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Lessons Learned From The Design of ITER Internal Components

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***Abstract*— The traditional design sequence for a fusion device starts from desired plasma performance and a coil set that optimizes formation of the required plasma equilibrium. Designers then reserve space for internal components, vessel, and cryostat. The ITER process has taught us that this traditional sequence should be revised. The discovery of convective transport in the plasma scrape-off layer has greatly increased heat flux to the first wall (FW), implying that power is flowing along field lines. Shaping of the FW's plasma-facing-surface and divertor components is now a critical design consideration and constraint. Plasma duration has increased to the point that active cooling of internal components is required. Ever more complex plasma diagnostics require complex openings to view the plasma and complex routing of items like cables behind the blanket. As we move toward the next generation of fusion machines, there is a need for many engineering diagnostics to monitor the operating state of actively cooled components. Internal coils for ELM, resistive wall mode, and plasma rotation control further complicate the region between blanket modules and vessel. Adding new components between the vessel and blanket removes material that is either shielding external components or breeding tritium in a reactor. Traditionally, such additional internal components are added during later design phases when space has been fixed. Using design by analysis during the conceptual design phase allows the space required for internal components to be more accurately defined so balanced trade-offs among magnets, vessel, and internal components are made. The result is a concept that is easier to integrate and does not disproportionately constrain later design phases for any one system.**

Keywords—fusion reactor design, plasma-facing component design, plasma-materials interactions, fusion reactor blanket components

I. INTRODUCTION

Powerful desktop computers and computer aided engineering software make it possible to design fusion devices with detailed 3D representations of complex internal components. The traditional design sequence starts from desired plasma performance and a coil set that optimizes formation of the required plasma equilibrium. Designers then reserve space for internal components, vessel, and cryostat. The ITER process has taught us that this traditional sequence should be revised. The discovery of convective transport in the plasma scrape-off layer has greatly increased heat flux to the first wall (FW), implying that power is flowing along field lines. Shaping of the FW's plasma-facing-surface and divertor components is now a critical design consideration and constraint. Plasma duration has increased to the point that active cooling of internal components is required. Ever more complex plasma diagnostics require complex openings to view the plasma and complex routing of items like cables behind the blanket. As we move toward the next generation of fusion machines, there is a need for many engineering diagnostics to monitor the operating state of actively cooled components. Internal coils for ELM, resistive wall mode, and plasma rotation control further complicate the region between blanket modules and vessel. Adding new components between the vessel and blanket removes material that is either shielding external components or breeding tritium in a reactor. Traditionally, such additional internal components are added during later design phases when space has been fixed. Using design by analysis during the conceptual design phase allows the space required for internal components to be more accurately defined so balanced trade-offs among magnets, vessel, and internal components are made. The result is a concept that is easier to integrate and does not disproportionately constrain later design phases for any one system.

Another major lesson learned is that the effect of very long power scrape-off lengths in the far scrape-off region on the FW causes a fundamental change in the FW's requirements. We increased the heat flux to the FW from about 0.5 MW/m^2 to about 5 MW/m^2. The technology required to remove 5 MW/m^2 is less forgiving of off-normal heat flux and more susceptible to fatigue and creep damage. Careful shaping of the FW surface is required to mitigate the effect of misalignment or different plasma edge profiles. Since the plasma touches the FW, events such as disruptions cause halo current to flow to the surface, constraining the design of FW components because of the need to minimize electromagnetic forces. Increased understanding of far scrape-off layer transport and discovery of techniques to reduce power flow in the far scrape-off layer would simplify the design for future devices and increase breeding ratio and expected lifetime of internal components.

