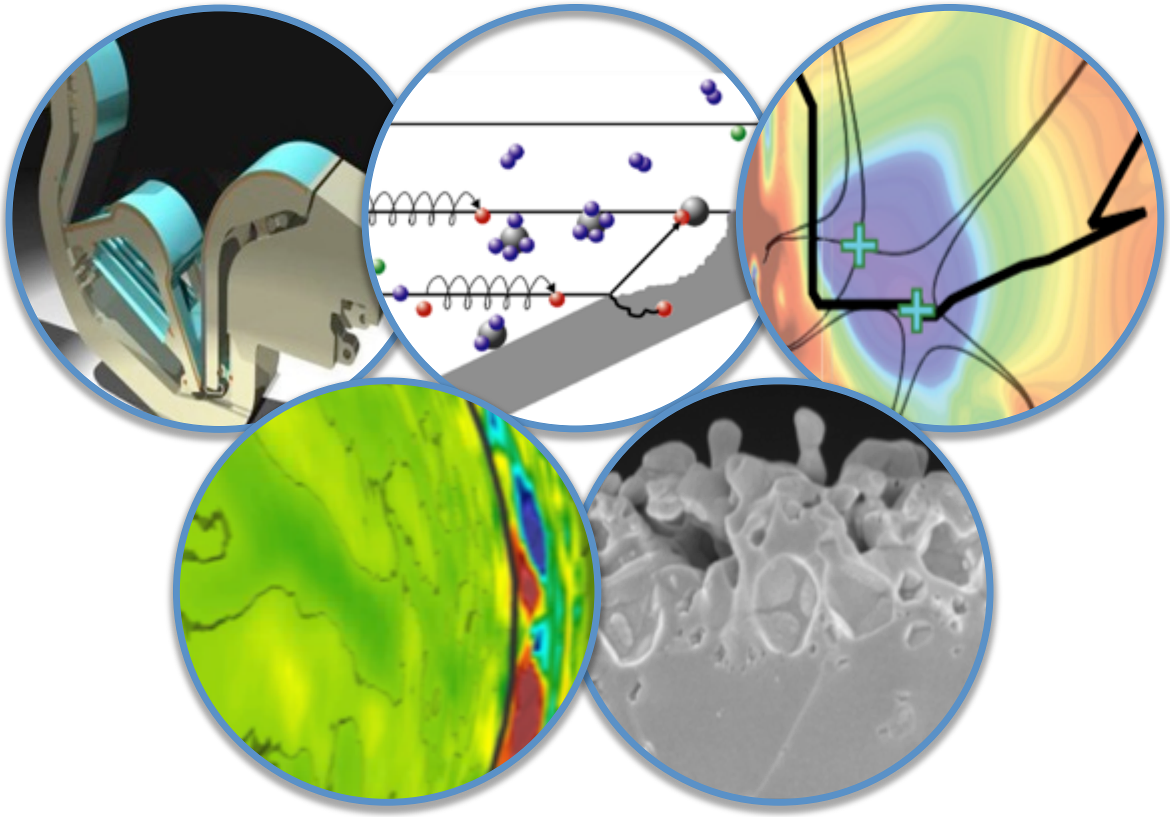
FUSION ENERGY SCIENCES WORKSHOP



ON PLASMA MATERIALS INTERACTIONS

Report on Science Challenges and Research Opportunities in Plasma Materials Interactions

MAY 4-7, 2015

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Top images in circles:

Left: **Schematic of the ITER divertor and components,** https://www.iter.org/mach/divertor

Center: **Plasma-surface interaction phenomena on an inclined target,** https://www.differ.nl/research/plasma_surface_interactions

Right: **Poloidal field contours in an innovative "snowflake" divertor,** [Soukhanovskii, V.A. et al., Phys. Plasmas **19**, 082504 (2012)]

Bottom images in circles:

Left: **Peta-scale calculation of the edge turbulence in a tokamak with the XGC1 code by S. Ku and C.S. Chang, and visualization by D. Pugmire,** https://www.olcf.ornl.gov/2014/02/14/the-bleeding-edge-of-fusion-research/

Right: **Tungsten micro-structural changes under helium bombardment,** https://www.differ.nl/research/plasma_surface_interactions/plasma_surface_interactions_engineering

FUSION ENERGY SCIENCES WORKSHOP ON PLASMA MATERIALS INTERACTIONS

Report on Science Challenges and Research Opportunities *for* Plasma Materials Interactions

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Co-Chair: Steven J. Zinkle, University of Tennessee – Knoxville
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Plasma-Surface Interactions
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Dennis L. Youchison, Sandia National Laboratory

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Fusion Energy Sciences
Office of Science
U.S Department of Energy
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Executive Summary

I. Executive Summary

The realization of controlled thermonuclear fusion as an energy source would transform society, providing a nearly limitless energy source with renewable fuel. Under the auspices of the U.S. Department of Energy, the Fusion Energy Sciences (FES) program management recently launched a series of technical workshops to "seek community engagement and input for future program planning activities" in the targeted areas of (1) Integrated Simulation for Magnetic Fusion Energy Sciences, (2) Control of Transients, (3) Plasma Science Frontiers, and (4) Plasma-Materials Interactions aka Plasma-Materials Interface (PMI).

Over the past decade, a number of strategic planning activities[1-6] have highlighted PMI and plasma facing components as a major knowledge gap, which should be a priority for fusion research towards ITER and future demonstration fusion energy systems. There is a strong international consensus that new PMI solutions are required in order for fusion to advance beyond ITER. The goal of the 2015 PMI community workshop was to review recent innovations and improvements in understanding the challenging PMI issues, identify high-priority scientific challenges in PMI, and to discuss potential options to address those challenges. The community response to the PMI research assessment was enthusiastic, with over 80 participants involved in the open workshop held at Princeton Plasma Physics Laboratory on May 4-7, 2015. The workshop provided a useful forum for the scientific community to review progress in scientific understanding achieved during the past decade, and to openly discuss high-priority unresolved research questions. One of the key outcomes of the workshop was a focused set of community-initiated Priority Research Directions (PRDs) for PMI.

Five PRDs were identified, labeled A-E, which represent community consensus on the most urgent near-term PMI scientific issues. For each PRD, an assessment was made of the scientific challenges, as well as a set of actions to address those challenges. No prioritization was attempted amongst these five PRDs. We note that ITER, an international collaborative project to substantially extend fusion science and technology, is implicitly a driver and beneficiary of the research described in these PRDs; specific ITER issues are discussed in the background and PRD chapters. For succinctness, we describe these PRDs directly below; a brief introduction to magnetic fusion and the workshop process/timeline is given in Chapter I, and panelists are listed in the Appendix.

PRD-A: Identify the present limits on power and particle handling, as well as tritium control and inventory, for solid and liquid plasma facing components, and extend performance to reactor relevant conditions with new transformative solutions

Due to the challenging power exhaust and tritium control environment in fusion systems, both solid and liquid materials should be considered as potential plasma facing components, with targeted lifetimes of several years. Breakthroughs in solid materials development, critical evaluation of liquid metals, and seminal advances in engineered materials and advanced manufacturing techniques, coupled with multi-scale theoretical computations, will be used to develop integrated plasma facing components that can function reliably in the severe fusion plasma environment.

<u>PRD-B</u>: Understand, develop and demonstrate innovative dissipative/detached divertor solutions for power exhaust and particle control, sufficient for extrapolation to steady-state reactor conditions

The magnetic divertor is the leading concept to separate plasma-materials interactions from the core plasma via a specially designed target region. Viable divertor solutions that can manipulate and stably control divertor plasma conditions will be investigated. These are needed so that the vast majority of the plasma power, that would otherwise concentrate and damage the target surfaces, is instead dissipated through the release of benign radiant heat, and/or via detachment of the plasma from the material boundaries. Such properties are necessary to minimize damage, e.g. melting and erosion, to the plasma facing components. Promising innovative concepts involving manipulation of both the magnetic fields and the containing surfaces in the divertor, as well as the materials used for the target surfaces – solid, liquid and vapor, will also be investigated.

<u>PRD-C</u>: Understand, develop and demonstrate innovative boundary plasma solutions for main chamber wall components, including tools for controllable sustained operation, sufficient for extrapolation to steady-state reactor application

A relatively cool boundary plasma region surrounds the hot fusion core plasma in a tokamak and makes contact with the main chamber walls, where hardware components such as sophisticated radio frequency (rf) wave actuators for plasma heating and current drive, are located. Recommended actions include 1) understanding both bulk plasma and impurity transport in the presence of intermittent turbulence to assess the impact of plasma fluxes on the vessel walls and the fate of eroded materials, and 2) investigating rf-specific effects in the boundary region to mitigate deleterious interactions. Discovering how these processes couple and influence the core plasma, and learning to control them in a reactor environment with new innovations, presents a new frontier in fusion physics.

<u>PRD-D</u>: Understand the science of evolving materials at reactor-relevant plasma conditions and how novel materials and manufacturing methods enable improved plasma performance

Plasma facing surfaces experience an evolving layer of material that is continuously re-constituted via erosion and re-deposition, leading to dynamic surface properties and plasma-surface interactions in fusion devices. The entailed actions are: 1) to understand the science of PMI on these dynamic surfaces at reactor-relevant conditions, and decipher the practical implications on heat and particle limits, and 2) to develop radiation tolerant materials that maintain material performance despite plasma (neutrons, T, He) induced material evolution, using both advanced manufacturing and modeling for tailoring of solid surfaces and evaluation of self-healing (liquid) structures.

<u>PRD-E</u>: Understand the mechanisms by which boundary solutions and plasma facing materials influence pedestal and core performance, and explore routes to maximize fusion performance

Conditions at the plasma boundary, both the divertor dedicated to handling the heat flux and the main chamber wall that comprises most of the surfaces, are known to affect the performance of the hot core plasma where fusion takes place. Optimization of the *properties of the outer 10 percent of the plasma, where self-organized transport barriers occur, also improves performance of the fusing core plasma. Physics in this region is complex and multi-scale, and our understanding is incomplete. This critical gap will be addressed through a coordinated program of experiments and modeling, centered on existing U.S. experiments and supplemented by collaborations on international devices, and extended via a new facility to develop optimized core-boundary solutions for future fusion devices.*

In addition to these five PRDs, several crosscutting high-impact research activities were identified, as evidenced by their prominence in multiple PRDs.

CC-1: Enhanced exploitation of existing machines for plasma-materials interactions studies

This opportunity would leverage our existing major tokamak investments with new plasma-materials interactions diagnostics, targeted upgrades, and additional dedicated run time and people. Fusion Energy Sciences has made substantial investments in its facilities, and enhanced resources for plasma-materials interactions would increase their capability for world-leading discovery science. Additional emphasis is placed on multi-disciplinary R&D that simultaneously characterizes both plasma and materials responses to plasma-materials interactions, in lieu of the standalone studies presently employed. Enhanced modeling and theory is coupled to the experimental efforts to ensure effective model validation and development, toward building a predictive capability.

CC-2: Examine long-pulse plasma-materials interactions science issues under reactor-relevant conditions of high accumulated plasma and neutron fluxes

The development of steady state operation will require mastering plasma-materials interactions science to develop plasma facing components with strong erosion resistance and/or self-healing capability during prolonged exposure (>10^6 sec) to high particle/heat fluxes and intense fusion neutrons. An improved understanding of the fundamental degradation mechanisms associated with plasma-materials interactions is needed to identify potential plasma-facing component materials and operational regimes. Complementary R&D can be done via collaboration on international, long-pulse toroidal devices, e.g. EAST, KSTAR, W7-X, and WEST, and long-pulse linear devices with controlled exposures and comprehensive diagnostic capability. Development of world-leading capability requires a new high fluence, linear divertor simulator with flexible target stations.

CC-3: Understand the science of liquid surfaces at reactor-relevant plasma conditions and examine the feasibility of liquid plasma facing component solutions

Peak heat and particle fluxes during both steady operation and transient events are projected to push solid materials up to or beyond their plasma exhaust capabilities. Thus a concerted evaluation of liquid plasma facing components is advocated. Control and stabilization of liquid flows, evaluation of materials options (most likely liquid metals), understanding of tritium retention and recovery, identification of power exhaust capability, and the prospect of new confinement regimes with low recycling liquid metals are the envisioned research lines. The integrated computational and experimental *activities require design via purpose-built test stands, basic evaluation in divertor simulators, and integrated evaluation in toroidal devices, with opportunities in both domestic and international facilities.*

CC-4: Develop integrated plasma-material solutions in a purpose-built divertor test tokamak

While existing worldwide facilities, both domestic and international, enable discovery science via complementary approaches, in-depth understanding of the science for projection to reactors needs a flexible facility that allows innovative divertor and plasma facing component options with rapid evaluation cycles. Such a facility should operate at high power density and high boundary plasma pressure, because the atomic physics governing divertor power dissipation adds a dimensional component to the dimensionless experiments normally used to explore core plasma physics. As envisioned, this facility would be unique in the world, and would contribute world-leading plasma-materials interactions science in targeted areas, as well as testing the compatibility of boundary solutions with attractive core plasmas.

Overall this workshop was an extremely valuable activity, bringing together community scientists for necessary discussions and technical assessments, enabling consensus-building on scientific issues and strategic directions. Several of the recommended actions require a modest enhancement or redirection of existing resources, while others require new resources. Nonetheless, the community is both enthusiastic and eager to embark along the R&D directions described in this report, which will produce world-leading discovery science while advancing into new frontiers of fusion energy development.

While this document is a stand-alone report, the concurrent FES workshops on ‘Control of Transient Events’ and ‘Integrated Simulations’ also discuss PMI issues and actions, and the reader is referred to those reports for additional perspective on PMI science challenges and opportunities.

Section References

1 “Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan for Magnetic Fusion Energy”, Fusion Energy Sciences Advisory Committee (FESAC) Greenwald Panel Report, October 2007, http://science.energy.gov/~/media/fes/fesac/pdf/2007/Fesac_planning_report.pdf (2007).

2 “Research Needs for Magnetic Fusion Energy Science”, Report of the Research Needs Workshop (ReNeW) report, June 9-13, 2009. (2009).

3 Fusion Nuclear Science Pathways Assessment, http://bp.pppl.gov/pub_report//2012/PPPL (2011).

4 Opportunities for Fusion Materials Science and Technology Research Now and During the ITER Era http://science.energy.gov/~/media/fes/pdf/workshop (2012).

5 http://science.energy.gov/~/media/fes/fesac/pdf/2013/Final (2013).

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Chapter I

Introduction

I. Introduction

I. 1. Basics of Magnetic Confinement Fusion and PMI issues

The realization of controlled thermonuclear fusion as an energy source would transform society, providing a nearly limitless energy source with renewable fuel. Achieving controlled thermonuclear fusion has been recognized by the National Academy of Engineering as one of the 14 Grand Challenges for the 21st Century, on a par with global access to clean water[7].

Stars are exquisite examples of fusion reactors that operate for billions of years. Fusion from our own sun provides the power source that largely dominates the Earth's energy economy. While stars use the gravity from large quantities of mass to produce fusion, on Earth we must use either intense magnetic fields, or the inertia from rapidly compressed fuel, to achieve fusion reactions.

In magnetic confinement fusion, the historical focus has been on understanding the physics of the core plasma where fusion reactions take place. As our theories and diagnostics have advanced, experiments have evolved from examining global energy confinement to local transport properties in the core. When the measured transport rates exceeded expected rates predicted by "neoclassical transport," the studies evolved again toward additional transport mechanisms, e.g. turbulence driven by plasma gradients. These studies have made impressive progress, enabling us to project with improving confidence to reactor-scale core physics, e.g. in the design of a multi-national fusion project named ITER.

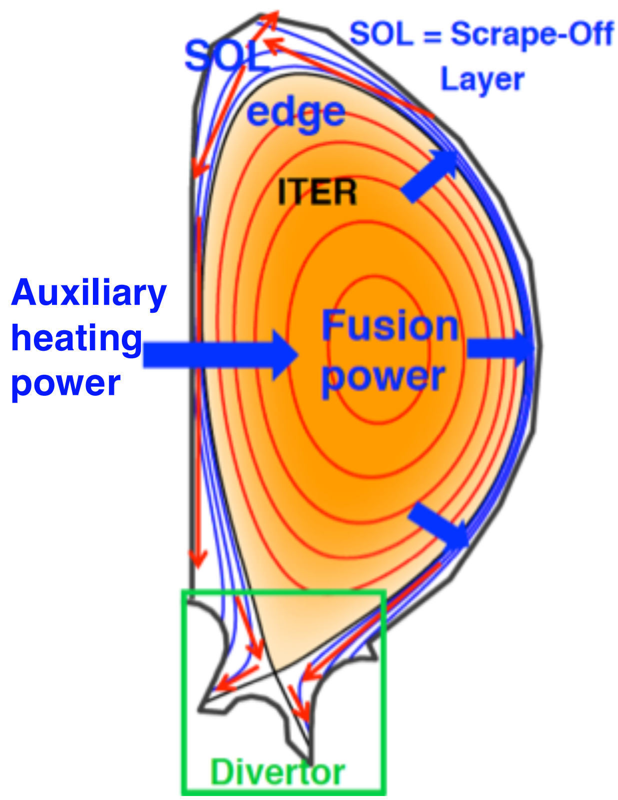

Fig. I-1: schematic of poloidal cross-section for ITER, indicating the regions of the boundary plasma: edge, scrape-off layer, and divertor. Red arrows indicate projection of dominant heat flows parallel to the magnetic field lines in the SOL and divertor.

Many years ago, designers implemented a magnetic topology separating the magnetic field lines that close upon themselves in the core from field lines that impinge on material surfaces ("open field lines"). This topology allowed us to "divert" the plasma away from the core to a specially designed target chamber to better manage the intense plasma-wall interactions (Figure 1-1). This spatial region just outside the core, including the magnetic divertor, is referred to as the boundary layer plasma, where the fourth state of matter meets the surrounding structures.

Over the past decade, a number of strategic planning activities have highlighted the gaps in PMI and plasma facing components as a major knowledge gap, which should be a priority for fusion research towards ITER and DEMO[1-6]. There is a strong international consensus that new PMI solutions are required in order for fusion to advance beyond ITER. The EU roadmap states, "A reliable solution to the problem of heat exhaust is probably the main challenge towards the realisation of magnetic confinement fusion."[8] In the United States, the priority of PMI research has been emphasized in the 2007 Greenwald report[2] and every FESAC or community study since then. Fusion reactor system studies find that power handling considerations limit the operating space and drive the overall size of net-energy producing fusion systems. PMI research offers opportunities to advance the science of both plasma and materials, and the potential for new discoveries arising from their interactions.

While refining our understanding for improved projections of ITER over the past decade, two sets of studies[9, 10] have reaffirmed the importance of the boundary layer plasma. Recently, it was found that the spatial region over which the energy of the plasma flows just outside the core to the magnetic divertor is rather narrow, more so than previously believed[9]. This heat flow channel width is insensitive to the size of the fusion device and the power flowing out of the core. Exhausting the fusion alpha and auxiliary input power in reactors will thus be very challenging, requiring substantial dissipation in the core plasma[11]. In addition, the world's largest fusion device, JET in the United Kingdom, implemented a wall with materials identical to those proposed for ITER: tungsten in the divertor and beryllium on the wall. It was found that operation with this "ITER-like" wall resulted in a 20 to 30 percent reduction in the energy confinement in early experiments[12]. While this reduction can be partially mitigated with nitrogen injection[10], the nitrogen seeding also produces ammonia that is incompatible with the specific technical design of the tritium plant commissioned for a deuterium-tritium campaign being planned for JET. Thus attention has turned toward optimization of the boundary plasma in terms of its compatibility with an attractive, fusing core plasma and its effect on wall material choices.

Several additional plasma facing materials issues have recently emerged that may impact the feasibility of fusion energy. Nano-scale fibrous structures have been observed to form on the surfaces of plasma facing materials such as tungsten during elevated temperature exposure to plasmas[13]. This could potentially lead to safety concerns due to enhanced dust formation, or conversely lead to enhanced tolerance for plasma-induced exfoliation from blistering. Consideration of neutron-induced cavity formation that will occur in the plasma facing materials in future fusion devices such as ITER also leads to the possibility of increased tritium sequestration that could adversely affect tritium site limits[14]. Thus, improved understanding of tritium transport and sequestration mechanisms in solid and liquids is needed, including the impact of neutron irradiation effects.

Excellent reference material on fusion physics, accessible to a general audience, is found in the ReNeW report (section: fusion primer)[2], and the National Academy of Sciences report: "Burning Plasma: Bringing a Star to Earth"[15].

