particle control in the divertor. Our calculated neutral pressures in the divertor (1.3-6.7 Pa) permit us to use viscous drag pumps to remove He and cryocondensation pumps for hydrogen isotopes. Only half of the available divertor ports are required for pumping.

Extensive calculations of eddy currents during vertical disruption events have shown that our concept of using pins on the vacuum vessel and sliding pins on the divertor plates is adequate for mounting of the divertor. Erosion of the divertor plate during disruptions is the life-limiting factor.

References

1 D.M. Meade, R.J. Thome, N.R. Sauthoff, et al., Physics Basis for the Fusion Ignition Research Experiment (FIRE), Presented at the EPS Meeting, Budapest, June 12-15, 2000.
2 D.M. Meade; "Fusion Ignition Research Experiment", *Comments on Plasma Phys. Controlled Fusion, Comments on Modern Physics,* **2**, pp. 81-97, 2000.
3 FIRE Design Team; "Fusion Ignition Research Experiment (FIRE) Engineering Status Report for Fiscal Year 2000"; Princeton Plasma Physics Laboratory Report No. 81-001030-IREFYRPT-FT.doc.
4 ITER Physics Basis, Nucl. Fusion **39** (1999) 2208
5 G.R. Smith, P.N. Brown, R.B. Campbell, D.A. Knoll, P.R. McHugh, M.E. Rensink and T.D. Rognlien, Techniques and Results of Tokamak-Edge Simulation, J. Nucl. Mater. **220-222** (1995) pp. 1024-1027
6 M.A. Ulrickson, *et al*., "Physics Basis for the Fusion Ignition Research Experiment (FIRE) Plasma Facing Components", To be Published in the Proceedings of the 21st Symposium on Fusion Technology (SOFT) August, 2000.
7 S. Mirnov, J. Wesley, N. Fujisawa, et al., Chapter 3, MHD Stability, Operation Limits and Disruptions, Nucl. Fusion **39** (1999) pp.2251-2389
8 C.E. Kessel, S.C. Jardin, R.H. Bulmer, R.D. Pillsbury, P.W. Wang, G.H. Neilson, D.J. Strickler, "Poloidal field control for the Tokamak Physics Experiment", Fusion Technology **30**, p.184-200, Nov 1996
9 R.E. Nygren, et al., "Recent High Heat Flux Tests on W-rod-armored Mockups", To be published in the proceedings of the 14th Plasma Surface Interactions Conference, Rosenheim, Germany, May 2000.
10 R.Watson, et al., "Development of High-Z Plasma Facing Components For ITER", Fus. Tech. **34** 443 (1998).
11 D.E. Driemeyer, *et al*., "Development of Direct HIP-Bonding Processes for Tungsten-Brush Armor Joining," Proc. 18th Sym. on Fus. Eng., Oct. 1999.
12 M. Gouge, P. Fisher, *et al*., Presented at the FIRE Design Review, Princeton, NJ, June, 2001 (see fire.pppl.gov).