II. PLASMA SCRAPE-OFF LAYER PROFILE

Prior to about 2004 (throughout the ITER Conceptual and Engineering Design Activities), it was believed that the conducted power in the scrape-off layer (SOL) had a short e-folding length. This meant the divertor received the majority of

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the plasma thermal power. Hence, the divertor was design for 10-12 MW/m^2 peak power density. First Wall heat flux was believed to be only plasma radiation and some charge-exchange particles (neutrals). Only 0.35 MW/m^2 heat flux was expected on the first wall (including a safety factor). Because there was no direct contact of plasma with the first wall, the first wall could be simple flat facets and alignment with the magnetic field was not necessary [1]. Each shield module was covered with four first wall panels. The first wall panels were attached to the shield by an oval stalk that helped resist disruption induced radial torque. Disruption induced eddy current forces did require the first wall to be divided into fingers to reduce loads. The fingers were oriented in the poloidal direction. A 1 m wide by 0.5 m thick shield module is illustrated in Figure 1. The coolant manifold was a multiple chamber box like structure with no internal voids. Some void space was required around the manifold and branch pipes, which fed coolant to each shield module, but the total loss of shield volume was less than 5%.

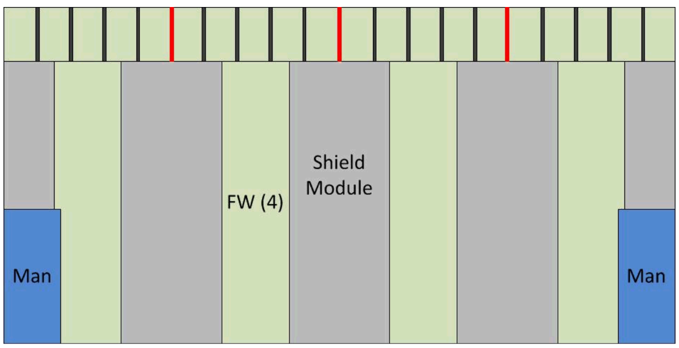

Figure 1. A schematic of the First wall arrangement during the ITER EDA showing the FW fingers and attachment stalks.

About 2004, data from several diverted machines running with “H” mode confinement revealed a very long tail on the far scrape-off layer power flux (see Figure 2) [2, 3, 4]. The energy in the tail was attributed to Edge Localized Mode (ELM) transport that is frequently seen in high confinement plasmas. This discovery caused fundamental changes in the ITER Blanket/Shield design.

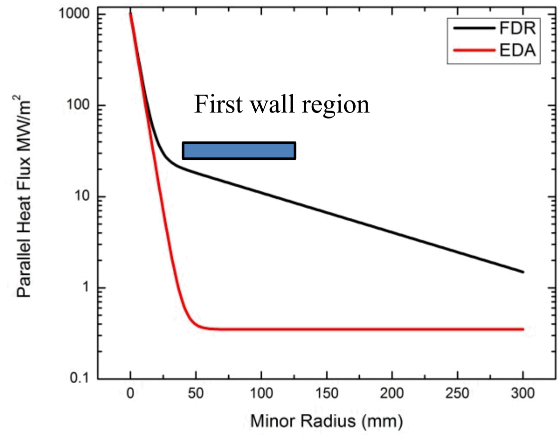

Figure 2. An illustration of the dramatic change in the plasma edge power profile that occurred about 2004.

Because plasma flowing along field lines was interacting with the first wall, it was necessary to carefully shape the plasma facing surface of the FW to avoid edges having very high heat flux (for a discussion of the need to shape see [5, 6, 7]). Each blanket module must be carefully aligned to flux surfaces [8, 9] to assure the shaping was would prevent plasma impinging on surfaces that were nearly normal to the flux lines. During disruptions, plasma current could flow from the plasma to the first wall through the component to the vessel and back through another component to the plasma (a phenomenon known as halo currents [10]). If this current flowed through poloidally oriented fingers, the forces would cause permanent deformation of the fingers (bent toward the plasma). These issues were resolved by changing to toroidally oriented fingers. The need for shaping meant there was room for only one FW panel per blanket module (for more details see [11]. At about the same time, internal ELM control coils were introduced behind the BM [12]. The ELM coils were introduced to cause frequent ELMs and avoid infrequent very large ELMS [13]. This pushed the coolant manifolds closer to the plasma facing surface because the manifold must be maintainable while the coils are not. It was also decided that the box like manifold did not allow for leak checking individual BM and it would be more difficult to maintain that individual manifold pipes for each BM. The manifold changes and ELM coils increased the size of the cutout in the BM and increased the void volume surrounding the manifold and coils. Figure 3 shows a schematic of the revised blanket module and the increased void fraction. Purple areas are voids and the void region at the bottom is an average over several small features such as mounts, branch pipes, and electrical straps.