I. 2. Workshop Process and Time Table of Activities

Over the past decade, a number of strategic planning activities[1-6] have highlighted PMI and plasmas facing components as a major knowledge gap, which should be a priority for fusion research towards ITER and future demonstration fusion energy systems. These activities included the 2009 Research Needs for Magnetic Fusion Energy Sciences Workshop (ReNeW) report and community white papers submitted for the FESAC 2014 Strategic Priorities panel assessment. Particularly with respect to the ReNeW workshop, it is timely to summarize updated community input in order to identify potential innovations or understanding that have emerged over the past six years relevant to the extremely challenging issue of PMI control.

In recognition of the growing importance of control of the plasma-material interface, FES initiated a community-led technical workshop on the science of PMI with two main elements:

A) Assess the leading scientific challenges in PMI
B) Assess technical options to address those challenges

The goal of the 2015 PMI community workshop was to review recent innovations and improvements in the understanding of PMI issues, identify high-priority scientific challenges in PMI, and discuss potential options to address those challenges. The community reaction to the PMI research assessment was enthusiastic, with over 80 participants involved in the community workshop held at Princeton Plasma Physics Laboratory on May 4-7, 2015. The workshop provided a useful forum for the scientific community to review progress in the scientific understanding achieved during the past decade and openly discuss high-priority unresolved research questions.

The process followed the community-led "Research Needs Workshops" model from Basic Energy Sciences, and the time frame for the activities is the decade 2015-2025. A common theme in the BES workshop reports is the identification of Priority (or alternately Principal) Research Directions. In the BES model, these PRDs are largely independent programmatic elements, and serve as a menu of activities for consideration by BES and community leaders. In this report, we have attempted to maintain separation between identified PRDs although there are obviously still cross-links, as fusion development is a more integrated problem. Common to the BES reports and this report is the identification of Crosscutting Research Opportunities, i.e. activities that would contribute to multiple PRDs and thus merit careful consideration.

The starting point for our community-led PMI workshop is the 2009 MFES Research Needs Workshop (ReNeW) strategic planning activity[2]. Specifically we were charged with re-evaluating the elements described in Thrusts 9-12 of the ReNeW report:

In addition, certain elements of Thrust 14, neutron effects on PMI, were also considered by the Thrust 10 subpanel. A multi-institutional panel of prominent scientists with a leader and deputy for each thrust, along with six-to-10 subpanel members in each area drawn from industry, national labs, and universities were assembled (see Appendix for complete list). Our activities included subpanel teleconferences, a face-to-face open three-day community workshop, writing assignments, and presentation of PRD content for community feedback via a Webinar. There were 80 white papers[16] submitted for this activity, along with 55 talks presented at the face-to-face meeting. Also, a group of senior fusion scientists served as advisors for identification of Crosscutting Research Opportunities, to add perspective to the challenges and actions presented in the PRDs.

Thrust #	PMI topic
9	Unfold the physics of boundary layer plasmas
10	Advancing PMI science and innovation
11	Engineering science innovations for plasma exhaust challenges
12	Compatibility of boundary solutions with attractive core scenarios

A timetable of activities is given below.

Date	Activity	Participants
Feb. 18	Subpanel kickoff videoconference	Workshop and subpanel leaders
March, April	Subpanel organization and conference calls as needed	Subpanel leaders and members
April 17 and April 24	Deadlines to request talk at workshop (BPO site), to submit white paper (BPO site), and to register for site access to PPPL	Community
May 4-7	Workshop on PMI at PPPL, Princeton, NJ	Community
May, June	Agree on Priority Research Directions and construct draft report	Leaders and panel members
June 30	Webinar for community feedback	Community
August 2015	Submit completed report	Leaders and panel members

While the ReNeW report was the starting point of this study, the panel considered subsequent documents from additional strategic planning documents, i.e.

- 2011 Fusion Nuclear Science Pathways Assessment report[3]
- 2012 FESAC report and white papers on "Materials Science and Technology Research Opportunities Now and in the ITER Era"[4]

- 2013 FESAC report and white papers on "Prioritization of Proposed Scientific User Facilities for the Office of Science"[17]
- 2014 FESAC report and associated white papers on "Strategic Planning"[6]

Chapter References

1 "Priorities, Gaps and Opportunities: Towards A Long-Range Strategic Plan for Magnetic Fusion Energy", Fusion Energy Sciences Advisory Committee (FESAC) Greenwald Panel Report, October 2007, http://science.energy.gov/~/media/fes/fesac/pdf/2007/Fesac_planning_report.pdf (2007).

2 "Research Needs for Magnetic Fusion Energy Science", Report of the Research Needs Workshop (ReNeW) report, June 9-13, 2009. (2009).

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13 Wright G. M. *et al.*, Nucl. Fusion **52**, 042003 (2012).

14 Zinkle, S.J. *et al.*, Nucl. Fusion **53**, 104024 (2013).

15 Burning Plasma: Bringing a Start to Earth, http://www.nap.edu/catalog/10816.html (2004).

16 White papers submitted for the PMI workshop, https://www.burningplasma.org/activities/?article=PMI Whitepapers (2015).

17 Prioritization of Proposed Scientific User Facilities for the Office of Science http://science.energy.gov/~/media/fes/fesac/pdf/2013/FESAC_Facilities_Report_Final.pdf (2013).

Chapter II

Technical Background: Science Challenges and Knowledge Gaps

II. Science Challenges and Knowledge Gaps

The intent of this chapter is to provide a technical introduction to the Priority Research Directions in subsequent chapters. As described in Chapter 1, the PMI panel was divided into four sub-panels that were charged with evaluating the "Thrusts" that were the product of the MFE ReNeW study. Each of these sub-panels identified the key scientific challenges and knowledge gaps, which are summarized in the four sections below.

II. 1. Boundary and Divertor Plasma Physics

II. 1. 1. Summary

The "boundary" of a tokamak plasma is composed of the zone of closed field lines just inside the separatrix, or last closed magnetic flux surface (LCFS), and of the open field lines, called the scrape-off layer (SOL), beyond. The "divertor" is the region beyond the magnetic X-point that accepts the majority of the heat and particle flux from the main plasma confined on the closed magnetic surfaces (Fig. II-1). Heat and particles that enter into the SOL from the main plasma flow freely along magnetic field lines. In the near-SOL zone, closest to the confined plasma, heat and particles are directed to the divertor, which must accommodate the very high heat and particle fluxes anticipated in a fusion power system. In the far-SOL zone these fluxes may impinge directly on material surfaces (e.g., first wall components, RF antenna structures) due to cross-field transport, even as they flow along the magnetic field.

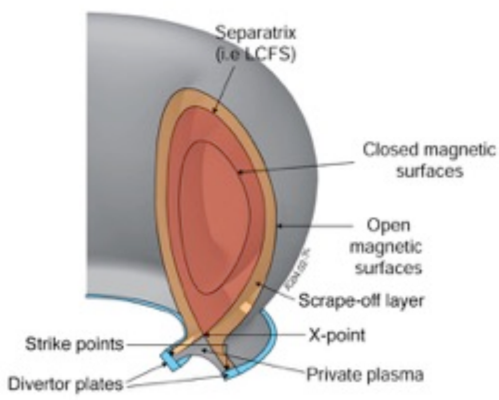

Figure II-1. Regions of a tokamak plasma (euro-fusion.org).

Our understanding of the physics of the main plasma, both in terms of thermal confinement and macroscopic stability, has made dramatic progress over decades of focused research, and is now largely consistent with the design basis for ITER. The fusion plasma physics community is confident that high confinement "H-mode" operation in ITER will be consistent with the attainment of high fusion gain, Q, and that ITER will not encounter macroscopic stability limits before it enters the high Q regime. However, the same confidence cannot be derived from the current state of boundary and divertor physics understanding. Recent advances have overturned some key prior projections, and the greatest risks for ITER are widely understood to be associated with boundary and divertor plasma physics. Furthermore, the extrapolation to DEMO will require strong innovation, rather than fine-tuning and adjustment to ITER's solutions.

In a coordinated national and international effort, enhanced experimental run time and improved diagnostics have recently been applied to measuring the power scrape-off width in the near SOL of H-mode tokamak experiments[1]. The result, which was not available at the time of the ReNeW studies, showed good reproducibility over a wide range of facilities and indicated, to the surprise of almost all researchers, that the power scrape-off width in the near SOL *does not increase with the overall size of the plasma,* and becomes narrower with increasing amplitude of the magnetic field in the poloidal direction, B_p, as shown in figure II-2. Since future devices like ITER will operate with much higher power per major radius than current experiments, and with higher magnetic field than almost all of them, this narrowing of the SOL points to an extremely high peak heat flux entering the divertor — perhaps 5x as high as previously anticipated. Calculations suggest that this will narrow the operating window for ITER[2], and very likely require a basically different, much more strongly dissipative operating scenario for DEMO, which is projected from these results to have ~ 4x higher heat flux even than ITER and a much greater operational duty cycle. It is hard to imagine that DEMO can handle the necessary heat flux with conventional divertor solutions, or that the erosion rate of solid surfaces can be kept below the required level of ~ 1mm/burn-year without extremely high dissipation.

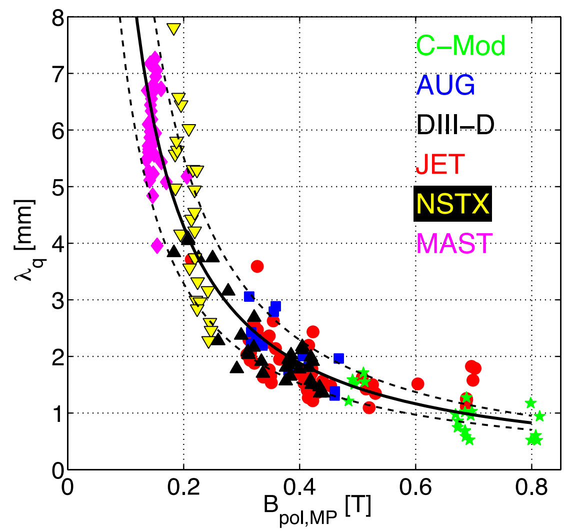

Figure II-2: Power scrape-off width vs. poloidal magnetic field from international database[1].

Research is required to understand the physical processes that can spread heat flux over a wide area of the divertor chamber, avoiding an unacceptably high peak value. These include plasma-neutral and plasma-impurity atomic interaction processes that can dissipate heat flux and plasma pressure. Among them: ionization, charge exchange, recombination and all forms of line and continuum radiation. All of these processes interact back on the divertor plasma and its intrinsic turbulence. Furthermore, the required high level of dissipation must be accomplished without introducing unacceptable levels of hydrogenic gas or impurities into the main chamber, because these have been shown to reduce the thermal insulation at the edge of the plasma that is crucial for maintenance of the high performance H mode. (See section II.4, "Compatibility of boundary solutions with attractive core scenarios.") Furthermore, enough neutral helium density must be maintained at the entrance to the chamber pumps to assure that the ash from the fusion process is efficiently removed. All of these processes are strongly influenced by the choice of the divertor and baffle configuration that controls the gas-dynamic flow of neutral gas or metal vapor to the plasma.

Although significant progress has been made since ReNeW, the physics of regimes with very high dissipation, and means for their control, are not well understood, and require major research focus. Furthermore, highly dissipative regimes become progressively more difficult to attain and control at higher heat flux, so research will be required at *heat fluxes approaching those anticipated in DEMO, which are not available on current tokamaks.*

In another research effort, again internationally coordinated, it has now been found that the field-aligned filamentary structures commonly known as “blobs,” identified before the ReNeW workshop, can carry both particles and significant localized power to the first wall of the main chamber. This unanticipated result leads to the projection of uncomfortably high heat fluxes in some regions and has driven a major redesign of ITER’s first wall components. Since much higher heat fluxes are projected for DEMO, the challenges for the future are very serious.

In addition to heat flux, a complex interplay takes place among the plasma “blob” transport to the first wall, the resulting generation of neutrals from surface recombination, and then hot charge-exchange neutral sputtering and erosion of the first wall. The impurities generated can be transported by turbulence to the main plasma, diluting the fusion fuel and impairing the key thermal insulation at the plasma edge. Furthermore, new detailed studies of the migration of eroded material around a tokamak have led to complex, unanswered questions about the build-up of redeposited materials, which can impair tokamak operation and retain tritium in the vacuum vessel.

Radio-frequency (RF) antennas and other actuators are particularly vulnerable to main chamber dynamics, and can act back on these dynamics. Attaining efficient wave propagation through a turbulent SOL plasma while avoiding damage to antenna surfaces from heat and particle fluxes from blobs and edge localized modes (ELMs) are primary challenges. In some cases significant RF power is observed to be absorbed in the SOL rather than in the core plasma. Parallel electric fields leaking from antennas can be rectified and produce strong sputtering that can be particularly problematic for high-Z wall materials.

Research is required to understand the plasma physics and RF-plasma-wall interaction physics of the far SOL, which is clearly dominated by complex turbulent transport with very high levels of fluctuation, including filamentary blobs. Analytic work, mid-scale modeling codes and most recently large-scale full-torus simulation codes are being applied to this problem. More experimental and theoretical work will be required to provide adequate understanding of the boundary plasma physics in DEMO-like regimes.

The developments discussed above, among many others, illustrate the rapid evolution of the understanding of boundary and divertor plasma physics. Despite the uncertainties, researchers have confidence that, with creativity on the part of the physics and engineering community, ITER will succeed in its mission to explore burning plasmas. Furthermore, the International Tokamak Physics Activity (ITPA), coordinated by ITER, has been instrumental in guiding global research in this area. But the path beyond ITER will certainly require deeper understanding and strong innovation.

Recent innovations include a range of United States-devised new magnetic and gas-dynamic configurations for divertors that have been conceived to spread heat flux, facilitate enhanced dissipation and ultimately support stable and nearly full detachment of the ionized plasma from the divertor surface. Initial tests are indeed very encouraging, but much more research is needed in this area. The United States has also been one of the leaders in developing the concept of liquid-metal surfaces for plasma-facing components. These can handle high steady and transient heat and particle fluxes without damage. These surfaces are projected to be self-protecting through evaporative and radiative cooling. It may even be possible to dissipate the high heat flux from DEMO in a "cloud" of metal vapor, providing a special form of fully detached operation to spread the intense heat flux.

Exciting ideas have also been proposed for controlling main chamber plasma-material interactions. These include the use of liquid metals and replenishable low-Z coatings. However this research is in its infancy. With regard to RF launchers that are used to heat the plasma and drive current, one exciting innovation is to locate these systems on the inside in major radius (the high-field side), particularly in double-null plasmas, as a means to provide immunity from blobs and turbulence, while at the same time enhancing wave access to the core plasma. Another innovation is a technique of aligning antenna straps perpendicular to the total magnetic field so as to reduce parallel electric field leakage, thereby ameliorating RF-induced sputtering effects.

These new understandings and innovations motivate Priority Research Directions B and C, which explain that greater research emphasis is required on current tokamaks, with appropriate upgrades, in the areas of divertor and boundary physics. Furthermore, there are opportunities for targeted research in these areas, complementing U.S. capabilities, on experiments abroad. Finally, it is clear that a new dedicated divertor and boundary test tokamak is required to address this key scientific issue, in parallel with ITER research on burning plasma physics. Such a machine will require both high heat fluxes to study the necessary physics, and the flexibility to accommodate tests of the newly devised innovations in magnetic configuration, gas dynamic configuration and materials.

II. 1. 2. Advancing fundamental understanding of the boundary and divertor

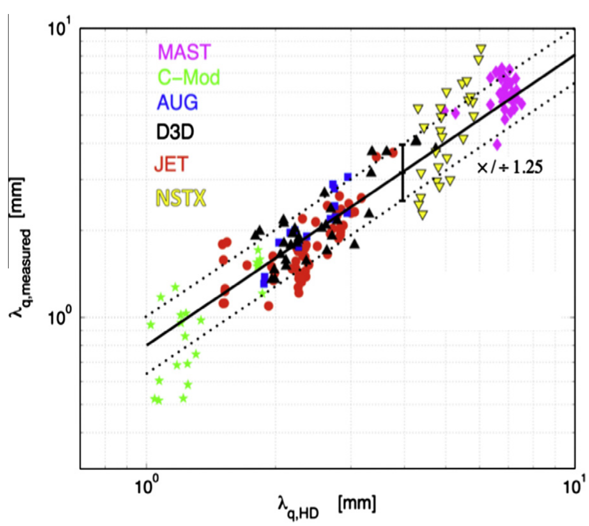

Figure II-3: Power scrape-off width vs. Heuristic Drift model ~ poloidal gyro-radius[4].

As a result of focused research efforts coordinated by the ITPA, the U.S. and international research communities have recently made remarkable progress in developing basic understandings of boundary and divertor physics. A prominent example is the robust, multi-machine empirical scaling that was developed for the near-SOL heat flux channel width in well-attached H-mode plasmas. The result, shown in figure II-2, is that this width scales about as $\lambda_q \sim 1/B_p$. If the near-separatrix region accounts for a fixed fraction of the power flow to the divertor, as is roughly the case in present experiments[1], the heat flux in the poloidal direction, q_p, varies as $P_{loss}/(R\lambda_q)$, and since this is due predominantly to parallel heat flux, we can write $q_p = q_{||}(B_p/B)$. As a result $q_{||}$ varies as $P_{loss}B/(B_pR\lambda_q)$. The heat flux that needs to be handled by the divertor is q_p (which is much less than $q_{||}$) and can be reduced by geometrical effects such as the spreading out of field lines and the tilt of the divertor plate, as well as by dissipation. On the other hand, theoretically[3] $q_{||}$ is the leading term determining the degree of difficulty of achieving a high dissipation fraction in DEMO. Using the empirical result, $\lambda_q \sim 1/B_p$, we can derive the highly sobering conclusion that $q_{||}$ varies as $P_{loss}B/R$. This implies that present experiments with, typically, peak $q_{||}$ in the range of hundreds of MW/m^2 will give way to ITER with a projected value of a few GW/m^2 and ultimately to DEMO with 10 or more GW/m^2.

An empirical scaling is not fully satisfactory for projection to the future nor, especially, for fundamental understanding. However empirical scaling is very effective for forcing realistic assessment of the future and focusing theoretical study. It is clear from dimensional analysis that $1/B_p$ is not a possible solution for a length, according to the physical laws of plasma physics, but that $T^{1/2}/B_p$ is quite possible. $T^{1/2}/B_p$ is proportional to the ion gyro-radius in the poloidal magnetic field, the so-called "ion poloidal gyro-radius." Remarkably, heuristic analysis[4] based on ion drifts and collisional electron parallel thermal conduction gives a result very close to the ion poloidal gyro-radius, and also fits the data well in both magnitude and scaling, as shown in figure II-3.

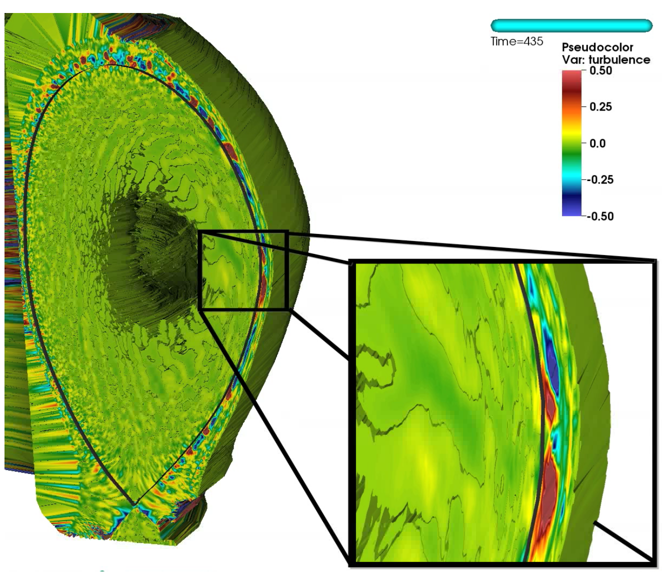

Figure II-4. Full-torus numerical simulation showing "blobs" (red and blue) in the scrape-off layer, beyond the separatrix (black curve).

It is crucial to go beyond empirical data and heuristic analysis to examine other explanations for the new measurements that are consistent with the laws of plasma physics, but not necessarily tied to the mechanisms discussed above. Leading alternatives are critical-gradient and turbulence models based on short-wavelength MHD-like activity in the boundary and SOL region. These varying scientific pictures are currently the subject of active experimental and theoretical study. Such multi-pronged scientific effort is important both for fundamental understanding and for corroborating the projections to ITER and beyond.

It is possible that the physical mechanism that gives rise to the blobs that dominate the far SOL in current experiments will play a larger role relative to the narrow "ion poloidal gyro-radius" feature in ITER and DEMO, and so will reduce the fraction of heat carried by the narrow feature. There is no clear signature of such an effect growing with size or field in current data[1], but more theoretical analysis and experimentation are required to clarify this point. Figure II-4 shows an example of peta-scale whole-device numerical modeling, applied to the boundary and divertor regions of a tokamak. Studies using this code display promising similarities to experimental measurements.