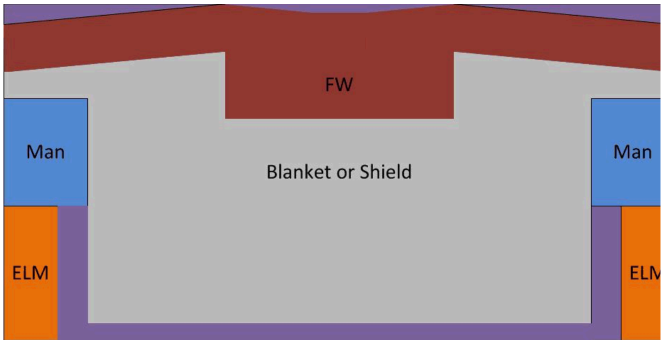

Figure 3. A schematic section of the final design of the ITER first wall showing the shaping needed because of plasma impingement on the FW.

Even with the strong shaping and careful alignment with flux surfaces, the peak heat load on up to 50% of the first wall is 5 MW/m^2. The peak heat load on the other half of the first wall is 1-2 MW/m^2 (including 0.35 MW/m^2 due to radiation and charge exchange). All of these values are much higher than the original assumption. Because of extensive slitting of the FW and EM control slits in the BM, the EM loads on the mounts are still within the allowable loads for slightly improved mounts. Further discussion of the major impact of these heat loads on more fusion reactor like future devices is below.

III. Blanket Space Reservation

Increased void space and changed average composition (more water and less steel) have combined to reduce the effective thickness of the shield. Even though the inner wall shield thickness was increased by 8%, the nuclear heating of the toroidal field coils does not meet the ITER requirements [15]. There are still regions on the outer wall where the specified maximum heating of the vessel shell is exceeded. We conclude that the estimates used to set the space reservation for shielding on ITER need to be increased by 10-20% to assure adequate shielding for future machines. This will increase the volume at high magnetic field and increase estimated machine cost, but failure to include adequate space for shielding will impact repetition rate, lifetime of components, and delay access for maintenance. A schematic of the revised ITER blanket with 10 and 20% more thickness is shown in Figure 5.

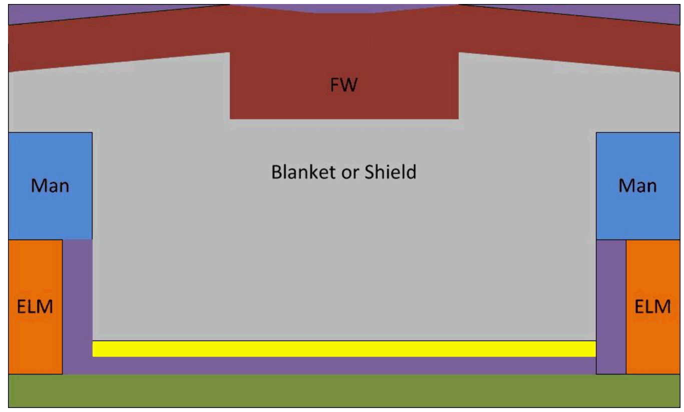

Figure 4. An illustration of the changes in the first wall section if 10% (yellow) is added to the thickness and 20% (green) is added to the thickness.

IV. Implications of First Wall Heat Loads

A peak heat load of 1-5 MW/m^2 on the first wall requires the use of a good thermal conductor on the plasma facing surface, good heat transfer coefficients to the coolant, relatively thin plasma facing surfaces, and quite possibly low Z tiles facing the plasma (due to erosion). It is instructive to examine the effect of the thermal conductivity of the heat removal layer under an assumed plasma facing material (Beryllium for this study). We assumed excellent heat transfer to the coolant (4.4 W/cm^2 K) and a limit of ½ the melting point of Be for the surface temperature limit. Simple 1D steady-state thermal solutions were used to simplify the interpretation of the results. Figure 5 shows the allowed thickness of the heat transfer layer and the Be tile as a function of the thermal conductivity of the heat transfer layer.

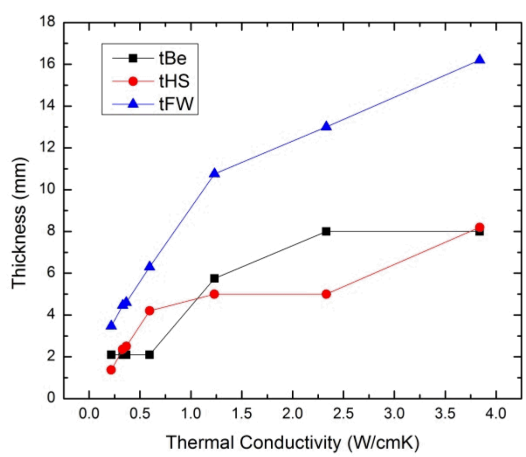

Figure 5. The maximum allowed thickness for the first wall when the heat flux is 5 MW/m^2 versus thermal conductivity of the heat removal layer.

Obviously, other thickness combinations between those shown are possible, but the total first wall thickness would remain roughly constant. In all cases, real materials were used irrespective of whether fabricability has been demonstrated. It is evident that a minimum conductivity of 0.8 to 1 W/cm K is required for 5 MW/m^2. If the thicknesses of the FW layers are kept constant, the surface and interface temperatures increase rapidly with decreasing thermal conductivity of the heat removal layer (See Figure 6). It is clear that the temperatures for the low conductivity cases are unacceptably high.

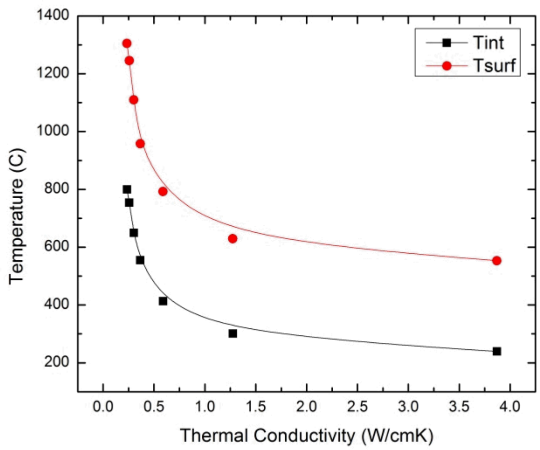

Figure 6. The change in first wall operating temperature as the thermal conductivity of the heat removal layer is changed under 5 MW/m^2 heat flux.

For lower peak heat flux, the minimum thermal conductivity is reduced and the total thickness of the FW can be increased which will increase lifetime under erosion. Figure 7 shows the allowed thickness for 2.0 MW/m^2 peak heat flux. The allowed thickness for the lowest thermal conductivity material is nearly 3 times greater than for 5 MW/m^2. The thickness need not be a thick as shown for the higher conductivity material because the operating temperature can be lower, which will reduce the thermal stress and increase fatigue lifetime. A minimum thermal conductivity of about 0.5 W/cm K is required for 1-2 MW/m^2 heat flux. It is well established that a thermal conductivity of about 0.4 W/cm K is sufficient for 0.35 MW/m^2 [16].

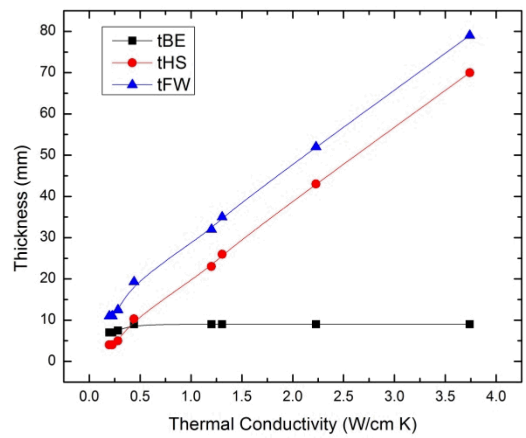

Figure 7. The maximum first wall thickness with 2 MW/m^2 heat flux versus the thermal conductivity of the heat removal layer.