The major scientific challenge that remains, beyond extending our new understanding of the intensity of the expected heat flux towards the divertor, is to understand the means of *mitigating* intense heat flux. The obvious, but limited, means is to spread the flux geometrically along fanned magnetic field lines to meet divertor plates at grazing incidence. Crucially, we need to understand how intense heat flux can be effectively dissipated through atomic physics mechanisms required to attain the necessary extreme levels of mitigation. The interplay between these mitigation mechanisms and transport in the divertor plasma is complex and not well understood. Research on existing tokamaks with enhanced diagnostics, together with theory and modeling, utilization of scientific personnel and run time devoted to this physics, will facilitate deeper understanding. We can also anticipate that experimentation with alternative magnetic and gas-dynamic configurations, as well as alternative materials — for example high-Z surfaces and liquid metals — will contribute to the needed scientific insight.

II. 1. 3. Experimental research on existing and upgraded U.S. facilities

Experimentation in the United States has led the world in the elucidation of the advantages of advanced divertor geometries. The United States pioneered the high-Z "vertical target plate" divertor, showing its advantages for obtaining a partially-detached condition for power dissipation. This design was subsequently adopted by ITER. High-Z divertor tokamaks are presently pushing the limits of this "conventional" divertor design with respect to power handling and core/divertor compatibility, both to inform ITER and to challenge divertor physics models.

U.S. researchers have also pioneered the development of advanced divertor magnetic *topologies* – exploring them theoretically and in experiments. For example, the so-called "snowflake" divertor magnetic configuration (Fig. II-5), in which two magnetic X-points approach each other in the divertor plasma, allows substantially enhanced power dissipation relative to a horizontal-plate divertor geometry, while avoiding degradation of the core plasma[5]. A number of other configurations, such as the X-Divertor and Super-X Divertor, have been proposed, and versions of these can be studied in existing U.S. experiments[6]. Enhanced resources for divertor physics, including run time, diagnostics, theory/modeling and personnel, are needed to take advantage of these configurations. Doing so will be particularly challenging in view of the planned closure of C-Mod. It will be crucial for experimental facilities to provide detailed diagnosis of these new magnetic configurations, in support of the growing scientific understanding of highly dissipative regimes. This will allow clear, scientific comparisons to be made with conventional configurations.

Experimentation in the United States has led in the study of various low-Z wall coatings in H-mode tokamaks. Both boron and lithium have been shown to reduce impurity levels and hydrogenic recycling. Such coatings substantially improve main plasma energy confinement as well. Capillary porous structures that hold liquid lithium contain it well and survive transient events such as ELMs and disruptions. The lithium contamination levels are extremely low in plasmas operated with their strike points directly on a lithium-filled capillary-porous structure. These results need to be extended to devices with underlying high-Z walls, and compared with operation in such devices without lithium. The United States has also contributed substantially to the study of solid material erosion and migration; this work needs to be extended to both liquid-metal and high-Z wall conditions.

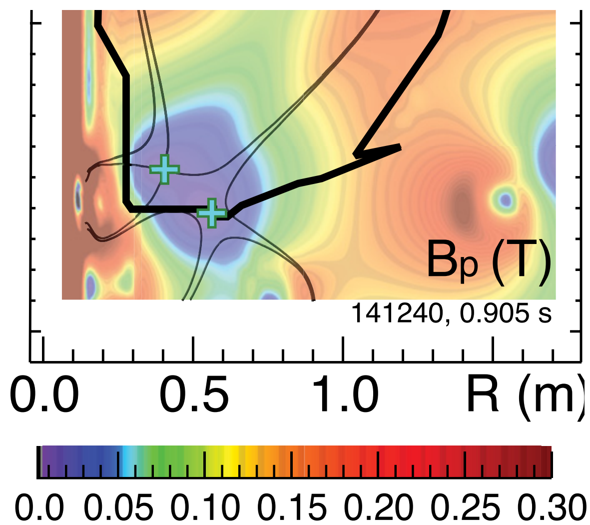

Figure II-5: Snowflake divertor; colors indicate strength of poloidal field[5].

There are at least three key foci for upgrades to U.S. facilities to investigate advanced divertor concepts. It will be necessary to optimize 1) the magnetic geometry and even topology of the divertor, 2) the gas dynamics that follows from the mechanical design, including divertor target, gas baffling and pumping, and 3) it will be necessary to use test facilities to develop means to introduce fresh liquid-metal (including lithium) surfaces into existing tokamaks to examine evaporative and radiative cooling and volumetric vapor shielding. Fast-flowing systems, once developed on test stands, can potentially be used to examine pumping and very high heat-flux handling.

In the area of the main chamber, key upgrades should include means to study replaceable low-Z coating, as well as slow-flowing and capillary-porous liquid metal surfaces. It is highly desirable, if very challenging, to test concepts for high field-side launch of ion-cyclotron range of frequency (ICRF) and lower hybrid waves for heating. This will build on the success of recent experiments with specially field-aligned RF antennas.

As detailed in PRDs B&C, there are multiple exciting opportunities to expand research in boundary and divertor physics on existing and upgraded experiments. While these can be complemented by research abroad, a significant new confinement facility will be required to advance the field and to sustain U.S. leadership in this area.

II. 1. 4. Research abroad

Research in Europe has recently focused on developing tokamak operation with tungsten divertors for ITER. One device has completed a changeover from carbon to tungsten divertor and main-chamber plasma-facing components; another has implemented ITER-like material choices, with a tungsten divertor and beryllium first wall. These alterations have had profound effects on plasma behavior. Such effects include order of magnitude reduction in fuel retention; necessity to employ low-Z impurity seeding to mitigate divertor heat fluxes; and necessity to mitigate high-Z contamination of the core plasma through central heating and/or low-Z wall coatings. These experiments are of very considerable interest, and the United States should participate under an expanded overall effort in the area of boundary and divertor physics. This work would build on U.S. experience with high-Z walls and would provide a baseline for comparison with advanced magnetic and gas-dynamic configurations and alternative materials, both liquid and solid. For example the all-tungsten experiment has recently published important results on full detachment[7]. Europe has a program focused on the Super-X divertor and another one on liquid metals.

The long-pulse machines in Asia complement the short-pulse capabilities of U.S. machines. The United States should certainly take advantage of these facilities to address key long-pulse boundary and divertor physics issues that cannot be studied on U.S. experimental facilities. These will include such topics as the effects of high-Z wall equilibration with plasma recycling, and the effects of long-term erosion, material migration/redeposition, and ultimately management of the "slag" that results from continuous redeposition of eroded material. Long-term control of dissipative scenarios should be especially challenging and scientifically fruitful. One of these Asian devices is already experimenting with lithium, so it provides a particularly promising opportunity for collaboration on this topic.

It should be recognized that overseas collaboration is not a substitute for domestic research since U.S. scientists cannot set priorities and guide research directions. The United States can be most effective if our researchers both lead on our own facilities and bring our special expertise to focused research on facilities abroad.

II. 1. 5. New facility for boundary and divertor research

Present experiments can obtain DEMO-relevant conditions in the vicinity of the divertor plate and perform important research on dissipative processes. But these experiments cannot achieve upstream parameters of plasma pressure and heat flux approaching those of fusion power systems. Such parameters are required to resolve the key issues for a dissipative divertor in DEMO, given the complex nonlinearity of highly dissipated divertors and the requirement, dictated by atomic physics, for absolute plasma parameters approaching those of DEMO. Present devices also lack the flexibility to provide high-power-density tests of alternative magnetic and gas-dynamic configurations, and cannot vary solid and liquid plasma-facing materials. In the judgment of the panel, an experiment with both DEMO-relevant upstream heat flux and pressure, together with high divertor and first-wall configuration and material flexibility, will be needed to proceed to DEMO with scientific confidence.

Already at the time of the ReNeW activity in 2009, the PMI community called for a new dedicated very long-pulse facility to study advanced boundary and divertor physics. The community recommended that the United States "*develop design options for a new facility with a DEMO-relevant boundary, to assess core-edge interaction issues and solutions. . . .Develop an accurate cost and schedule for this facility, and construct it*." Since that time, the European Road Map to the Realization of Fusion Energy has also identified this research area and such a facility as critically important, "*The risk exists that the baseline strategy pursued in ITER cannot be extrapolated to a fusion power plant. . . . Since the extrapolation from proof-of-principle devices to ITER/DEMO based on modeling alone is considered too large, a dedicated test on specifically upgraded existing facilities or on a dedicated Divertor Tokamak Test (DTT) facility will be necessary."* A working group in Europe is now investigating this option, but there is no assurance that Europe will move forward with a DTT facility. Indeed there may be opportunities for collaboration.

Recently a new high-power-density divertor test tokamak facility has been analyzed in the United States.[8] It features long divertor legs and a flexible poloidal field configuration, along with flexibility in gas dynamics and in the use of solid and liquid plasma-facing materials. At the time of ReNeW, two concepts for a boundary and divertor physics machine were under consideration,[9,10] both with high power, long pulses and hot first walls. This new short-pulse concept adds considerably to the range of options available for consideration. A national working group should be established to develop options for a United States-led divertor test tokamak.

The European Roadmap argues that control of boundary and divertor physics is "*probably the main challenge towards the realization of magnetic confinement fusion.*" The United States has the opportunity to be the world leader in this area, and should seize the opportunity.

PRD B and PRD C examine in more detail the physics of the divertor and plasma boundary, respectively, and provide detailed recommendations on research action plans in these areas.

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II. 2. Plasma-Surface Interactions

II. 2. 1. Introduction

The simplified view of plasma-material interactions involves particles escaping the core plasma and striking the wall. These particles can include highly energetic neutrons that penetrate deep inside the wall structure, and energetic ions implanted near the surface as a result of the efficient electronic stopping power. There is a certain probability that the collision of energetic ions with wall atoms will result in the release of an atom of the wall material, which release is defined as sputtering. This sputtered atom can then become an impurity in the core plasma.

In reality, the situation is considerably more complex, as shown in Figure II-6. Incoming energetic particles escaping from the burning plasma include fuel ions, neutral gas atoms, helium ash, and singly and multiply charged impurity ions. Each of these can unleash a variety of processes at the surface of the material: implantation, reflection, physical sputtering, chemical erosion, electron emission, to name just a few. Particles released from the surface can undergo a series of reactions with the plasma that is in contact with the material. The wall material, too, undergoes drastic changes due to the large amount of incoming particles that imbed in the surface. These implanted particles damage the surface, creating vacancies, interstitial atoms, dislocation loops, etc., and can produce amorphous surface layers in some materials. The embedded particles can then react chemically with other atoms in the surface, diffusing and coalescing to form bubbles or blisters.

Each of these processes operates on different temporal and spatial scales, further complicating the picture. The resulting reconstituted surface often has little resemblance to the material that was originally designed for use in that particular location near the edge of the burning plasma.

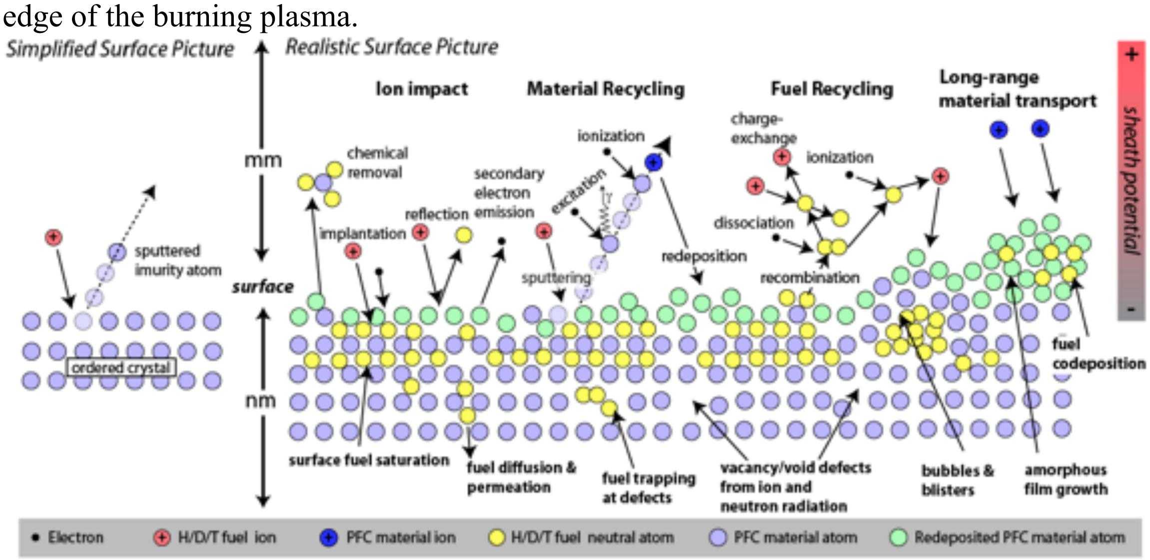

*Figure II-6: **I**llustration of the complex, synergistic, and multi-scale surface interactions occurring at the plasma–material interface in a realistic magnetic fusion plasma environment. H, hydrogen; D, deuterium; T, tritium; PFC, plasma facing component; γ, gamma ray.*

The grand challenge of gaining a physical understanding and establishing a predictive modeling capability in the field of plasma-material interactions requires that such complex and diverse physics, which occurs over a wide range of length (Ångströms to meters) and time scales (femtoseconds to years) be addressed simultaneously. The plasma and the material surface are strongly coupled to each other. The characteristics of the incident plasma are governed to a large extent by properties of the material surface such as recycling and erosion. Yet both the material and its properties evolve as a result of the plasma exposure, the feedback of which leads to an evolution of the incident plasma characteristics. Figure II-7 illustrates some of the interrelated processes involved in plasma-material interactions, and the multi-scale nature of the spatial and temporal variations involved.

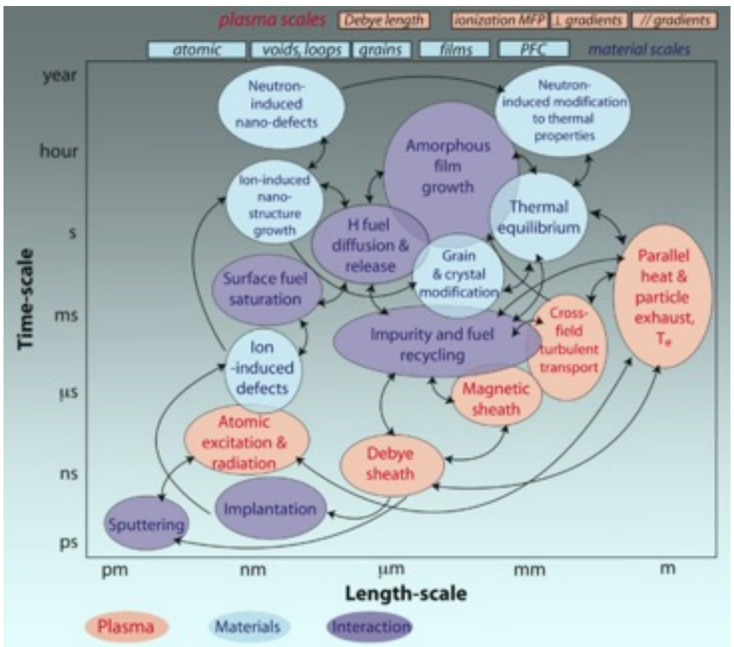

*Figure II-7**:** Graphical representation of the multiple time and length-scales involved in the inherently coupled processes and phenomena that dictate plasma materials interactions in the boundary plasma region of magnetic fusion devices. Processes occurring within the plasma are denoted in light red, while those in the near-surface and bulk materials are in light blue. The important plasma–materials interactions are identified in light purple.*

The multi-scale nature of the physical processes involved poses challenges to the modeling and experimental characterization of both the individual and coupled processes. The multitude of time- and length-scales controlling material evolution and device performance requires the development not only of detailed physics models and computational strategies at each scale, but also the development and implementation of quantifiable diagnostic techniques that will allow robust and vigorous testing and validation of the models' predictions. Figure II-8 represents the suite of existing models that must be coupled to develop an understanding of the plasma-material interface. Information flows up from atomistic-based methods towards the continuum calculations, beginning with ab initio calculations (VASP) based on electron wave functions and eventually leading up to fluid-based continuum models (SOLPS) of the edge plasma.

Information from each scale informs calculations based in other scales through an information-passing, scale-bridging hierarchy. At each of these scales, experimental measurements of the reconstituted surfaces and their properties must be designed to provide validation of the calculations, and offer guidance to each stage of the multi-scale modeling approach.

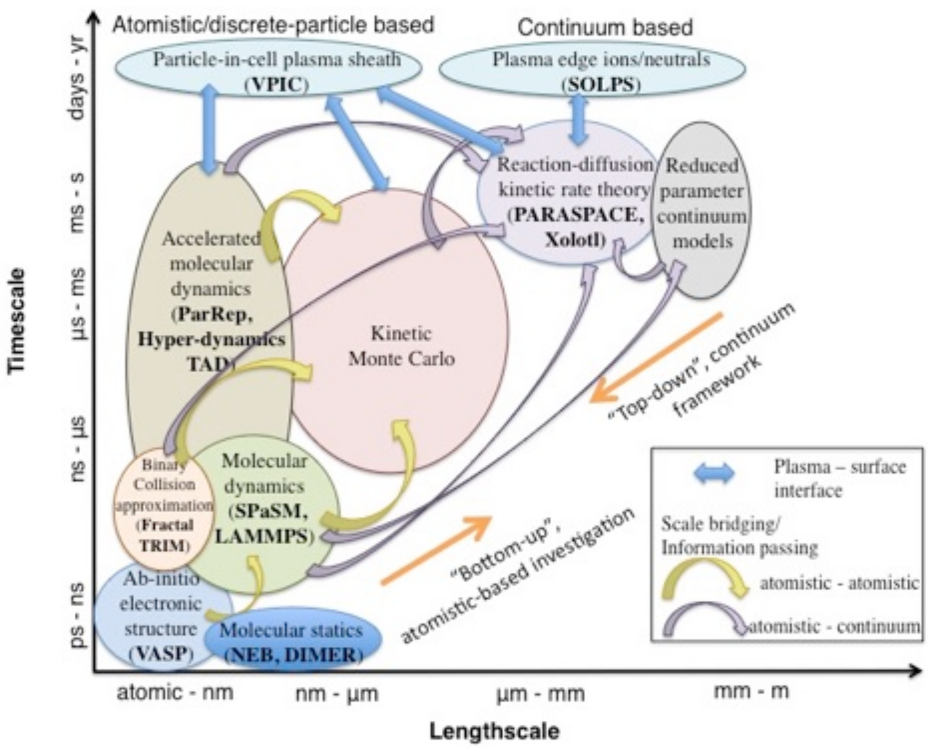

Figure II-8: Illustration of a multi-scale modeling approach for plasma-surface interaction, including the names of some of the well known codes operating at each scale (in bold)

II. 2. 2. Enhanced experimental and theoretical capabilities

Since the time of the ReNeW report[1], the scientific understanding of plasma-material interactions (PMI) has grown considerably, but remains far from mature. Many of the recommendations from ReNeW for emphasizing effort in this area are being pursued, both in the domestic U.S. program and internationally. However, many outstanding issues exist and continued emphasis of this field of research will be required on the path toward building a burning-plasma confinement device.

The international community, in particular, has emphasized research in plasma-material interactions by investing heavily in the construction of several new devices and by upgrading other facilities and continuing to plan further work in this area. In the European Union, the JET device has converted its plasma-facing material from an essentially carbon-dominated wall to a material mix that matches that planned for ITER, including a beryllium main chamber wall and a tungsten divertor, JET ITER-like Wall (ILW)[2]. The Tore Supra device is also being reconfigured from a machine containing carbon plasma-facing components to a diverted all-metallic plasma-facing component machine, WEST[3]. The European Union has also constructed new linear plasma facilities: MAGNUM-PSI[4] and PSI-2[5]. The MAGNUM-PSI device, in particular, expands the operational regime accessible to laboratory plasmas to operations with a plasma ion flux above 10^{24} m^{-2} s^{-1}. In Asia, two superconducting tokamaks have achieved H-mode operation, EAST[6] in China and KSTAR[7] in South Korea. Japan is also embarking on construction of a superconducting tokamak, JT-60SA[8] and Germany is on the verge of making its first plasma in the superconducting stellarator Wendelstein 7-X[9]. These devices offer tremendous promise for long-pulse operation with relevant heating scenarios in the future.

The U.S. Program has been less ambitious in terms of construction of new devices, although there are proposals awaiting approval for construction of a new high-power linear plasma device[10] and a hot-walled high magnetic field confinement device[11]. ReNeW also recommended upgrading existing facilities. The U.S. Program has invested in upgrades to existing facilities and increased run time for PMI based experiments on its toroidal facilities.