While the very thin first wall plasma facing surface required for high heat flux with low conductivity materials is a problem from an erosion lifetime perspective, stress in such thin layers is also a problem for a nuclear device because of limits on the membrane stress in the first wall. Figure 8 shows the relative membrane stress as a function of the thermal conductivity of the heat transfer layer with the thickness limitation discussed above. While the lower conductivity materials may have higher yield strength due to alloying, they are not 10-20 times stronger than the best high conductivity materials. It must also be noted that Helium gas is often proposed for cooling reactor blankets because it is compatible with high temperature (above 500 C) and high efficiency power conversion. However, He cooling operates at higher pressure than water used on ITER and requires complicated heat removal layers to achieve high heat transfer coefficients. Even thicker first wall panels are likely for He cooling.

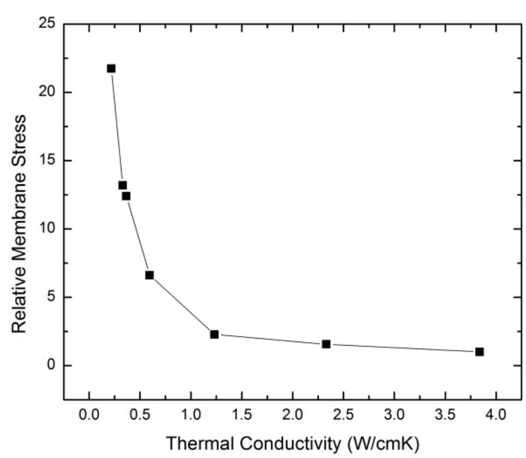

Figure 8. The relative increase in membrane stress in the first wall as the thickness is reduced due to lower thermal conductivity as described in the text.

Thermal conductivity and electrical conductivity are correlated for metallic materials. This means that very large EM loads will be imposed on the first wall during disruptions. The heat removal layer of the first wall must be supported and the overall first wall supported by the blanket or the disruption induced forces could damage the first wall and/or blanket. The support needed for the fingers on the first wall on ITER is about twice the thickness of the heat removal structure and an additional beam in a channel in the shield block is needed for ITER (see Figure 3). Since the first wall materials on ITER (copper alloy, stainless steel and water) are all good shielding materials, there is no penalty due to a separate removable first wall. However, in a fusion reactor a separate first wall that can handle 5 MW/m^2 will contain no breeding material and the breeding volume will be significantly reduced by the first wall. An actively cooled first wall will both reduce the peak neutron energy and reduce the neutron flux reaching the breeder region. It is evident from Figure 3 that the separable first wall removes another 20% of the blanket volume (in addition to that removed by the coolant manifold and internal coils). While some breeding might be recovered by increasing the blanket thickness, high heat flux to the first wall of a reactor may reduce the tritium breeding ratio to less than unity. Even if the first wall heat flux is reduced to 2 MW/m^2, the first wall thickness cannot be reduced enough to eliminate this concern because of disruption eddy current forces and halo currents will still flow to the surface and the first wall must still be shaped. Further discussion of this topic is in the conclusion section.

If the maximum heat load on the first wall were 5 MW/m^2, robust first wall designs are easily found and the predicted lifetime due to erosion is sufficient for long duration operation. But disruptions and large uncontrolled ELMs cause heat loads that greatly exceed the steady-state value. Figure 9 shows the steady-state and transient heat loads in context. The thin heat removal structures needed for removal of high heat flux are incompatible with the very high transient loads. The risk is both melting of the heat removal layer and thermal fatigue

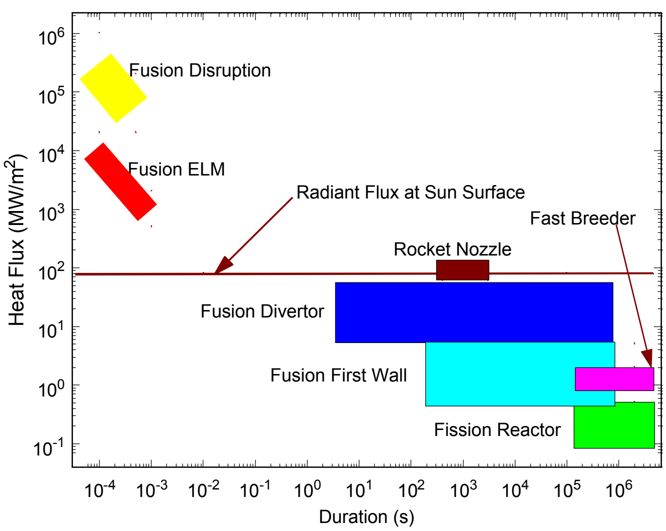

Figure 9. A comparison of typical fusion heat fluxes in the context of other systems. The transient loads on the left are the challenge for fusion first walls.

cracking of the layer. An additional risk to a portion of the first wall is run-away electron generation during a disruption. Unmitigated run-away electron beams can easily melt the entire plasma facing portion of a first wall panel and release coolant into the vessel. Events like these are the primary reason the first wall on ITER is fully remotely replaceable.

V. Design By Analysis

There are many powerful tools that run on workstations, which can be used to evaluate trade-offs among the conflicting requirements described above. The tools include computational fluid dynamics, neutron transport heating and activation, electromagnetic force analysis, thermal and structural analysis using direct input from the other tools in the list. Increasingly, the primary capabilities of the individual codes are being combined into multi-physics models where several phenomena are analyzed in the same model [17]. It is no longer necessary to use rules-of-thumb or correlations to assess the space requirements or estimate the performance of systems like blankets. Each of these tools requires clean geometry to facilitate the mesh generation process. Clean geometry means the model is free of: 1) tangential intersections (e.g., equal diameter cylinders crossing); 2) fillets and chamfers (these are added to high stress areas when detailed analysis shows the need during final design); 3) under-cuts and weld relief grooves (added late in the design); 4) small surfaces (any surface much smaller than the smallest mesh size needed to resolve fields like temperature gradients or force gradients); 5) bolt holes or inserts; and 6) details which can only be added once the overall temperature, stress, load, or force distribution is determined from preliminary analysis. The majority of the tools can work directly with component geometry generated using computer aided design (CAD) tools (a few still require translation using commercial translation software). As the design matures, it becomes increasingly necessary to maintain an analysis version of the geometry and a more detailed design that is evolving toward the final manufacturing model. It is essential that these two models be self-consistent, which means that configuration control must be implemented early in the design process. Options must be clearly tracked and abandoned options isolated from the active design path. The analysts must work closely with the CAD operator to assure the analysis models are clean enough but also contain essential features for each analysis type.

VI. Internal Coils

In a reactor like device, any internal coils must be placed behind the shield or blanket to protect the coil insulators from radiation damage. One consequence of this placement of the coils is the AC fields generated by the coils will be shielded by the conducting structure between the coil and the plasma. ELM pacing only needs resonant magnetic fields that are DC or very slowly varying. The presence of the blanket in front of the coils will not interfere with ELM pacing as long a sufficient ampere turns are provided. Another possible use of internal coils is for resistive wall mode (RWM) control [18]. RWM control requires detection of plasma magnetic structures and active feedback of the coil current based on the detected signal. Both the detected signal and the AC field generated by the coils will be altered by the blanket structure.

To determine the impact of a thick conducting blanket on AC fields generated by internal coils, we constructed a simplified single turn equatorial ELM coil similar to the ITER coil and placed it in the context of the ITER equatorial blanket modules and equatorial port structures. Figure 10 and Figure 11 show the coil and surrounding blanket modules from both the plasma side and the vessel side.