One of the keys to understanding and controlling PMI is the ability to collect data on the evolution of the materials surrounding the confined plasma. Ideally, one would like data obtained during, or at least after, each individual shot. In addition, since the plasma's contact with the wall varies significantly poloidally (and possibly toroidally as well) as one moves from the divertor to the first wall, one would like data from a variety of locations within the vessel. To date, the majority of the data from confinement devices are based on the removal of samples during periodic machine shutdowns. Unfortunately, these samples tend to have an archeological aspect, since the samples accumulated information from a variety of different exposure conditions and experiments during operation of the confinement facility.

Recently, the Alcator C-Mod team has developed an ion beam diagnostic (AIMS[12]) that can be used between discharges to monitor the evolution of plasma-exposed surfaces. This diagnostic, shown schematically in Figure II-9, does not require the removal of material from the tokamak. Another unique feature of this diagnostic is that the toroidal magnetic field of the tokamak can be used to steer the ion beam to a variety of surfaces within the vessel, allowing for interrogation of the spatial variations across the surfaces. Other toroidal devices have installed "surface science stations" that allow for extraction and analysis of samples without exposure to air. These systems provide insight into the chemical bonding nature of, and gas retention in, plasma-exposed samples. All of these impressive diagnostic capabilities are used to help validate material migration and surface physics models that are being developed.

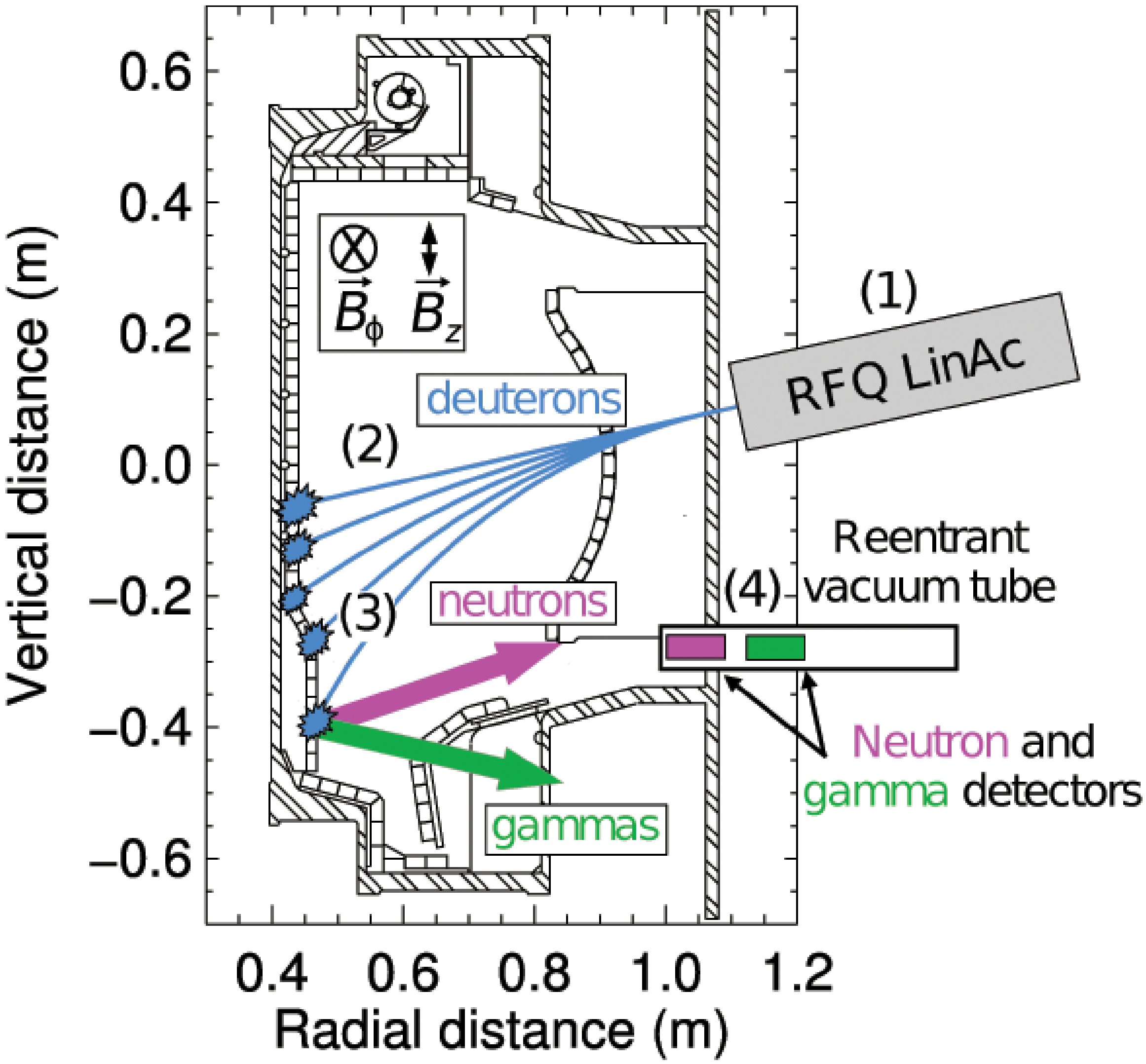

Figure II-9. Schematic representation of at AIMS diagnostic on Alcator C-Mod

Linear plasma devices in the U.S. Program have also seen investments and upgrades. A tandem accelerator is coupled to the steady-state plasma in DIONISIS[13], allowing real-time measurements of changes in surfaces exposed to plasma. In PISCES[14], lasers have been installed to study changes to surfaces resulting from a series of repetitive transient heating pulses during plasma exposure. Finally, in-situ PMI diagnostics, such as laser-induced breakdown spectroscopy (LIBS), are under development to measure real-time compositional changes due to prolonged plasma interactions with surfaces.

These improved experimental capabilities are leading to more thorough validation of an array of models, from global material migration models down to electronic density functional theory, including all the spatial and temporal scales in between. The United States has promoted the advancement of multiscale modeling activities that attempt to couple the output from smaller-scale models as input to the next larger-scale models. Each of these scale models is in turn coupled to appropriate experimental techniques to ensure that the models capture the necessary science for insight into the processes involved and extrapolation to other scenarios.

II. 2. 3. Increased understanding of PMI science

In some cases, the investment in increased capabilities has led to a deeper understanding of the science underlying the interaction of plasma with materials. In other cases, the investments have highlighted how far away we still are from a predictive capability with respect to the global importance of the surrounding materials on core plasma performance. A classic example of this is evident in the switch of the plasma facing materials in JET from carbon dominated to a mix that mimics the material selection for ITER, i.e. beryllium and tungsten. In 2013, ITER made the decision to remove carbon from its divertor and begin operation with a full tungsten divertor. The JET-ILW configuration became the ideal location to test the ITER material mix. The JET-ILW experiment showed a reduction in accumulation of fuel species retained within the vacuum vessel due to co-deposition, as predicted[15]. Measurements of material deposition patterns were used to verify prediction of global material migration models and co-deposits recently collected from the surfaces of the device are being used to validate models of the thermal release behavior of hydrogen isotopes from these co-deposits. The validation of these models has increased the accuracy of tritium accumulation estimates for ITER and provided an increased confidence in those predictions.

However, operation of the JET-ILW experiment also revealed an unanticipated reduction in core plasma performance. While this reduction is still not completely understood, it is believed that the change in recycling at the wall alters the profiles of both the fuel and impurity edge plasma in the vicinity of the H-mode transport barrier and reduces the plasma pressure at the top of the H-mode pedestal. The overall energy confinement appears to increase nearly linearly with the H-mode pedestal pressure. Because of these unexpected results, the JET-ILW experiment provides an excellent platform for furthering our understanding of the role of the strong coupling between the physical wall and the confined core plasma.

The edge plasma profiles also determine the behavior of transient power and particle expulsions, e.g. ELMs, from this region toward the wall. A great deal of knowledge has been obtained since ReNeW in the understanding of how the periodic and repetitive nature of ELMs affects the materials facing the plasma. Originally, it was thought that as long as one avoided exceeding the energy density necessary to melt the material (~ $1MJ/m^2$ for tungsten), the material would survive with little alteration of the surface. It is now understood that fatigue plays a major role in the response of plasma-exposed surfaces to transient power loading. As the number of repetitive pulses onto surfaces increases, the energy density limit for preventing surface deformation decreases. It is now understood that the thermal expansion associated with localized surface heating and cooling induces stresses in the surface that lead to roughening of the surface after many transient heating events. This roughening eventually results in small-scale cracks in the surface, which in turn leads to larger-scale cracking and eventually degrades the power handling capability of the material. This decrease in material performance has been observed in high heat-flux testing both in electron beam facilities and in repetitive transient plasma pulsing of targets in linear plasma devices. The impact of these results points clearly toward the need for developing new materials, manufacturing processes, or even liquid plasma-facing components, that result in surfaces that are more resistant to fatigue degeneration.

Extensive studies of surface changes have also focused on targets exposed to high-fluence quiescent plasma. Models for the behavior of gas atoms within materials have evolved to the point at which comparisons between theoretical and experimental results are now possible. Perhaps the highest-profile work involves the behavior of hydrogen and helium atoms within a tungsten lattice. Ab initio calculations have resulted in the understanding of the interaction of both hydrogen and helium with vacancy defects in tungsten. The ability to predict the binding energy of gas atoms with the tungsten lattice has enabled better interpretation of measurements of retention in tungsten due to plasma bombardment. The interplay between hydrogen and helium both competing for trapping sites has also been investigated with density functional theory (DFT) and this has led to the development of potentials used in Molecular Dynamics (MD) simulations of plasma-exposed tungsten surfaces.

Precipitation of helium atoms into small clusters and eventual growth of these clusters into small bubbles has been observed in MD calculations. The dynamics of the nucleation of helium nanobubbles in tungsten observed experimentally (see Figure II-10) supports these near-surface models. Comparisons of the three-component model (tungsten & hydrogen & helium) with experimental results are now being attempted through the development of the next scale of the models using the input from the MD simulations. The development of accelerated MD and kinetic Monte Carlo codes offers the possibility to predict the evolution of surfaces during high-flux plasma exposures and eventually compare these predictions with experimental observations.

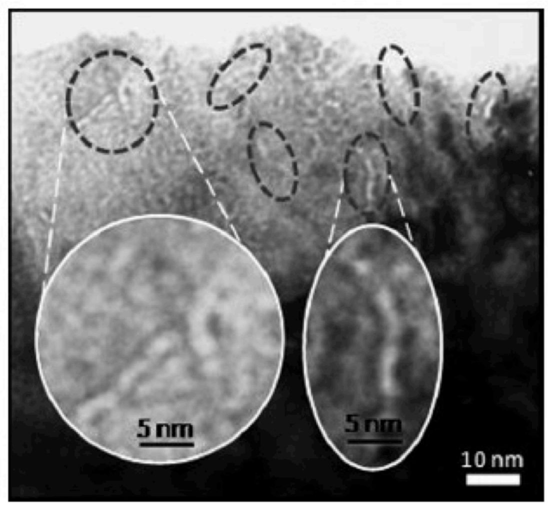

Figure II-10: Transmission electron microscope image showing the formation of helium induced nanobubbles in the surface of tungsten exposed to deuterium-helium mixture plasma[17]

An area in which the development of these mesoscale models will clearly add value lies in the understanding of experimental observations of nano-structured surfaces that have been observed to form on a variety of metals during high-fluence plasma exposure. The classic example of this effect is tungsten nanostructures (so-called tungsten fuzz) that form on high-temperature tungsten surfaces exposed to helium ion bombardment (e.g. Fig. II-11). The properties of fuzzy tungsten surfaces have been studied extensively in linear plasma devices, and the formation of such surfaces has been observed in a confinement device when the conditions are right, yet no fundamental understanding of the formation mechanism exists. The advent of models capable of simulating seconds, minutes and hours of plasma interactions with surfaces offers promise of understanding a host of PMI issues.

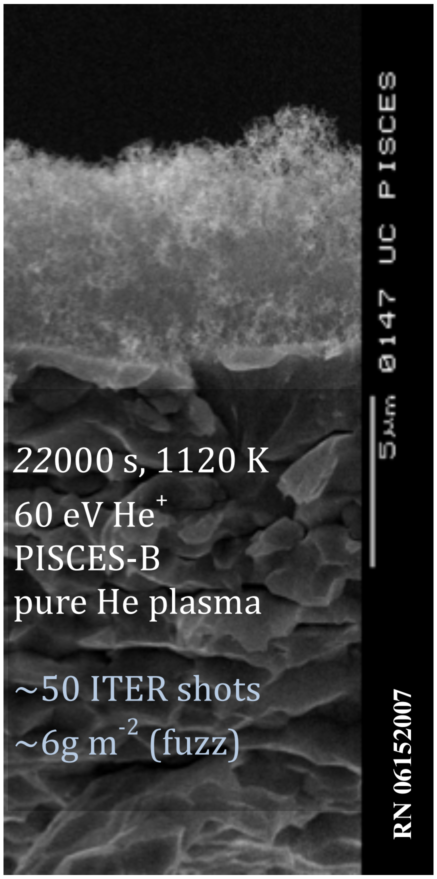

Figure II-11: A secondary electron microscope image of a tungsten surface turned into "tungsten fuzz" by helium plasma exposure[18]

The predictive understanding that is achieved by modeling the fundamental response of the individual atoms in material surfaces to energetic particle loads can successfully be used to interpret data measured within the core of confinement facilities. One classic example of this successful coupling of MD modeling with confinement experiments can be seen in understanding the science behind the sputtering of beryllium-deuterium molecules from surfaces. The behavior of this sputtering was first characterized in linear plasma devices. Subsequent modeling of deuterium bombardment of beryllium surfaces using MD determined that the release of beryllium-deuterium molecules was due to chemically assisted physical sputtering. While this process had been known to occur in carbon-based materials, it was previously unknown in metallic surfaces. MD was used to predict trends in beryllium-deuterium sputtering that were later verified using a linear plasma device, and these coupled modeling-experimental results were eventually used to understand the accumulation of beryllium impurity ions within the core of the JET-ILW plasma.

Similar simulations of the fundamental behavior of materials are being used to investigate the hydrogen mobility and trapping in tungsten. These models allow scientists to evaluate the behavior of neutron-damaged materials in future burning- plasma confinement devices, and to validate different experimental techniques capable of simulating fusion-grade plasma conditions that are presently not obtainable. The Tritium Plasma Experiment (TPE[16]) at Idaho National Laboratory (INL) has performed the first plasma exposures of neutron irradiating tungsten, and the results of retention studies of these samples are being compared with materials that have been damaged using energetic-ion beams. Ion beam damage has been used extensively to simulate neutron damage for a wide range of measurements, including those made in retention studies and studies of changes in thermal conductivity and hardening of materials. These basic science studies will be used to create a database of material properties that will be needed for any design efforts for future nuclear fusion devices.

It is well known, from research into fission materials, that neutron irradiation will lead to degradation of the thermo-mechanical properties of materials. Coupling this degradation with the increased heat loads expected from DEMO-class confinement machines leads to legitimate questions as to whether any solid surface can survive for long in such an environment. The leading alternative concept is a flowing liquid metal surface for the plasma-facing material. Such surfaces have the potential to address heat removal concerns, resist damage from transients and reduce concerns arising from neutron-irradiation effects. For these reasons, plasma-interactions with liquid surfaces have continued to flourish since the ReNeW report was published in 2009.

Lithium-coated surfaces are used in many confinement devices and have been shown to influence the behavior of ELMs in the National Spherical Torus Experiment (NSTX) and been used in the EAST tokamak in China to enable H-mode operation. Scientific investigations into material loss from, and vapor shielding of, liquid surfaces in contact with high-flux plasma are also continuing in linear plasma devices.

The fundamental physics understanding of the thermoelectric magnetohydrodynamics force in a heated, conducting liquid has thus enabled the development of a free-flowing liquid surface that has been deployed in a confinement device as a proof-of-principle demonstration of a flowing liquid-metal divertor component. The technology necessary to recirculate and purify the flowing liquid has been proposed as the next step in the development of this alternative plasma-facing material solution.

In summary, progress has been substantial in understanding the plasma-material interface and the changes that occur to materials when subjected to the extreme conditions near an energetic plasma. The combination of controlled experimentation in off-line devices, development of fundamental science-based models, and observations in actual plasma confinement facilities has proven successful in advancing the scientific basis for PMI research. The following section will describe the primary unsolved PMI questions and propose a series of actions that can be followed to achieve continued success in answering outstanding PMI issues.

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Section II. 3. Plasma Facing Components

II. 3. 1. Introduction

Plasma Facing Components (PFCs) such as the first wall and divertor, and special PFCs such as RF launchers, Faraday shields and ECRH mirrors, will need to operate at higher temperatures and handle high particle and heat loads. These are highlighted in Fig. II-12 in a pre-conceptual power plant design. The development of actively cooled PFCs required for the extreme and multi-physics environment seen by these components in future facilities such as the Fusion Nuclear Science Facility (FNSF) and the DEMO is a critical path topic for fusion research.

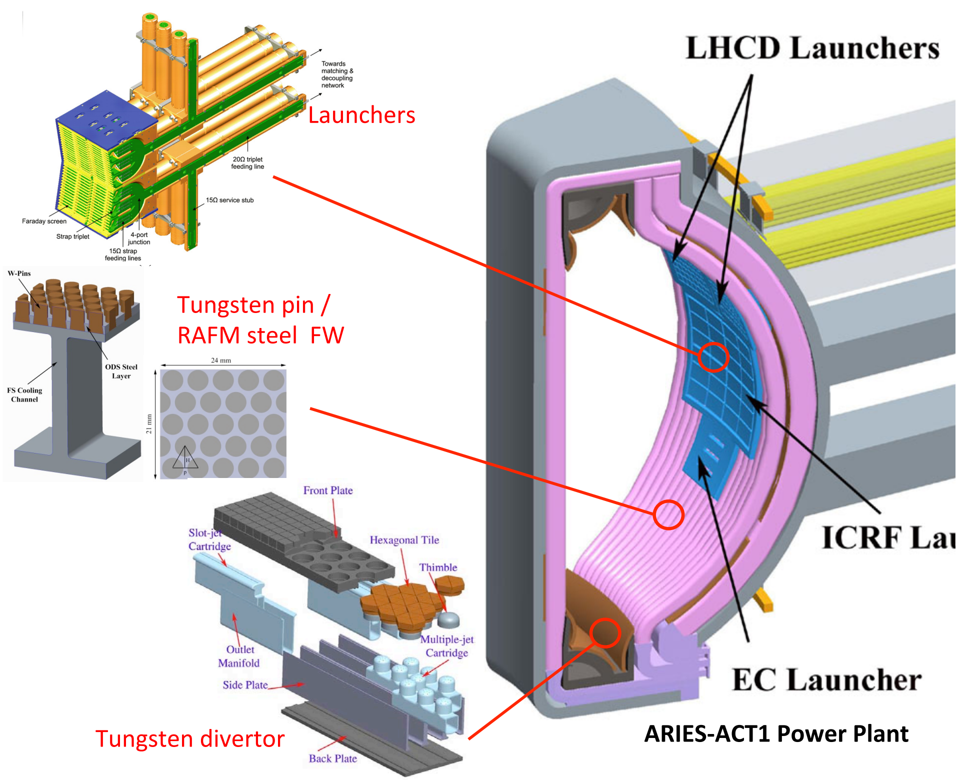

Figure II-12: Cutaway view of the ARIES-ACT1 pre-conceptual power plant design, highlighting the main plasma facing components, the first wall (FW), the divertor, and an RF launcher. Each of these must endure the neutron and plasma loading for very long durations. For the whole device, the FW-type structure would occupy ~ 75-80 percent of the plasma-facing surface, the divertor ~ 15-20 percent, and the special PFCs ~ 5 percent.