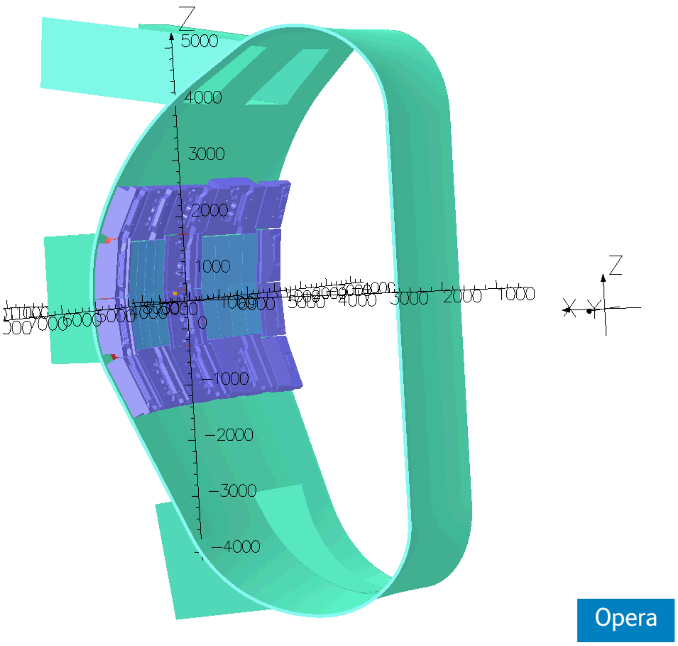

Figure 10. The geometry used to assess the shielding of internal coil fields behind a conducting blanket structure including the vacuum vessel.

We used the OPERA code [19] steady-state AC analysis module to calculate the AC transfer function for the ELM coil. The field amplitude at the edge of the plasma is plotted versus excitation frequency in Figure 12. The field amplitude falls to 1/e the DC value at 13 Hz. The phase of the field at the plasma relative to the phase at the coil is shown versus frequency in Figure 13. From these results we conclude the blanket modules act like a low pass filter for fields generated by the coils. The maximum useable frequency is about ZZ Hz. Above this frequency, the attenuation of the field and frequency dependence of the phase render the coils useless for feedback.

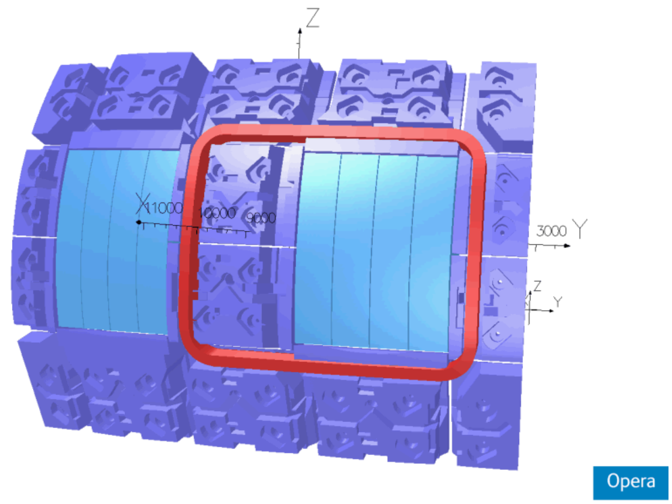

Figure 11. A view from behind the blanket modules showing the ELM coil and diagnostic first wall modules in the equatorial port.

In addition, the magnetic field from the plasma will be similarly attenuated and phase shifted, so the sensors that detect plasma field will be inaccurate. Because the conductive blanket modules provide some stabilization of RWM, the shielded coils may still be useful, but simulations of the feedback must include the low pass filter aspects imposed by the conducting blanket and the attenuation and phase shifting of the plasma magnetic signal.

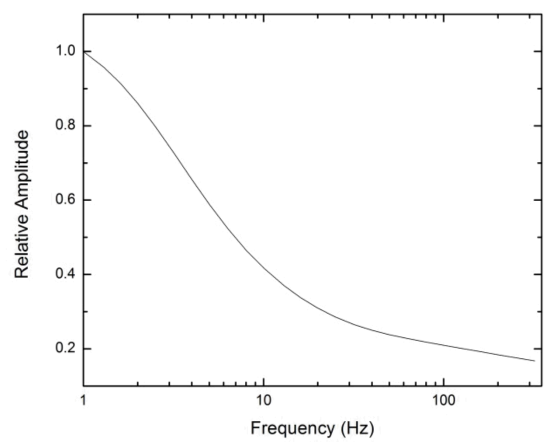

Figure 12. The relative amplitude of the B field at the plasma edge from the ELM coil versus frequency of the excitation.