II. 3. 2. Research progress since the ReNeW report:

Important steps toward qualified integrated plasma facing components have been obtained for ITER, for the divertor[1] and the first wall, and are shown in a flow diagram in

Fig. II-13. Extensive material development, testing, and manufacturing activities over more than a decade, mostly outside the United States, have resulted in a high-performance divertor component consisting of tungsten monoblocks joined to water-cooled swirl tubes and hypervapotron heat sinks in roughly half of the beryllium-clad ITER first wall. Although all these materials and coolants used in ITER are not long-term relevant for fusion, the research and development process is similar. Manufacturing a reliable divertor, first wall, and in-vessel components will require identification of the loading features (steady state and simulation of off-normal events); identification/development of viable materials and their properties; development of advanced joining techniques for use between plasma-facing armor and actively cooled heat sinks; and scale-up strategies for advancing from small to full-size components. The testing for qualification of PFCs under high heat flux both steady and transient (cyclic), and very high heat flux/short duration (ELMs, disruptions) pulses, and low DPA neutron irradiation, which provided a path to reliably performing/manufacturable divertor and first wall items. Qualification of materials for devices beyond ITER will involve the additional aspects of high fusion-neutron damage measured in displacements per atom (DPA); very long plasma durations; higher operating temperatures; requirements for tritium breeding and thermal conversion to electricity; and advanced neutron irradiation resistance materials and coolants.

	component design	material qualification	joining techniques	mockups	neutron irradiation	prototype
divertor	monoblocks flat tiles (macrobrush)	armour: CFC and W heat sink: DS-Cu CuCrZr	CFC/Cu: laser texturing -beam HT-brazing HIPing W/Cu: casting HT-brazing HIPing	approx. 100 small scale components qualification: ultrasonic thermography HHF-testing: ≥ $^{-2}$, ≥ 1000 cycles	materials and components T_{irr}: 200 – 700°C fluence: 0.1 – 1 dpa (PIE plus in-pile testing)	medium scale VT full scale VT full scale dome
first wall	flat tiles (shield block design with poloidal coolant channels)	armour: beryllium heat sink: DS-Cu CuCrZr	Be/DS-Cu: HT-HIPing HT-brazing Be/CuCrZr: LT-HIPing fast brazing DS-Cu/SS, CuCrZr/SS: HAT-HIPing	approx. 50 small scale components qualification: ultrasonic HHF-testing: ≥ 2.5 MWm^{-2}, ≥ 30.000 cycles	materials and components T_{irr}: up to 700°C fluence: 0.3 – 1.0 dpa (PIE plus in-pile testing in preparation)	several full scale primary first wall components

Figure II-13. Flow diagram of the development/qualification program for the tungsten/CFC divertor PFCs, and the Be and CuCrZr FW PFCs for ITER, demonstrating the evolution of single material test through integrated component test, with progressively more prototypical loading conditions.

As an alternative to water, helium cooling has many advantages in a nuclear system due to its inherently safe, inert chemical properties, lack of corrosion, vacuum compatibility, single-phase heat transfer without the possibility of a critical heat flux (CHF) excursion, lack of neutron activation, and easy separation from tritium. Helium coolant is the primary candidate for most power plant designs, and would apply to both solid PFCs and the substrate for liquid metal PFCs. Most importantly for DEMO and commercial fusion power, helium can be used at higher temperatures, and provides the potential to access very high thermal conversion efficiencies. Since helium has a low thermal mass, ρC_p, it requires the use of high mass flow rates, implying higher densities and operating pressures and greatly enhanced heat transfer area and turbulence promoters for efficient heat transfer. Tremendous progress occurred in this regard prior to ReNeW, with exceptional heat transfer demonstrations[2-5]. Since the ReNeW report, this work has continued, with the European Union demonstrating a nine-module assembly of a helium-cooled modular jet design[6-8]. In the United States, work at Georgia Institute of Technology[9], examined jet-impingement cooling with surrogate gas coolant (air) and structure (brass) using engineering scaling approaches, and now is pursuing helium with tungsten, shown in Fig. II-14.

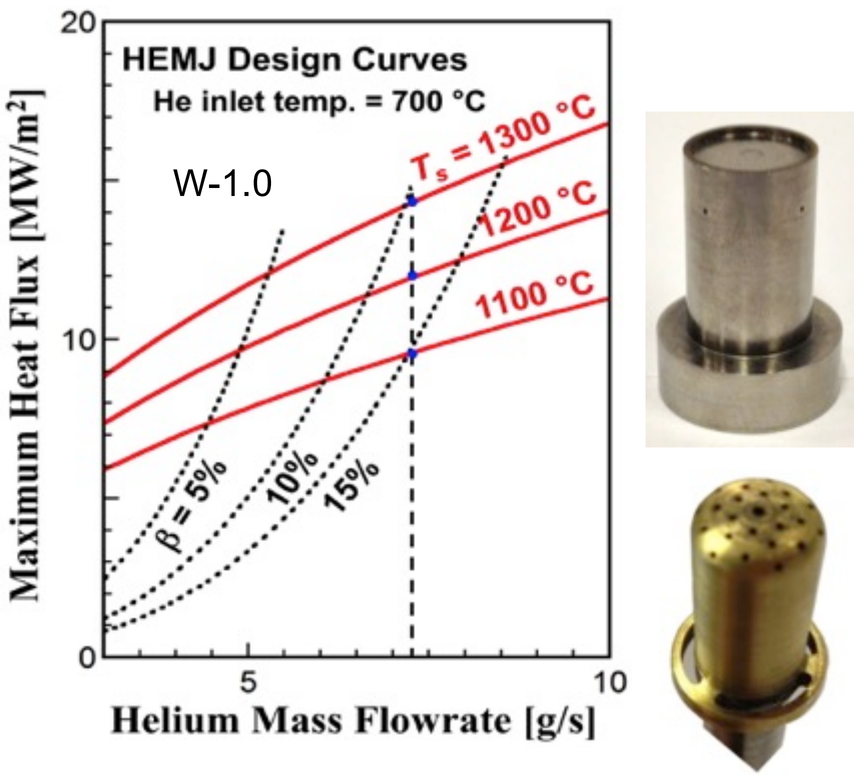

Figure II-14: The WL10 outer shell [left] and stainless steel inner cartridge [right] for the single HEMJ module test section tested in the Georgia Tech helium loop. The base of the outer shell has a diameter of 25.4 mm. The graph shows the maximum heat flux as a function of the helium gas flow rate, for a given surface temperature, along with the associated ratio of pumping power to thermal power removed (β).[12]

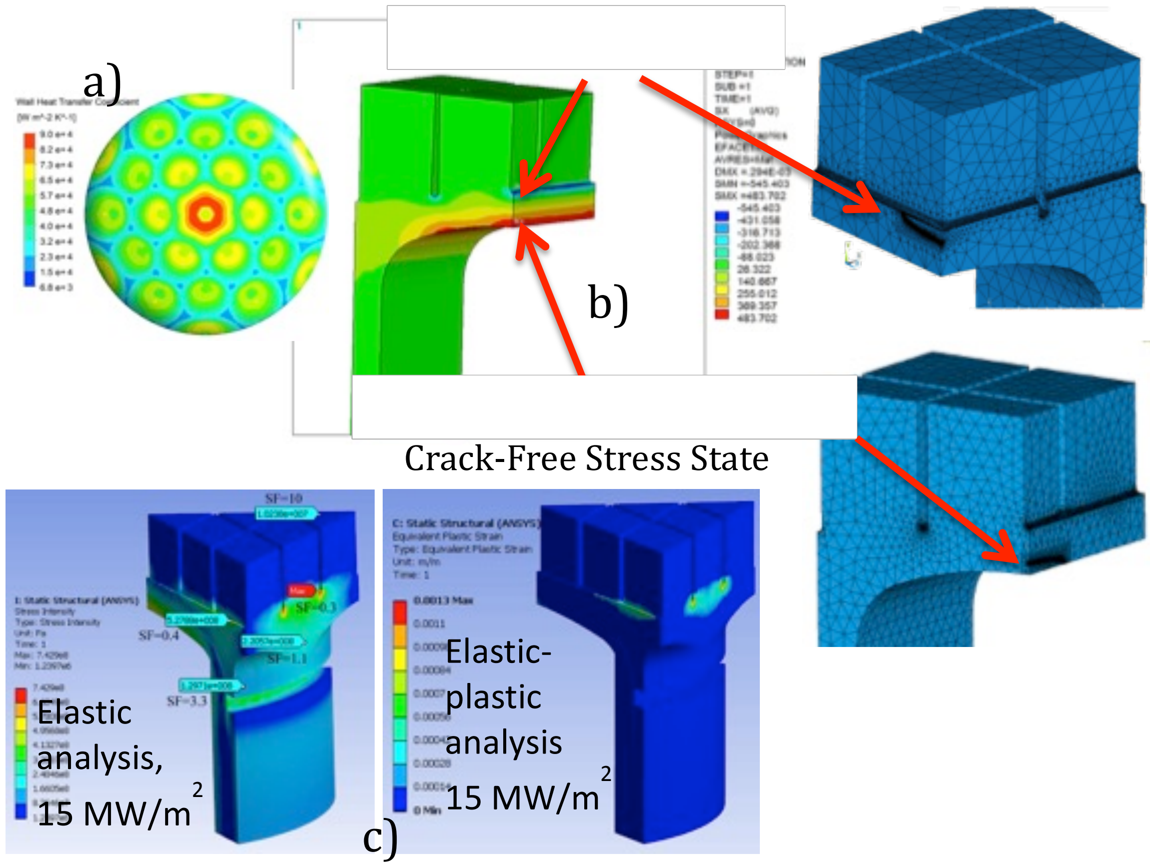

Figure II-15: Analysis performed for a tungsten finger high heat flux divertor in pre-conceptual power plant study, ARIES-ACT1. The computational fluid dynamics (CFD) in a) shows the heat transfer coefficient for the jet impingement design on the surface where the maximum heat is removed; fracture mechanics analysis b) included detailed crack representation at the maximum stress locations in hot and cold states; and thermo-mechanics c) was done both with elastic and elastic-plastic treatments, showing the latter provides stress relief with no impact on component performance or lifetime.

These studies provided both experimental results and computational analysis for comparisons, and contributed to the design of ARIES power plant studies[10], which are shown in Fig. II-15, with computational fluid dynamics, thermo-mechanics and fracture mechanic analysis. Recent simulations identified the tolerable ELM size to avoid melting for a given inter-ELM heating level, and fracture mechanics for the solid tungsten divertor design have shown that the divertor may survive the thermal cycle of operation and shutdown. However, this has not included ductility loss from irradiation. Thermal creep, which is a failure mechanism of structures at high temperature and stress for extended periods, can also be a significant issue for a tungsten divertor[11].

At the time of ReNeW, U.S. industry was instrumental in developing low-cost, near-net-shape fabrication techniques for our small helium heat sink mock-ups shown in Fig. II-16 for post-ITER first wall, divertor and ion cyclotron (IC) applications. The technology was evolving to include large area (0.3m x 0.3m), multiple channel components with integrated manifolds and a minimum of joints. Several multi-channel devices, shown in

Fig. II-17, were tested in 2010. However, the technology remained in the Small Business Innovation Research program, was subjected to budget constraints and was not incorporated into the broader U.S. PFC base program. HEMJ jet technology was tested[12] directly using the helium loop at Georgia Tech in 2013.

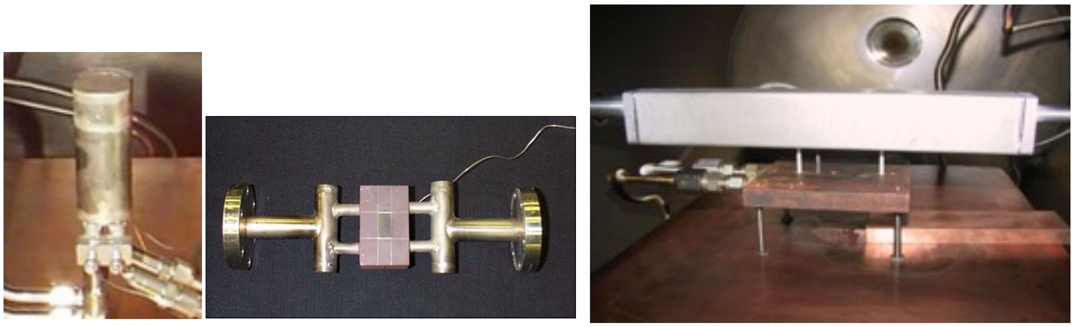

Figure II-16: All-tungsten porous media heat sink (left)[2], all-copper porous media heat sink (middle)[3], all-moly porous foam tee-tube heat sink (right)[4]

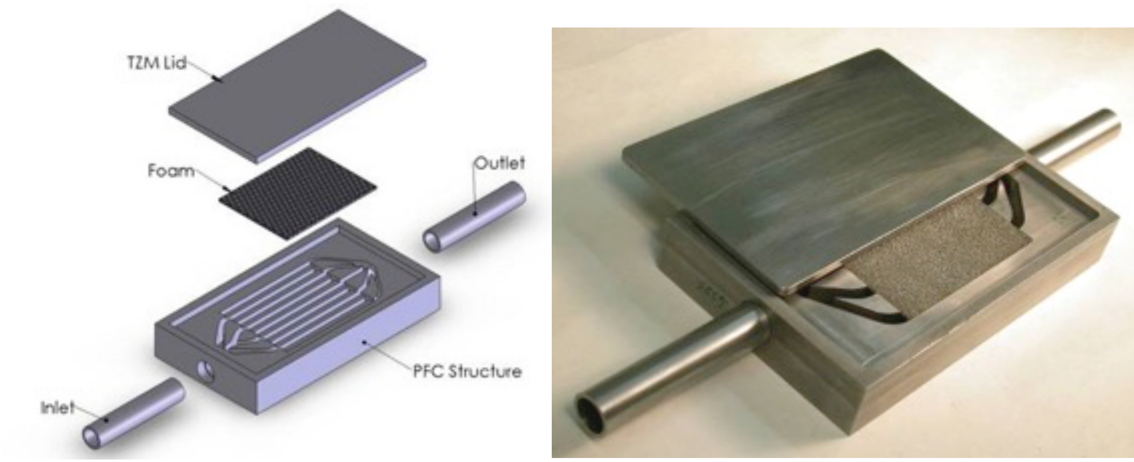

Figure II-17. Multi-channel helium-cooled TZM/Moly heat sink with moly foam: 3D CAD (left panel) and fabricated test device (right panel)[5]

A revitalized activity in PFC testing and development should consider the creation of monolithic PFCs consisting of refractory heat sinks and refractory armor such as tungsten rods or lamellae with no joints or thermal expansion mismatches. Another area relevant to PFCs is that where refractory armor or structure joins to reduced activation ferritic martensitic steel (RAFM) heat sinks or manifolding. These transitions would be needed for low heat flux applications, or the inevitable transition to standard piping in the fusion core. The Japanese recently demonstrated diffusion bonding of tungsten to RAFM steel for high temperature blanket applications. E-beam welding and explosion bonding were also pursued.

Since ReNeW, small business with the assistance of the DOE labs was instrumental in the development of advanced high-temperature refractory metal helium-helium heat exchangers for use as recuperators or closed-loop regenerators. These Brayton cycle power conversion devices used refractory foam porous media and were designed and fabricated to operate at 1000^{o} C. In addition, a related helium-lithium heat exchanger was designed and fabricated for liquid-metal power conversion. None of these devices were fully tested.

Originally, many possible plasma-facing materials were considered for ITER's divertor. But for a variety of reasons ITER selected a tungsten armor monoblock on a water-cooled copper-chromium-zirconium heat sink[13]. With a lack of alternatives, tungsten has also become the leading PFC material candidate for future fusion facilities such as FNSF, DEMO and power plants. Despite a rather sparse database on the performance of tungsten in the appropriate regimes for fusion, new designs and detailed studies followed. Significant efforts are under way both inside[14,15] and outside[16] the United States to analyze tungsten's microstructure and mechanical properties before and after neutron irradiation These efforts have identified serious limitations of unalloyed tungsten. Creative solutions beyond the use of pure tungsten are essential to the success of future fusion facilities. Although tungsten has been chosen as the main plasma-facing material (armor only, not as load bearing structural material) in ITER, and is the leading candidate for future fusion reactors, serious doubts remain about whether unalloyed tungsten will be able to withstand the required heat flux while experiencing temperature gradients and property changes produced by both neutrons and plasma ions. Current alloys are unacceptable as a structural material, and efforts to develop new hybrid materials (e.g. foil composites, alloys, dispersion strengthened, fiber composites) for PFCs must be undertaken and guided by performance results from neutron, plasma and high heat flux experimental facilities. The production of reliable high-performance heat sinks for FNSF and DEMO PFCs will require further refractory materials development, innovative fabrication techniques and clever thermal engineering, in conjunction with power-relevant testing.

A new area has emerged in which metallic materials are laid down by powder-sinter metallurgy. (For a history of this process, often called advanced manufacturing, see http://www.me.utexas.edu/news/2012/0712_sls_history.php). This new area provides tremendous advantages over older techniques by allowing an entire component to be manufactured by instruction to provide specific material properties — a capability ideally suited to plasma-facing components with an extreme set of multiple constraints and loading conditions. Advanced manufacturing techniques, such as spark plasma sintering, enable controlled compositional grading between dissimilar materials, thereby eliminating joints.

Except for ITER first wall qualification, most of the testing of components subjected to both high steady state and transient heat loads has occurred outside the United States since ReNeW. These tests include the plasma gun experiments[17], initially used to examine the impact of ELMs, and more recently e-beam experiments[18]. Reproducing the correct loading conditions in the simultaneous multi-feature loading environment is very difficult. Experimental studies like these are critical to understanding the evolution of materials under high heat loading and conditions that approach the prototypical operating conditions of an FNSF or DEMO.

Since the ReNeW report, advanced magnetic divertor configurations, such as the snowflake, X-divertor, or super-X divertor, have been proposed and may contribute to reducing heat fluxes on the divertor in future devices. These approaches are being examined in present tokamak facilities[19-22]. Simulations of ITER-like divertor geometries reaching ~ 70 percent radiated power[23] and highly radiative divertors (>95 percent, detached)[24] are likely to play a significant role in divertor power handling as well, and require significantly more experimental validation. These potential developments are covered in more detail in Thrust 9 in the ReNeW report.

Tritium management will become a major factor in future fusion facilities such as FNSF, DEMO and power plants. There is a limited worldwide supply of available tritium, with an exorbitant cost per kg. To ensure sufficient tritium to support fusion power, future devices are being designed to breed their own tritium. Design calculations show that it should be possible to breed sufficient tritium in the blankets, but any breeding inefficiencies or unexpected retention of tritium in the PFCs will challenge tritium self-sufficiency, making it imperative to track tritium through the entire fuel-cycle system. Tritium accountability is already significant in ITER, even with a low-permeation tungsten divertor. Tritium science, including tritium behavior in solid/liquids/interfaces; tritium extraction from the breeder and liquid metal PFC; and plasma implantation, permeation and retention must all be quantitatively understood to minimize losses and account for the movement and inventory of the radioactive isotope of hydrogen. Since ReNeW, the U.S.-Japan TITAN program[25,26] has shown that neutron- and ion-irradiated samples, demonstrated in Fig. II-18, have significantly different hydrogen trapping features, and both differ from hydrogen retention in non-irradiated material. The complex behavior of trapping and migration of tritium in PFCs exposed to plasma and neutrons is not well understood and requires both material characterization and tritium exposure to reveal the underlying physics. In lithium liquid-metal PFCs, the formation of helium bubbles (from neutron–lithium reactions) and their impact on the trapping and transporting of tritium needs to be better understood.

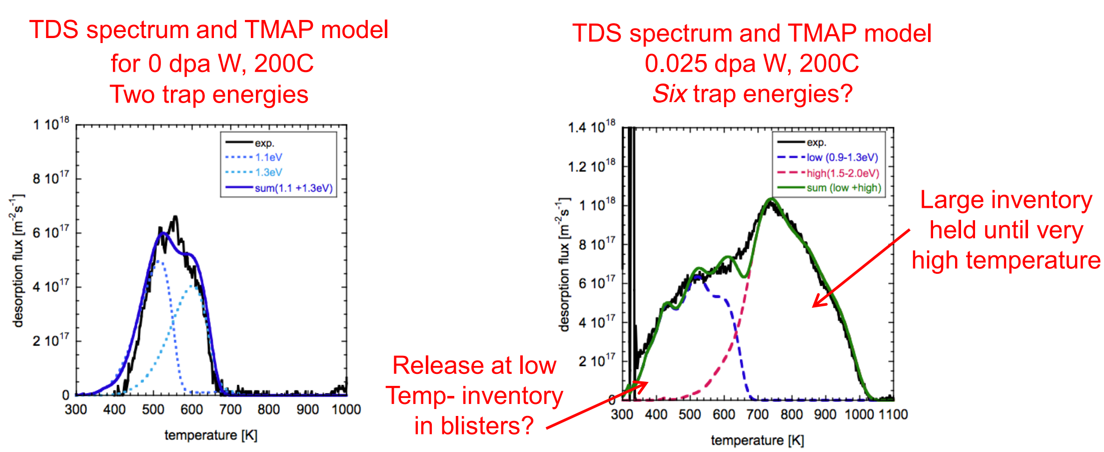

Figure II-18: Thermal desorption experiments showing the deuterium release from a non-irradiated sample and a sample irradiated in the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory, both of which were then tested in a linear plasma device (TPE at INL). A strong shift of tritium release to higher temperatures can be seen for the irradiated sample, indicating the presence of new stronger (i.e. higher energy required to release) traps for the hydrogen.

To meet power exhaust challenges for future devices, R&D on innovative power handling concepts is needed. This R&D should include evaluation of a variety of heat transfer enhancement techniques in the heatsink, such as jets or porous media, as well as internal surface roughening, swirl tubes and 3-D fins to enhance heat transfer to the coolant. This should also include application of sacrificial low-Z coatings on PFC armor for vapor shielding, the development of ductile refractory coatings for repair and management of plasma erosion, and structural materials more compatible with the plasma facing material from a thermal expansion perspective. One must also include extensive thermo-mechanical modeling as a partner to all experimental studies. Impactful high heat flux testing should allow large panel test pieces, helium-cooled heat-sinks with integrated manifolding and diagnostics, and the inclusion of a complete helium heat-transport system. This will enable researchers to carefully evaluate the influence of mass flow rate, pumping power, operating system pressure, residence time, and flow instabilities due to non-uniform heating, on the thermal performance and reliability of the components. In addition, single- and multiple-effect tests, and ultimately fully integrated test results, can validate the thermal modeling and identify failure mechanisms.

Around the time of ReNeW and up to the present, it is recognized that refractories and other solid materials for PFC applications require substantial R&D for use in a fusion reactor. As an alternative to solid PFCs, there are candidates for liquid metal-based PFCs, including gallium, tin, lithium, and tin-lithium eutectics. Among these, lithium and probably the tin-lithium eutectic could provide a low recycling surface as discussed under Thrust 12 in the ReNeW report, while other liquid metals are high recycling. A flowing liquid metal PFC would have limited residence time (at most tens of seconds, while for fast-flow systems it could be as low as 100 ms) in a fusion reactor, before removal and recirculation. Hence erosion, helium and neutron damage, and tritium retention are not significant issues (provided that low recycling liquid metals, such as lithium, can be adequately purged of tritium). PMI issues (sputtering, evaporation) would be limited to the liquid metal PFC, whereas the solid substrate supporting the liquid experiences the fusion neutron damage. The ability to separate the PMI and neutron damage can potentially simplify material qualification for reactors. The possibility of using thin layers of liquid permits intensely cooled systems, with the plasma-exposed surface closely coupled to the substrate solid and the underlying coolant (e.g. helium).

However, liquid metal PFC development is in an early stage, and many issues remain to be explored in sufficient detail to identify their viability and to pursue integrated plasma facing components. Prominent issues for both high and low recycling liquid metals include the entire problem of introducing the liquid metal to, and removing it from, the reactor, and inducing acceptable flow to transport the fluid from inlet to outlet, without adversely affecting plasma operations. MHD effects caused by the excitation of electrical currents in the liquid metal PFC must not cause macroscopic influx of the liquid metal into the plasma, or slow or thicken flows such that their surfaces overheat. Sputtering and evaporation must be kept to acceptable levels including temperature-enhanced erosion, and this dictates the temperature limit for the coolant. Heat removal must be effective below these temperature limits and be compatible with the power conversion system. Coverage of the underlying substrate by the liquid metal, in the case of slow flow, must be complete and not subject to dry-out, since the substrate will not be designed for exposure to plasma. For jets or droplet arrays, or open-surface channel flow, splashing and surface variations must be eliminated. For capillary systems, clogging and non-uniform or overly thick coverage must be avoided. The design of inlet manifolds and fluid collection systems is a challenge for either type of system. Tritium inventory, extraction, and migration through the liquid metal into underlying coolant channels must be investigated; since different liquid metals have differing affinities for hydrogen, this work is specific to each candidate liquid metal and eutectic. Finally, for lithium, the physics consequences of low recycling walls for tokamak equilibria must be thoroughly explored, since the implications, both positive and negative, for reactor design can be considerable. This last issue explicitly links liquid metal PMI and the fusion core, and has been the principal area in which advances have been made in the last 5 years.

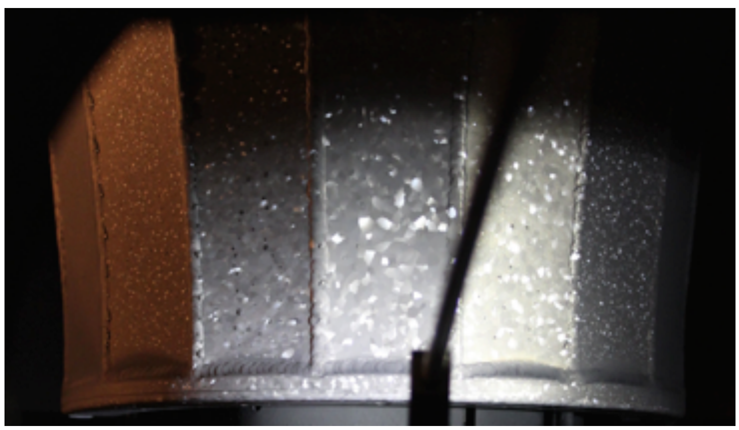

Figure II-19: View of the high field-side lithium-coated wall in LTX, showing the "spangle" pattern that forms when the liquid lithium film PFC is allowed to cool and solidify

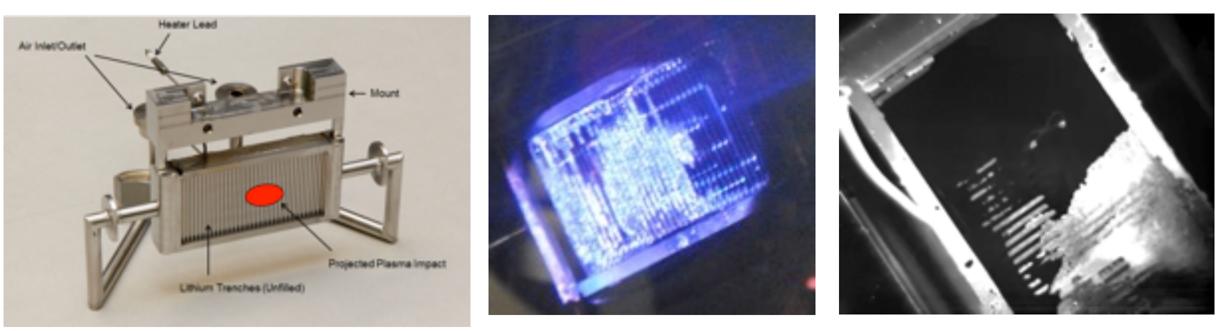

Figure II-20: The thermoelectric-MHD Liquid Metal Infused Trenches (LiMIT) system tested on Magnum PSI in the Netherlands. The middle frame shows flowing molten lithium in visible light during a shot. On the right is an IR camera view.

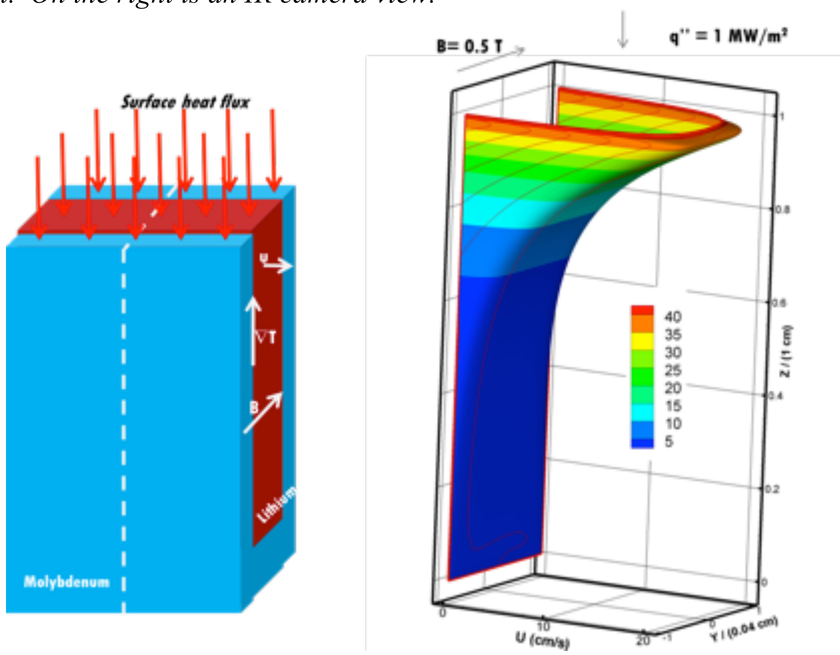

Figure II-21: Simulation of self-consistent thermoelectric MHD driven lithium flow in a trench showing surface velocities approaching 20 cm/s in a 0.5 T toroidal field subject to 1 MW/m^2 surface heat flux

Since ReNeW, there has been a significant increase in the number of experiments employing lithium as a coating for PFCs. A number of tokamaks have reported enhanced confinement or reduced edge transients (ELMs) with lithium coatings; these results are further discussed in Thrusts 9 and 12 in the ReNeW report. Liquid lithium PFCs have now been tested in FTU[27], NSTX[28], and HT-7[29]. A test limiter employing flowing liquid lithium films (FLiLi) has been briefly tested in EAST[30]. In these experiments, surface tension has successfully been used to restrain motion of the liquid metal during plasma transients (e.g. MHD, disruptions). No recent experiment has reported an uncontrolled influx of lithium into the confined plasma, or excessive concentrations of lithium in the plasma core. In the LTX device, tokamak discharges have been demonstrated which are completely bounded by liquid lithium surfaces (except for diagnostic penetrations and electrical breaks), which view 80 percent of the last closed flux surface of the plasma[31] (see Fig. II-19 for an image of the lithium coated walls). Very low (<0.5 percent) core concentrations of lithium are found in LTX with liquid lithium wall temperatures up to 240 °C. Energy confinement with solid and liquid lithium walls in these experiments exceeded[31,32] ITER H-mode scalings by up to a factor of 4. However, the technological development of liquid metal PFCs has lagged far behind testing in confinement devices. The primary exception is the development of flowing systems using thermoelectric MHD to drive flow[33]. Liquid Metal Infused Trenches (LiMIT) utilizes both the high fields and the heat fluxes present in plasma devices to circulate liquid lithium in open channels[34]. This was demonstrated at Illinois with an electron beam[35], at Magnum-PSI in a linear plasma test stand[36], and on a mid-sized tokamak[37]. Figure II-20 shows the gas-cooled apparatus used in Magnum-PSI operating between 1 and 3 MW/m^2. Accompanying simulations of free surface, surface tension and thermoelectric driven MHD flow[38,39], see example Fig. II-21, are essential to understand the coupled flow and heat transfer behavior to both better understand complicated experiments and extend the results beyond their limited parameters to FNSF and reactor conditions.

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Section II. 4: Compatibility of Boundary Solutions with Attractive Core Scenarios

II. 4. 1. Introduction to core-boundary integration challenge

The potential solutions for heat and particle exhaust in future tokamaks have the important constraint that they must be compatible with the operation of the core plasma needed for efficient production of fusion energy. Required core conditions include high levels of stored energy at the appropriate density and temperature for fusion production, and the sustainment of the plasma through efficient current drive techniques.

The challenge of heat and particle flux control coupled to high performance core plasmas often leads to conflicting requirements where the core and edge plasmas meet. Dissipation of the exhaust power to material surfaces would be best accomplished by cold dense plasma with copious radiation. Yet, the core plasma must be hot, at a temperature typically 150 million K, with controlled density for high confinement and fusion energy sustainment. Injected impurities are envisioned to provide additional radiative dissipation in the plasma boundary, while at the same time the impurities in the core must be extremely low to not dilute the fusion fuel, or cool the plasma below the required temperatures. To reduce erosion and maintain PFCs in the main chamber a significant spacing between the confined plasma and vessel walls is desired. However, such a gap could limit the stability of the core plasma and reduce the efficiency of plasma sustainment technologies. Transients of heat and particle flux to material surfaces must also be controlled, or eliminated. Yet some of the most promising techniques for doing this rely on reducing the edge pressure gradient that gives rise to high performance in the core plasma.

The conflicting requirements for the edge and core plasma will become an even greater challenge as fusion energy development moves from existing experiments to future large burning plasma tokamaks such as ITER, and on to DEMO. The 2009 ReNeW report recognized the challenges involved in achieving an attractive steady-state burning core plasma whose exhaust power is handled in a sustainable manner. Key gaps were identified between our present experience and conditions for a DEMO, including:

- *Power density a factor of four or more greater than in ITER*
- *Continuous operation resulting in energy and particle throughput 100-200 times larger than ITER*
- *Elevated surface operating temperature for efficient electricity production*
- *Tritium fuel cycle control for safety and breeding requirements, which implies retention orders of magnitude below that in present experiments*

- *Steady-state plasma sustainment and control, extending the relatively short (several second) durations of inductive operation, to non-inductive continuous operation*

These gaps and the complexity of the integrated physical processes in the boundary plasma make it difficult to extrapolate with confidence from existing experiments to future burning plasma devices. If left unresolved these issues of core-boundary integration have the potential to lead to serious performance degradation in future burning plasma devices, including ITER. The critical role the boundary plasma plays in providing conditions for the burning core plasma while simultaneously handling its exhaust motivates an expanded research program to understand and integrate the physics of the boundary plasma and to explore innovative solutions to this challenge.

II. 4. 2. Research progress since the 2009 ReNeW report and current status:

The interface between the edge plasma and the confined central plasma takes place at the magnetic separatrix, defining a transition from open to closed magnetic field lines. In high confinement plasmas (H-mode) a transport barrier spontaneously develops in this region resulting in a narrow layer of steep gradients in density, temperature and pressure[1]. The top of the H-mode transport barrier marks the inner surface of the edge plasma and serves as the boundary condition for the central plasma. The plasma parameters at the top of the pedestal directly impact the operation of the core plasma, with the pedestal top pressure largely determining its ultimate performance and fusion gain[2].

The recently developed EPED model[3] successfully describes pedestal pressure by combining a local pressure gradient limit for short wavelength MHD stability with longer wavelength MHD stability of the entire pedestal pressure profile. This model has successfully reproduced, within ~20 percent, the pedestal top pressure across a range of devices and conditions as shown in Fig. II-22. While this highly significant accomplishment now provides a basis for predicting the pedestal pressure in future burning plasma tokamaks, the EPED model requires as input all of the tokamak's operational

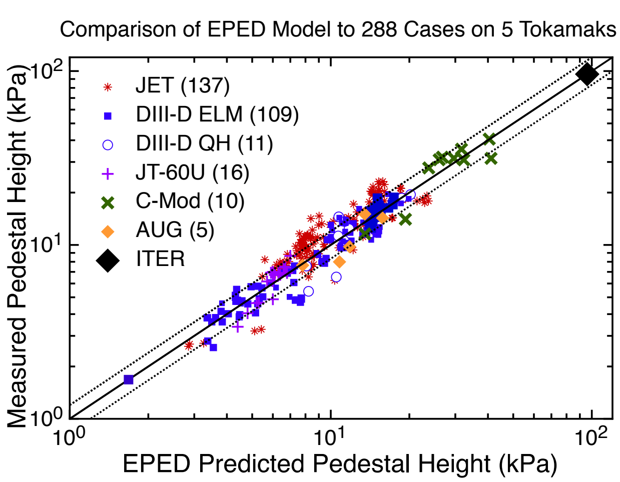

Figure II-22: Comparison of EPED model predictions of pedestal pressure height with data from a range of experiments and conditions[3] parameters, including the plasma shape, plasma current, toroidal field, total stored energy, and in particular the pedestal density. In addition, the underlying physics basis of the EPED model remains to be fully validated, both theoretically and experimentally. Physics validation of the EPED model will be an important aspect of determining its range of validity, particularly for operation with a highly dissipative divertor where pedestal degradation is often observed in existing devices.

The pedestal top density, for a given pedestal pressure, will also play a significant role in determining overall tokamak performance. To first order the pedestal density must remain below a level that would trigger a transition back to low confinement, L-mode. In addition the density should be optimized for efficient plasma current sustainment. Existing current drive technologies are generally more efficient at higher core plasma temperatures and lower densities. Fusion reactor design studies find overall fusion performance, taking into account current drive requirements, can be optimized below the maximum achievable density. The pedestal top density is expected to be largely determined by conditions in the scrape-off layer (SOL) and divertor. Requirements for divertor heat flux control will likely result in high plasma density at the separatrix and possibly a large flux of neutrals in the X-point region. Existing transport models are not yet able to predict how the pedestal density profile will respond to these divertor and SOL conditions.

The pedestal performance will also be affected by impurities that have been eroded from material surfaces or injected into the divertor in order to increase radiative dissipation of exhaust power. Excessive impurity buildup in the pedestal can result in radiative loss exceeding that required to maintain a robust pedestal. Impurities also increase collisionality in the pedestal and affect performance by suppressing the edge current and potentially degrading the pedestal transport barrier. Impurities in the pedestal are also transported to the central plasma diluting the fuel and reducing fusion power. Again existing models are not yet capable of predicting the transport of impurities from their source to the top of the pedestal.

Finally, the operational limits to robust pedestal operation must be determined. Existing models of pedestal prediction, including EPED, assume a fully developed H-mode transport barrier with operation in the Type I ELMing regime[4]. Divertor heat and particle flux control in existing experiments often result in high density and collisionality, neutral flux, or impurity density that degrades the pedestal beyond that expected from the EPED model. A model of the H-mode transport barrier that defines its limits of robust operation is clearly needed to determine the compatibility of divertor heat and particle flux control solutions with optimized pedestal operation.

Controlling divertor heat flux to a manageable level is a central role for the boundary plasma, and must be compatible with a high-performance core plasma. Since the 2009

ReNeW report, a multi machine comparison, initiated in the U.S. program, revealed an even greater than expected challenge in controlling divertor heat flux while maintaining high confinement in future devices. This study found that the SOL and divertor heat flux width, which is inversely related to the peak parallel heat flux, in present devices scale as $\sim 1/B_{pol}^{\text{outer midplane}}$, with no apparent dependence on plasma size or on power entering the SOL (Fig. II-2)[5]. This result implies a predicted heat flux width for ITER of 1 mm, similar to that measured on Alcator C-Mod, a device of similar magnetic field strength, but many times smaller. The practical implication of this scaling for ITER is a parallel heat flux of ~3-10 GW/m^2 that will require higher divertor density and neutral pressure for acceptable heat flux dissipation. This additional divertor dissipation is expected to require ITER's operational parameters near the limits of high confinement operation[6].

The realization of the increasing challenge of providing divertor heat flux control for high performance burning plasmas and the limitations already observed in providing such control in existing high power tokamaks has motivated the exploration of innovative divertor configurations that offer potential benefits significantly beyond conventional divertor geometries currently deployed.

Since the 2009 ReNeW report experimental investigation has begun on several advanced divertor configurations and significant new theoretical developments have occurred. A range of advanced magnetic geometries had been developed theoretically prior to 2009: the X-divertor, snowflake divertor and Super-X divertor, all of which had been predicted to reduce heat flux. Important new theoretical developments since 2009 include predictions of significant improvements with these geometries on heat flux dissipation[7-9] at conditions that are likely to be more compatible with high confinement H-mode operation and overall tokamak performance. These configurations, by a variety of mechanisms, aim to induce heat flux dissipation and divertor detachment at lower midplane and pedestal densities than for conventional configurations, and to isolate the radiative region and high neutral pressures in the divertor, thereby avoiding degradation of the H-mode pedestal. Innovation in magnetic divertor concepts is continuing; for example, the X-point target divertor[10] and the "double decker" divertor[11], building upon the X-Divertor and Super-X Divertor, have been proposed.

Initial experiments to examine these innovative configurations since the ReNeW report have provided promising results[12-14]. These initial investigations have successfully demonstrated the spreading of divertor heat flux over a larger area and inducing divertor detachment at lower pedestal density. Calculations indicate that it may even be possible to implement a moderate X-divertor on ITER without any modifications of the baseline hardware[15]. More details of these concepts can be found in Section II-1. Significant research remains, however, before the potential of these divertor concepts for future burning plasma tokamaks can be fully evaluated. The lack of quantitative understanding of the divertor and SOL requirements for achieving divertor detachment combined with uncertainty in the mechanisms that lead to H-mode pedestal degradation observed with divertor detachment in existing experiments makes it difficult to quantitatively use these concepts to design an optimized divertor for future burning plasma tokamaks. While research must continue on understanding these constraints, it appears that advanced divertor geometries can alter the interaction between the detachment front and the H-mode pedestal, thereby enabling more optimized operation.