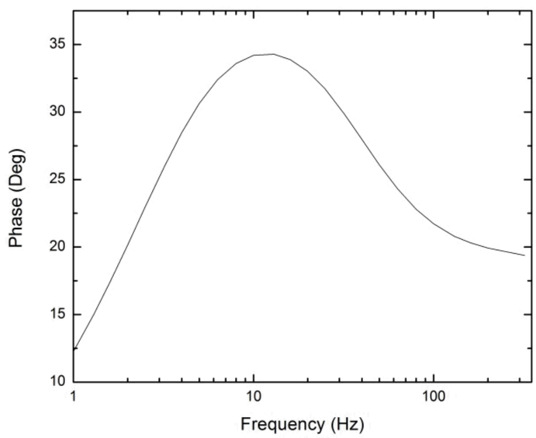

Figure 13. The phase of the B field at the plasma edge from the ELM coil relative to the phase of the excitation of the coil.

VII. CONCLUSIONS

If the Scrape-off Layer power profile shown in Figure AA is seen on a reactor, the first wall of the breeding blanket will be a thick heat removal structure with no breeding capability. Tritium breeding will be severely reduced (likely to less than unity). The only solutions of this issue are to: 1) reduce the scrape-off length for power in the far SOL; 2) operate in a plasma regime that is free of ELMs; or 3) find a much more efficient tritium breeding material that can tolerate the loss of neutron flux. The first option would require better understanding of the transport mechanisms responsible for the long tail followed by invention of some way to enhance the parallel transport to reduce the scrape-off length for power. One implication of any such technique would be to increase the power flow to the divertor baffle, which might just shift the problem to another region of the machine. The second option typically means operating in a lower confinement regime like “L” mode. This would have a big impact on the conclusions about how big a reactor needs to be and the cost of fusion power. Resolution of this issue through a coordinated measurement and modeling effort must be a high priority activity during the conceptual design phase of any next step beyond ITER. Failure to find relief from the long tail on the SOL power profile will likely make a magnetic fusion reactor impractical.

Electromagnetic forces and the need to be able to perform remote maintenance to repair damage due to steady-state erosion or disruption damage, all push blanket modules to be smaller rather than large structures. This means that there will be large manifolds feeding coolant to the blanket modules. A continuing need for internal coils behind the blanket increases the volume removed from the blanket. The space reserved for a breeding blanket must be approximately 30% greater in radial extent than is typically assumed in reactor design studies. The added thickness is also needed to assure the vessel and external components have sufficiently low neutron flux and fluence to meet regulatory requirements. Extra radial build will increase machine size and cost, which will be painful. Failure to provide adequate radial space for the blanket will cause greater pain in loss of run time, delayed maintenance activities due to excessive activation, inability to do maintenance due to Helium production, or failure to meet machine fluence goals.

Continued use of rules of thumb or correlations to perform design studies for fusion reactors will cause further erosion of the confidence in the results of such studies. It is imperative that the best analysis codes for neutron transport, thermal-fluid analysis, and electromagnetic force simulation be used for the next generation design studies. Any design point will then be supported by a strong fundamental science basis and cost estimates will be more realistic. Use of the available tools will require discipline in creation of 3D geometry models of components to make those models suitable for meshing and analysis. Any extra effort required to create such models will be repaid many times over when manufacturing details are added because the clean models can be tightly configuration controlled and the details are added to a strong self-consistent framework.

If methods can be found to mitigate or even eliminate both disruptions and ELMs, first wall and blanket designs can both be made more reliable and have greater fatigue life. Optimization of blanket design in the absence of transient heat loads will likely lead to improved tritium breeding and improved thermal efficiency. It is hoped that ITER will provide a great deal of information on how to control transients.

Finally, if internal coils are to be used for plasma control on a next step device, the electromagnetic shielding effects of the blanket on both the signals from the plasma and the field generated by the coils must be analyzed with realistic blanket modules. The thermal conductivity required for adequate heat transfer implies high electrical conductivity and good shielding of transient magnetic fields by the blanket. The use of metallic liquids for either breeding or cooling in the blanket will enhance the shielding.

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