The challenge of proposed burning plasma relevant PFC material choices on core plasma scenarios has been highlighted by recent results from the European tokamaks ASDEX Upgrade (AUG, Germany) and JET (UK) with high-Z metal plasma facing components. Until recently most diverted tokamaks used low-Z (carbon) PFCs or, as in the case of Alcator C-Mod with molybdenum PFCs, use a low-Z (boron) coating to achieve high confinement H-modes[16]. To test proposed reactor PFC materials, ASDEX-Upgrade has gradually replaced all PFCs with tungsten. JET has taken the approach of replacing their carbon PFCs with ITER's PFC material choice, beryllium for the main chamber and tungsten for the divertor. In both devices the transition to high-Z PFCs has restricted operational parameters, and resulted in reduced overall performance that was not predicted, nor yet fully understood. High-Z PFCs have required specialized techniques to limit the buildup of high-Z impurities in the core plasma, such as increased gas fueling and RF heating. The combination of PFC effects on impurity and recycling influx and reduced operational space has often led to reduced plasma performance[17]. Furthermore, experiments on JET[18] and on C-Mod[19] have shown that attaining high quality core confinement requires maintaining power flow through the pedestal greater than the L-to-H power threshold. Taken together, these results indicate that developing acceptable plasma material interaction with reactor relevant PFC materials simultaneously compatible with high core plasma performance will be considerably more difficult than previously expected. In fact, to date the maximum exhausted SOL power densities that have been dissipated to within steady-state limits for solid materials while still maintaining acceptable core confinement[16,20] are a factor of 2-5 times smaller than those expected for ITER and *a factor of 2-20* times smaller than those expected for DEMO[7]. Due to the strong sensitivity of fusion gain to core confinement, even relatively modest reductions (~10 percent) in confinement to produce acceptable plasma-material interactions may produce a much larger impact in overall device performance.

Investigations have also begun on Low-Z alternatives, such as liquid lithium PFCs. Liquid lithium PFCs have the favorable attributes of being continually replenished and of not being reshaped or modified by erosion/redeposition. Fast flowing lithium divertor targets also offer the potential for very high power handling. Additional advantages and constraints imposed by liquid lithium PFCs are described in Sections II-1 and II-2. An important distinction, and potential advantage, of liquid lithium PFCs compared to solid targets is that they can have very low fraction of particles leaving the plasma that re-enter as cold neutrals. Recycling coefficients of as low as 0.1 – 0.2 may be accessible. Benefits of lithium PFCs seen in present machines (NSTX-U, LTX, EAST) have included increases in energy confinement, low lithium (and other impurity) accumulation in the core, and elimination of large ELMs[21]. Lithium coatings and lithium edge injection have also been observed to increase the edge pedestal width and height[22]. The low recycling and edge fueling would also imply the need for additional central fueling if liquid lithium PFCs are deployed in future burning plasma tokamaks. These experimental observations and issues demonstrate again the need for reliable, validated models of pedestal transport.

Interactions of the plasma boundary with the core plasma include not only the divertor and main limiting surfaces, whose primary function is to handle plasma heat and particle fluxes, but also the in-vessel components of actuators needed to heat, control and sustain the fusion plasma. While a significant concern for all PFC components, the challenges for actuating RF heating and current drive are great. The 2009 ReNeW report highlighted important areas of research for RF antennas and launchers. These included:

- *How can the predictive capability of plasma edge models, including material interaction, be enhanced?*
- *Can these enhanced models incorporate the formation of radiofrequency sheaths produced by radiofrequency waves transiting between the radiofrequency antennas and absorption in the core plasma?*
- *Can innovative concepts be developed that move sensitive front-end components far from the plasma edge?*

Good progress has been made in each of these questions since ReNeW, clarifying the issues for core-edge integration, and in some cases offering prospects for solutions. RF codes have been extended to calculate wave propagation and absorption in the SOL plasma. These have highlighted the importance of the SOL in wave propagation and eventual absorption for both lower hybrid (LH) and ICRF, and the need for detailed measurements of the SOL characteristics in order to predict the location of possible RF interactions with the plasma-facing materials. Progress has been made in formulating the attributes of sheaths generated by the RF waves and incorporating them in RF codes as an appropriate boundary condition.

Several new concepts have been put forward to minimize the interaction between the plasma and launcher components. Installing RF antennas and launch components on the inner wall takes advantage of the very low density, quiescent high-field-side SOL that is obtained in double null operation[23,24]. For ICRF the quiescent plasma of the high-field-side would permit the antenna to be closer to the separatrix for better coupling with reduced plasma interaction. For high-field-side LH launchers, in addition to the reduced plasma interaction, the production of fast electrons by the launcher may be ameliorated since they would drift away from the launcher. An alternative concept, the helicon, allows for a travelling wave launcher with light coupling to the plasma, allowing for positioning in the far SOL. Success for this concept will again require detailed understanding of wave propagation in the SOL.

Field aligned antennas take advantage of the symmetry of the plasma dielectric to minimize undesired RF field interaction. It was found on Alcator C-Mod that impurities originating directly from the antenna were essentially eliminated for a field-aligned antenna. However, both Alcator and NSTX experiments have highlighted the importance of RF generated impurities *away* from the launcher structure generated by RF fields or potential changes in the scrape-off plasma. Understanding and control of these effects remain a challenge.

The injection of RF may also be used to directly affect the pedestal and its performance. Experiments on Alcator[25] and EAST[26] have shown a strong modification of the pedestal and changes to ELMs during injection of LH waves. The exact mechanisms are unknown and the application to transient control needs to be further explored.

Optimizations and predictions of overall tokamak performance in future burning plasma tokamaks assume operation with a robust pedestal in the Type I ELMing regime. However, scalings of repetitive heat pulses from Type I ELMs and tests of PFCs under pulsed heat loads[4] have made clear that ELMs must be either drastically reduced in size or completely suppressed or naturally avoided. Recognizing the importance of this issue, a separate FES Workshop has been devoted to Transients. The U.S. has been a leader in this area with a number of potential solutions being advanced as detailed in the Transients Workshop report. Active mitigation or suppression has been achieved by several methods, including resonant magnetic perturbations from in-vessel coils, and the rapid injection of fuel pellets. These techniques aim to increase transport through the pedestal before the pressure in the pedestal reaches an MHD limit. Significant advances have also been made in developing regimes that are naturally free of ELMs. The Quiescent H-mode, developed at DIII-D, has been extended to more ITER-relevant conditions. A new and quite different regime without ELMs, the I-mode, pioneered at Alcator C-Mod and recently extended to other tokamaks, features a thermal transport barrier without a particle barrier. The injection of lithium in NSTX has also led to high performance ELM-free operation. As with the active techniques, these natural ELM-free regimes provide additional transport, particularly particle transport, through the pedestal.

It is increasingly clear that all of the ELM control regimes or techniques have effects on pedestal and divertor physics beyond that of the Type I ELMy H-mode. All of these techniques, to a greater or lesser extent, result in increased particle transport through the pedestal. This transport may potentially decrease the negative impact of impurities, either seeded or generated from PFC materials. On the other hand, the additional transport leads to reduced density and may increase the challenge of developing divertor solutions. In addition, the non-axisymmetric fields from ELM control internal coils significantly modify the divertor heat flux profile that must be accommodated. None of these ELM control methods has yet been demonstrated with dissipative detached divertor operation. Thus, while developing techniques to avoid ELMs is the major focus of the Transients Workshop, research into PMI and integrated solutions must take into account these differences. Similarly, any proposed solutions must be compatible with low probability of disruptions, whether by MHD instabilities or potentially induced by material influxes.

II. 4. 3. Summary of priority core-boundary integration challenges

The above summary of research progress and status makes clear that the challenges for development of integrated boundary solutions compatible with attractive core scenarios remain significant and complex. Large effects, both positive and negative, from changes in boundary materials, geometry and actuators have been demonstrated. However, it is not yet possible to understand and predict these effects sufficiently to project with confidence to future fusion devices. Based on importance and impact of these effects, and on readiness to make substantial progress in the next several years, we recommend the following Basic Research Needs as focus areas. The scientific questions, and proposed research plans to address them, are described in much more detail in the Priority Research Directions chapters. Some of the relevant and needed accompanying research is described in other 2015 FES Workshop reports.

1) **Improve understanding and prediction of pedestal transport and the influence of conditions at the plasma boundary to allow prediction and optimization of core-boundary solutions in future devices.**

As discussed above, the influence of plasma-facing boundary solutions occurs primarily through their effect on the transport barrier region just inside the last closed flux surface. The parameters at the top of this barrier, or pedestal, in turn provide critical boundary conditions to the performance of the hot fusion plasma. While great progress has been made in predicting the limits of pressure at the pedestal top, set by magnetohydrodynamic (MHD) stability, present models cannot separately predict the profiles of electron and ion temperature, and also main species or impurity density. These depend on local fueling and impurity sources and on transport, including turbulence and neoclassical effects. Model predictions for the pedestal also are strictly valid only for the Type I ELMs regime, whereas these types of ELMs must be reduced in amplitude or avoided altogether in burning plasmas. Such scenarios will significantly change the pedestal parameters and transport.

The improved divertor and SOL physics described in Section II-1, and the R&D resulting from PRD B and C, should result in accurate predictions of the density, temperature and particle fluxes at the separatrix. Corresponding predictive understanding of the pedestal region will enable us to understand the impacts of these conditions on the core plasma, and assess the tradeoffs inherent in developing integrated core-boundary solutions. This together with the efforts described in the Integrated Simulation Workshop report should allow us to optimize attractive, steady-state fusion scenarios. The scientific issues and research to address them are described in Chapter VII, under **PRD E: "Understand the mechanisms by which boundary solutions and plasma facing materials influence pedestal and core performance, and explore routes to maximize fusion performance".**

2) Determine the effects of candidate plasma facing materials at the divertor and main chamber on integrated core scenarios.

Materials at plasma-facing surfaces set boundary conditions for the edge plasma that can affect fusion plasmas via many different mechanisms, including dilution, changes in Z_{eff}, radiated power and changes in fueling from recycling. Some of these effects will be understood via the pedestal transport studies discussed above. However, there is a complex interplay between the local sources and penetration of impurities to the main plasma, which will depend on the heat and particle fluxes, material locations, and magnetic configuration. Ultimately, direct experimental comparisons of candidate materials, both high-Z and low-Z, will be required to assess and compare the impacts, in conditions as close as possible to those expected in a reactor, and to test our improving predictive capability. Experimental experience with liquid lithium in high confinement, diverted plasmas is particularly thin, and neither high-Z nor lithium have yet been used with advanced divertor configurations. Effects of the elevated PFC temperatures required for reactors have not been assessed for any material. Aspects of this research are part of **PRD B: "Understand, develop and demonstrate innovative dissipative /detached divertor solutions for power exhaust and particle control, sufficient for extrapolation to steady-state reactor conditions**" (Chapter IV), and in **PRD C:** "**Understand, develop and demonstrate innovative boundary plasma solutions for main chamber wall components, including actuators for sustainment and control, sufficient for extrapolation to steady-state reactor application**" (Chapter V). .

3) Develop actuators for sustainment and control compatible with boundary and core scenarios

A number of challenges remain in the understanding and demonstration of external actuators that can meet control and sustainment requirements for a fusion reactor. The challenges are both in the plasma-actuator interaction and maintaining compatibility with core scenarios. Some of these challenges are general, e.g. high current drive efficiency at lower density, and compatibility of in-vessel components with local heat fluxes and radiation fluence. Other challenges are specific to each actuator technique. For ICRF, RF-enhanced impurity production remains a challenge. For example, the JET ITER-like wall results indicate that while electron heating via ICRF can flush tungsten impurities from the core, the RF can also enhance tungsten influx from near or in the divertor. While the field aligned ICRF antenna eliminates impurities generated directly from the antenna, it does not affect the RF-driven potential in the SOL and the enhanced flux of impurities away from the antenna. Thus measurements of impurity sources and transport at multiple locations will be required to optimize this technology. The predicted benefits of high-field-side launch, for both ICRH and Lower Hybrid Current Drive (LHCD) remain to be tested experimentally. LHCD can modify the pedestal and ELMs, via mechanisms that are yet to be understood but may provide beneficial control tools. For Electron Cyclotron Heating (ECH), the challenge is control of the localized absorption region without a highly reflective mirror close to the plasma. Again solutions have been proposed but have not yet been implemented on experiments. Challenges for neutral beam injection (NBI) include the large ducts at the low field side, which pose serious design difficulties in a reactor, together with high beam energies required for fusion-scale devices and difficulty running beams in steady state. Since the actuators for heating and current drive, and most control tools such as coils, will be located in the main chamber and have strong dependences on SOL plasmas, this research is included as part of **PRD C: "Understand, develop and demonstrate innovative boundary plasma solutions for main chamber wall components, including actuators for sustainment and control, sufficient for extrapolation to steady-state reactor application**" (Chapter V).

4) **Assess compatibility of boundary and core scenarios with transient control solutions.**

It will be essential to operate future devices in regimes without large ELMs, whether controlled or suppressed by external means, or in regimes which naturally avoid ELMs. As discussed above, while several techniques and regimes exist (e.g. RMP H-modes, pellet pacing, QH-mode, I-mode), each of them has distinct differences in pedestal and SOL physics which will need to be considered in understanding and optimizing core-boundary solutions. Some of these may prove beneficial. For example, naturally high particle and impurity transport may reduce the impact of high-Z impurities. Others may prove challenging, such as compatibility with high density detached divertors, or potential changes in heat flux locations due to external or natural instabilities. While development and investigation of ELM control techniques is covered under the research plan of the Transients Workshop report, it will be important that the research outlined in this report not be restricted to ELMy H-modes; it must include and even focus on conditions not dominated by large ELMs. This includes in particular **PRD E: "Understand the mechanisms by which boundary solutions and plasma facing materials influence pedestal and core performance, and explore routes to maximize fusion performance."** Also boundary research in PRD B and PRD C should consider solutions compatible with ELM control techniques and regimes to be applied in future burning plasmas.

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Chapter III

Priority Research Direction 'A' – PMI Material Limits

III. Priority Research Direction 'A' – PMI Material Limits

PRD-A: Identify the present limits on power and particle handling, as well as tritium control, for solid and liquid plasma facing components, and extend performance to reactor-relevant conditions with new transformative solutions

Energetic plasma particles impinge on surrounding surfaces in both a steady-state and time-dependent manner. The capability of materials, and components made of these materials, to withstand both the steady and transient heat and particle fluxes is a critical aspect feature for plasma facing components. Also tritium fuel, a precious and scarce commodity needed for fusion, must be tracked carefully within reactors. The behavior of tritium originating from both the plasma and the breeding blanket must be understood and predictable for the required high-precision accounting. Due to this uniquely harsh environment, both solid and liquid materials should be considered as PFCs. In reactors these components must last for a few to several years, for technical and economic viability. Breakthroughs in solid materials development (e.g. grading, composites, fibers, nano-structuring), critical evaluation of liquid metals (e.g. lithium, lithium-tin, gallium, tin), and seminal advances in manufacturing techniques, coupled with multi-scale theoretical computations, will be used to develop integrated PFCs that simultaneously address the myriad of loading, constraints and functions.

III. 1. Additional Background and Main Scientific Questions

The interactions that occur between the plasma and plasma-facing materials provide a challenging obstacle to realizing fusion power production. Ultimately, the plasma must be operating continuously for approximately a year between routine maintenance, with sufficient performance to provide the fusion power for conversion to electricity. The plasma-facing surface is part of a larger integral component (e.g. the divertor, first wall/blanket, and plasma heating/current drive components) that is designed to remove both the plasma surface and volumetric nuclear heating, and otherwise survive the environment without adversely affecting plasma performance. Simultaneously, these components must provide other essential functions (e.g. breed sufficient tritium fuel, maintain vacuum, launch waves into the plasma, control tritium transport, provide high quality heat for conversion to electricity), over reasonable lifetimes of ~ two to five years, if possible. Functional and robust plasma facing components that meet these requirements are likely to require a combination of SOL plasma physics improvements (e.g. advanced magnetic divertors, operating regimes without large transients, optimization of RF launch location), materials development (tungsten nano-structured, tungsten fiber/foils, liquid metals), and advanced component designs (functionally graded structures, helium jet impingement cooling, optimized material-function solutions for entire plasma facing component). This priority research direction concentrates on the many aspects of engineering science development and design. The results of the other priority research directions in this document provide critical information for the integrated component development described here. The PFCs are the first wall (integrated into the breeding blanket), the divertor, and special in-vessel components such as launchers or antennas, and diagnostics that penetrate through and are supported by the first wall/blanket. The complex loading environment that these components would experience includes, 1) plasma surface heating and particle bombardment, 2) PFC surface morphology evolution through erosion, redeposition, and migration, and localized arcing, stress-induced cracking, and melting, essentially transforming/removing structurally sound material, 3) PFC material dust and debris production, 4) high electromagnetic mechanical loads from halo currents and eddy currents produced by plasma disruptions, 5) tritium implantation, and 6) fusion nuclear volumetric heating, and material damage/transmutation, producing thermo-mechanical property evolution. These conditions are combined with hydrogen and helium in the material matrix from neutron irradiation, high temperatures, high pressure/stresses, corrosion (chemical interactions), high magnetic fields, and vibration from fluid flows. An important consideration for PFCs is their integration, since they are not just surfaces, but actually volumetric structures. They contain (sacrificial) plasma facing armor, structure, and coolant, which can include multiple materials, braze fillers, diffusion barriers or coatings. In the case of the first wall, it is the first few centimeters of the much larger blanket, which is approximately 1 - 1.5 meters thick and absorbs the vast majority of the neutron heating and must breed tritium in lithium compounds. The divertor is generally a dedicated heat removal component to handle local concentrated heat loads. Wave launchers (e.g. antennas or waveguides, or mirrors) provide an efficient medium to transport waves up to the first wall region and into the plasma. Each of these components requires a comprehensive design solution that incorporates all constraints on their operation.

Research is needed to identify viable materials and designs for these components in environments that are as prototypical as possible for a fusion DEMO or commercial power plant. This research will begin with smaller scale facilities, concentrating on basic and separate effects, i.e. individual materials and take place in accessible and not necessarily prototypical environments The research can then expand to multiple-effects/multiple interactions that include a higher level of integration (toward a component), and higher levels of prototypical parameters (both in magnitude and multiplicity). The very long plasma durations, of about one year, provide a tremendous new constraint for PFCs that present confinement devices, and even ITER, obviate due to very low plasma availability. The long plasma duration issues include plasma erosion, re-deposition and migration, higher neutron wall loading measured in displacement per atom (DPA), transmutation and helium production, tritium retention, dust and debris production, and plasma-facing surface evolution. If transients, such as ELMs, are anticipated, these long pulse lengths will lead to very large numbers of thermal cycles, and potential fatigue failure. Reliablility, availability, maintainability and inspectability (RAMI) provide critical constraints for the very long-pulse operation anticipated in next step facilities.

PFCs made of refractory metals (tungsten, molybdenum, tantalum, etc.) are considered the most favorable for their high melting temperatures, high thermal conductivity, weak changes in thermal properties under neutron irradiation, higher resistance to neutron damage (low DPA/MW-yr/m^2) and low sputtering yields and/or high sputter energy thresholds. On the other hand, these metals can provide a source for high-Z impurities in the core plasma that affect its performance and, if they melt from an excessive heat load during transients can provide a continuous leading edge. The metals are also brittle and can crack under loading and/or have limited temperature windows for avoiding significant reductions in ductility. The properties and behavior of these refractory metal components in the appropriate fusion environment are not well characterized, and experiments are required to address the following issues:

- Non-nuclear properties characterization
- Sputtering, erosion, re-deposition
- High temperature properties, operating windows
- Non-structural armor, allowed operational parameters
- Structural (brittle material design prescription)
- Manufacturing with the metal, and components from the material
- Transient loading (ELMs, disruptions, heating and particles) response
- Gas cooling, viability and thermal design (CFD, TM)
- Impurity production, as well as dust and debris production
- Tritium retention in PFCs, dust, and debris
- Ferromagnetic impacts with reduced activation ferritic martensitic steel
- Neutron irradiation property characterization (including damage, activation and transmutation)
- Remote handling and radioactive waste disposal

Liquid metal (LM) PFCs have the potential to alleviate some of the difficult constraints that solid PFCs face. LMs include lithium, gallium, tin, and lithium-tin eutectics. The materials' function as a PFC would not be affected by erosion or material modification from neutrons or plasma particles. The solid substrate that supports these liquids would only see neutron irradiation and is protected from the plasma. Lithium or lithium eutectics would have a high affinity for tritium and deuterium at low operating temperatures, and would provide a low recycling environment for the core plasma that seems to provide significant confinement improvements in existing plasma experiments. The other liquid metals would provide high recycling surfaces. Liquid metals have sufficient thermal conductivity for good thermal conduction and can provide vapor shielding under transient heat loads. On the other hand, electrically conducting liquids will have MHD interactions that can disturb the surface allowing material to enter the plasma, and can laminarize the flow, thereby reducing turbulence and convective heat transport. Sputtering and evaporation must be controlled by limiting the operating temperature. The substrate material must remain covered to survive, and the design for how the liquid metal enters and exits the plasma chamber or divertor region needs to be identified. The properties and behavior of these liquid metal components in the appropriate fusion environment are not well characterized, and experiments are required to address:

- Temperature windows to control evaporation and sputtering to required levels
- High heat flux handling with a flowing LM system, with self-consistent vapor shielding and MHD effects on free surface

- Design and testing of an integrated component including substrate, coolant, and flowing LM plasma facing material
- MHD effects on flowing LM free surface and heat transfer
- Compatibility and wetting of LM and substrate materials (including chemical reactivity, corrosion and embrittlement)
- Tritium (hydrogen) behavior and removal from LM
- Entry and exit design for LM into plasma chamber and divertor, MHD pressure drop and flow control
- Response of LM/substrate to transient loading (ELMs) and off-normal loading (disruptions)

There are numerous demonstrations of these solid and liquid PFCs handling high heat loads, with solids generally more mature in their development and applications than liquids; however, both have significant uncertainty in their performance (either as a material or as a component) over long periods in a fusion prototypical environment.

Tritium is a fuel in the deuterium-tritium (DT) fusion reaction and must be supplied to the plasma chamber via pellets or gas injection. In general, the tritium burnup (amount of injected tritium actually consumed in fusion reactions) is not well predicted, but is expected to be 5-15 percent or less. This implies large throughput of tritium from injection to exhaust. While in the plasma chamber, the tritium can be implanted in the PFCs if at higher energy, or recycled from a solid PFC or non-lithium liquid PFC when at low energy. For a liquid lithium PFC at modest temperatures (<400 C), the tritium that comes in contact is expected to be bound regardless of its energy. Implanted tritium can diffuse through a solid PFC to the coolant, or through the liquid metal surface to the substrate's coolant. The tritium may become attached to dust or other debris, or trapped in the re-deposited surface of solid PFCs. Simultaneously, tritium is being bred in lithium compounds in the blanket, and this tritium can diffuse into surrounding structures, ultimately reaching the PFCs. The neutron irradiation of all solid materials will generate damage in various forms that can serve as traps for tritium and deuterium in their matrix. This increases the trapping of tritium that is implanted into or diffuses from the coolant into PFC structures. Very precise accounting of the tritium is necessary to predict inventory and avoid losses and leakage into the environment, which requires high fidelity physics modeling, validated against representative experiments, of the processes under the environmental conditions.

Among a number of heating and current drive schemes, RF launchers are among the most promising actuators. For LH and ICRF power, the launching structure needs to be next to the plasma. Thus, the launching structures are subject to the same severe exposure conditions as the first wall with additional requirements. For LHRF, the present vision is for a launcher array of reduced height waveguides with additional protection limiters. By contrast, an ICRF launcher is a set of inductive straps housed in a protective cavity open to the plasma. An ICRF antenna will also require a Faraday shield to inhibit electrostatic coupling; the Faraday shield constitutes a prominent plasma facing system. In addition to the first wall material requirements, the materials that carry RF currents require relatively high conductivity and good surface quality. Poor surface quality could result in excessive losses, slow conditioning and reduced power density limits. Contacts between structural elements that carry RF current need to have well-defined surfaces with low contact resistance. Active cooling of inductive straps, Faraday shields, and waveguides will be required. In addition to standard plasma material interaction, the RF launchers need to be able to withstand RF-enhanced plasma material interactions due to strong, local RF fields. For example, convective cells and RF-enhanced sheaths result in greater plasma flux onto the launcher, and increased ion sputtering energy. To transfer the RF to the plasma, an insulating vacuum window is required and should have minimal RF losses with good thermal conductivity. For maximum lifetime, the window should be located where the neutron flux-induced damage can be minimized.

Electron cyclotron radio frequency (ECRF) heating and current drive is also a potential candidate for fusion. Although it does not require coupling to the plasma, as do LHRF and ICRF, its waves (beam) do need to be steered and brought to the first wall to enter the plasma. In order to take full advantage of EC, which has flexible deposition in the plasma, a means for directing the waves is needed. This is performed on present devices, and also on ITER, by mirrors located both well behind the first wall, and very close to the first wall. If mirrors are pursued they must maintain their reflectivity and controllability of direction. Other strategies can be considered as pre-directed waveguides for example. The waveguides, which penetrate the blanket and other structures, must maintain their characteristics to guarantee that the proper mode is launched into the plasma. Like LHRF and ICRF, a vacuum window is required and must be located where its damage can be minimized.

There are several external constraints on the PFCs as fusion moves into the burning plasma and fusion energy regime. Low activation materials are preferred in order to reduce the toxicity and lifetime of radioactive waste and the decay heat produced. This allows for lower waste ratings, which facilitate simpler disposal. In general, water is not a viable coolant in a fusion power plant due to its high pressures, strong reactions with lithium materials, difficulty in tritium removal, and incompatibility with neutron-resistant structural material's operating temperature ranges. Water also restricts the operating temperatures limiting the thermal conversion efficiency. The blanket surrounding the plasma must breed tritium efficiently through neutron reactions with lithium nuclei. Any excess structural material provides a significant parasitic absorption medium and must be minimized, implying that large elaborate structure concepts on the first wall are not viable as fusion energy-relevant PFCs, while they are tolerable for a divertor component. The divertor receives approximately 20 percent of the power available for thermal conversion to electricity, and it is desirable to recover this power rather than reject it. This implies that low temperature or inefficient operation of the divertor is unacceptable.

At present, neither functional materials, nor the requisite computational tools, nor the underlying knowledge base currently exist for reliable assessments of integrity and lifetime of fusion plasma facing components (or in-vessel structures as a whole). Current cooling designs are often based on thermal-hydraulic correlations that require fully developed flow conditions. Such conditions seldom exist for complex 3D components with relatively short flow paths with many bends and manifolds. RF-sheath interactions require more sophisticated multi-physics computational tools that are only now becoming available. New design and in-service performance computational tools must be developed to replace simplistic high temperature design and operational rules. These tools must ultimately be incorporated in design codes and regulatory requirements.

The greatest challenge is our lack of understanding of several aspects of material behavior. Specific examples of this limited understanding include

- Failure mechanisms in tungsten alloys
- Radiation damage effects on mechanical properties in the presence of fusion-relevant helium concentrations
- The evolving surface morphology of plasma facing structures
- Synergistic effects of radiation and thermomechanical damage in first wall and divertor components
- Synergistic plasma heating and MHD effects on free surface liquid metal flow, surface deformation and heat transfer
- Models of ferromagnetic materials (reduced activation ferritic martensitic steels), especially in the presence of transient magnetic fields.

In addition to these deficiencies, we have only limited understanding of macroscopic failure mechanisms, especially in the harsh environment experienced by a fusion component. For example, modeling damage due to the interaction of creep and fatigue is already a difficult challenge; adding radiation damage, helium, etc. increases the uncertainty dramatically.

While some progress on enhanced understanding of these phenomena using exposed sample tests is possible, we cannot properly address failure mechanisms without comprehensive structural models that include coolant pressure, coolant chemistry, static thermal gradients, thermal transients, and radiation damage. In addition, all of this work needs to consider modern fabrication techniques and the use of engineered materials, which may be necessary for the success of fusion and certainly impact materials and component behavior. Thus, a multi-disciplinary, multi-scale effort is needed to comprehensively address the materials-design interface and permit substantial progress towards the design of high performance, optimized components. This effort is aimed at answering the following questions:

- What are the maximum steady-state heat fluxes and temperatures for actively cooled <u>solid</u> and <u>liquid</u> components?

- What are the tolerable peak heat loads and transient durations for actively cooled <u>solid</u> and <u>liquid</u> components?

- What are the effects of tritium implantation and permeation, and tritium retention in <u>liquid</u> and <u>solid</u> PFCs? What will be the inventory and how will tritium be removed from lithium LM PFCs?

- How will the fusion neutron-induced transmutation and helium production affect the PFC's function, bulk and surface?

- What processes will limit the lifetime of PFCs? These may include fusion neutron irradiation, erosion, thermo-mechanical cycling, and plasma-facing surface morphology evolution.

- How can advanced manufacturing be utilized to extend performance and lifetime limits?

There is a corresponding set of simulation needs coupled to the questions above:

- Take advantage of multi-physics computations to address the multi-loading/multi-feature environment seen by PFCs in their service

- Develop computational tools for free-surface liquid metal analysis (MHD)

- Model tritium implantation, co-deposition, entrainment in dust and debris, and transport processes in real-time reactor systems

- Develop the multi-scale modeling of PFCs to develop advanced materials/components using advanced manufacturing that can combine the materials-design-manufacturing aspects into one

- Develop a qualification program for PFCs to provide reliable and robust components to a fusion reactor

III. 1. 1. Characterization of facilities

Test stands intended to isolate specific behavior (single or few effects) and establish basic database, and examples include

1. High heat flux facility
2. Liquid metal MHD free surface flow loops, allowing vacuum/plasma exposure, and entry and exit development for first wall and divertor LM concepts
3. Liquid metal free surface interaction experiments, allowing exposure to high heat flux and/or plasma particle flux, to allow vapor shielding and other effects to be studied
4. Hydrogen removal from LM, and substrate/LM corrosion studies, wetting properties

Linear plasma simulators provide very long durations for PMI evolution: multiple plasma species can be tested, test articles can be exchanged easily and well-diagnosed. The plasma particles are typically mono-energetic, the magnetic field geometry is generally not prototypical of a tokamak, and PFC/plasma geometry aspects are not tokamak-representative. These facilities can be represented by the PISCES class of devices. There is a tritium and irradiated sample capable linear plasma simulator, i.e.TPE at INL.

Tokamak confinement facilities provide the actual environmental conditions, such as magnetic field geometry and PFC geometry, ion energy distribution, PMI conditions (e.g. recycling) and plasma core-edge to PFC consistency. U.S. facilities would include NSTX-U, DIII-D, and C-Mod. These devices have short plasma durations (plasma pulse few tens of seconds, long-pulse tokamaks may reach 300-1000 seconds), low duty cycles (~0.5 percent, plasma on-time to time between discharges), and low operation time (15 weeks run campaign, ~ 7 percent of a calendar year), making long continuous durations inaccessible. The capability for tritium plasma or irradiated PFC testing is largely absent, with the exception of ITER.

Toroidal magnetized systems to accommodate axisymmetric free-surface LM flow, such as LTX, are needed to explore the first wall PFC, and possibly also divertor, component concepts. The HIDRA facility is anticipated to be run as either a tokamak or stellarator and can offer another platform for integrated toroidal LM studies. These facilities typically have low B-fields and lower plasma performance than anticipated in next-step fusion facilities, but can provide the complex geometric features for exploration. Stellarator facilities may provide a long-duration environment for solid or liquid metal PFC/PMI studies.

Upgrades to existing facilities

Establishing the maximum steady state heat and transient fluxes, and operating temperature windows for actively cooled PFCs, requires the use of high heat flux facilities (e-beams or plasma-arc lamps). It would be beneficial to have these devices upgraded such that they can test neutron-irradiated materials and components. High temperature LM-MHD facilities being used for closed-channel blanket MHD experiments (e.g. MTOR/MaPLE) can also be upgraded to accommodate free surface flow and heat transfer experiments. Instrumentation upgrades for measuring flow that are compatible with high-temperature LM alloys are an essential element. Upgrade and application of existing accelerator-driven neutron sources (SNS, MTS) with suitable material test stations (temperature controlled irradiation conditions) would give more prototypical high-energy neutron damage data (high helium/DPA ratio). Upgrading of existing linear plasma facilities (e.g. PISCES, TPE) to enhance one or more features to be more prototypical of next-step devices. Possible features could include heat flux, particle flux, transient loading capability, actively cooled samples, ion energy improvements, and improved diagnostics. Upgrades (e.g. sample exposure systems) to existing toroidal facilities can provide testing of a broader range of liquid metal concepts, and provide more reactor-relevant plasma environments.

Leveraging international facilities

Leveraging one or more e-beam High Heat Flux facilities (e.g. Judith) could be of advantage since they already include the capability to expose neutron-irradiated samples. For the exposure of plasma-facing materials and components to high heat and particle fluxes at high fluence, linear plasma devices in the Netherlands could be utilized (Pilot-PSI or Magnum-PSI). Those devices could be used for testing solid and liquid plasma facing components. The advantage of Magnum-PSI is that it can test materials with steady-state plasmas and ELM-like frequent transients simultaneously. Leveraging existing accelerator-driven neutron sources like SINQ could be advantageous, albeit the disadvantage of having less controlled irradiation conditions. The use of IFMIF (part of the "Broader Approach" in the ITER activities), which is not yet operating, could provide a powerful high-volume fusion neutron test facility. This facility exists in virtually all countries' long-term plans for fusion development. A reduced version of this facility has been proposed.

New starts

New advanced linear plasma devices with more fusion-reactor prototypical plasma conditions could give the needed high-fluence data on hydrogenic retention, including those for a-priori neutron-irradiated materials. Such devices would also add value in the investigation of vapor shielding effects in high heat flux experiments with liquid metal plasma facing components. A new dedicated toroidal device, or extensive upgrade to an existing device, with a liquid metal main chamber wall and a flowing liquid metal divertor would give information for the overall integrated concept of a liquid metal as a PFC. Such a device would also produce information about the material migration of the liquid metal and the core-edge integration, providing information on plasma purity, plasma performance and stability. A short-pulse tokamak dedicated divertor experiment could provide the platform to better understand and optimize the complex behavior of the power and particle handling with both solid and liquid PFCs. The U.S. has limited high heat flux testing capability, and no LM-MHD facilities dedicated to free surface flow and heat transfer. Thus a new or extensively upgraded capability is needed.

III. 2. Action Plan

III. 2. 1. Establish maximum steady-state heat and particle fluxes, and operating temperature windows for actively cooled plasma facings components

Solid refractory materials are favored for plasma-facing applications, and new versions of these materials are being pursued (e.g. alloyed, fibers, composites, nano-structured) to address the shortcomings of the pure materials, both with and without neutron irradiation. Much more effort is required to develop monolithic or jointless structures using these new graded materials. These materials would need to be quantified in terms of their power handling, particle handling, and viable operating temperature windows. The integration with a relevant coolant, such as helium, is needed to understand the behavior under realistic conditions. This requires a level of design sufficient to maximize the component's performance to the extent possible.

Free surface liquid metal PFCs, whether fast-flowing, slow-flowing or capillary, require quantifying their power handling, particle handling, and operating temperature window. The integration with a relevant coolant for the substrate, in the case of capillary and slow-flow, and possibly also fast-flow, is needed to understand the behavior under realistic conditions. This requires a level of design sufficient to maximize the component's performance to the extent possible. A divertor application and a first wall application would require different test approaches due to the physical extent of the flows, the demands of inlet and outlet for the liquid metal, liquid metal inventories, background magnetic field geometry, and so forth. Testing of the actual liquid metal under consideration is required, and the requirement for lithium safety would influence where this testing can be done. The effects of MHD interactions and vapor shielding can be important, and the liquid metal to helium (substrate) heat transfer should be assessed.

These experimental activities would be done in a *high heat flux (HHF) facility*, typically e-beam driven for solid PFC materials and neutral beam or radiant heaters for LM surfaces to include magnetic fields and associated MHD effects. Particle testing would require a new high density, high power *linear plasma facility*, in which some simultaneous level of additional heating could also be applied. For liquid metals, the effects of vapor shielding can be important, and the liquid metal to helium (substrate) heat transfer should be assessed. The ability of flowing liquid metal systems with relevant MHD effects to be tested in a linear plasma device needs to be assessed. The capability to test irradiated materials in all these facilities is desirable. Active cooling is important, and must be accommodated in the testing facilities. The coolant loops must be capable of obtaining prototypical operating conditions. Existing loops are inadequate, and substantial new investment is required. The progressive nature of these tests would include simple material geometries and cooling approaches, evolving toward fully optimized designs. The application of the PFC in a *tokamak confinement facility* is desirable for the most complete representation of the fusion environment, and is likely required for examining first wall liquid metal concepts.

Theory/simulation development: The development and routine use of multi-physics tools for the modeling of PFCs under the various loading conditions is critical to developing designs. This can include thermo-mechanics, fluid dynamics and MHD, material evolution and service life, thermal hydraulics, high temperature behavior, melting and evaporation, fracture mechanics, and corrosion and mass transfer.

Theory/simulation development: The development of free surface liquid metal MHD tools to model the fluid under the varying conditions that PFCs endure, in conjunction with the substrate and other factors mentioned in the previous note.

III. 2. 2. Determine the tolerable maximum transient heat loads and transient durations for actively cooled plasma facings components

The objective is to determine the peak transient heat flux that a component can survive given the armor properties and cooling capabilities of the heat sink and the duration and footprint of the transient event. Melting and thermal fatigue leading to cracking and erosion are critical issues especially for the armor material and this sets the threshold for the number of such survivable events. Simulations using transient computational fluid dynamics solvers, dynamic thermo-mechanical solvers and kinetic Monte Carlo codes like the HEIGHTS package are necessary for design analysis and optimization[1].

These experimental activities would be similar to steady-state loading for solid and liquid PFCs, and would utilize the same facilities, albeit with loading durations and magnitudes that simulate the range from typical of the components thermal response time (slow ~ seconds, similar to several times steady state flux) to well below this time scale (ELMs, ~ 1 ms, very high flux). In this area, liquid metals have an additional challenge: the capability to simulate transient magnetic field and halo current impacts on liquid surfaces needs more assessment.

Extremely high heat flux, pulsed, high duty cycle, and high availability test beds are required for these studies. Fast-response diagnostics are a critical component of this research, such as fast response infrared and laser diagnostics, and spectrometers using fast focal-plane array detectors. Real-time evaporation monitors are required for liquid metals. High-speed data acquisition and control, and a high degree of automation are necessary for these facilities. Linear devices have the advantage of providing both steady-state high heat flux plasmas and frequent transient heat or plasma ELM-like pulses simultaneously.

In addition, the transient must usually be superimposed on a steady state loading condition to investigate the true component environment. This requires a high level of integration between the low duty cycle test apparatus and the pulsed device. Plasma surface heating is one example of synergistic testing required for a PFC qualification program. Others may include simultaneous neutron irradiation, neutral beam particle or RF heating effects, or simultaneous, short duration electromagnetic-induced mechanical loading.

III. 2. 3. Assess effects of tritium implantation, permeation, and retention in PFCs

The plasma side interaction with the plasma-facing materials provides a source of tritium and deuterium to the PFC, either a solid or a liquid. The energy that the particles have determines the depth that they penetrate the PFC. Once the hydrogen is in the PFC material it can diffuse, become trapped, and re-enter the plasma region or travel all the way to the coolant. Hydrogen will be generated by fusion neutrons through transmutation in the solid PFC matrix, and tritium will likely be permeating from the breeding region through the coolant to the PFC bulk, in either solid or liquid PFCs. The neutron irradiation of solid PFCs will produce trapping sites for tritium (hydrogen) in the bulk material. In the case of lithium liquid metal PFCs, tritium would be generated in the PFC liquid and transported with the liquid. Depending on the design, the tritium might be able to diffuse into the substrate. Understanding the processes and being able to predict them accurately is absolutely required in the case of tritium to allow the level of accountancy and safety anticipated for fusion reactors.

Tritium from the plasma can ultimately be retained under different mechanisms. In liquid lithium PFCs the liquid is expected to retain any tritium or deuterium that impinges on it due to strong affinity of lithium for hydrogen. This would imply that the liquid can carry the tritium out of the plasma chamber to where it must be efficiently removed before the liquid is brought back into the plasma chamber. For non-lithium liquid metals this would not be the case. The solid PFCs can retain the tritium via co-deposition with the PFC or other material, or have the tritium implanted in defects produced by neutron irradiation or PMI. Tritium can also be retained in dust or other debris created from solid or liquid PFCs.

Liquid PFCs would require an experimental facility that obtained flow conditions while exposed to hydrogen plasma — possibly a new linear plasma facility but preferably a longer-pulse confinement facility. The production of dust and debris should be quantified, together with the impacts of materials forming in the liquid metal that can affect this tritium behavior. For solid PFCs, although a linear facility may provide some information, a long-pulse confinement facility is best in order to provide a prototypical environment for the investigation of co-deposition of tritium by migrated material. Ultra-low amounts of tritium might be necessary for both liquid and solids in order to find the various reservoirs for tritium retention in the high hydrogen background of confinement facilities or linear devices. The neutron-induced trapping affect for solids will likely only be accessible in a linear facility with an irradiated sample, such as TPE at INL, or a new high-power, high-fluence linear device.

The facilities most able to examine the implantation physics are the *linear plasma facilities*, which may use deuterium as a surrogate (PISCES-like or upgrade/new), or use tritium in an appropriately qualified device (e.g. TPE at INL). Very long durations are needed to see the PFC surface evolution and its impact on the tritium processes. In addition, it is advantageous to observe both the hydrogen plasma and helium plasma bombardment simultaneously. A *tokamak confinement facility* is likely needed to get the integrated spectrum of plasma particle energies on the PFC; however, present facilities are unlikely to reach those of next step fusion devices in terms of particle energies. The examination of neutron-exposed samples is of particular interest to understand the enhanced trapping, and requires a facility that can handle such samples.

ITER may make significant contributions in tritium inventory and control for PFCs. The tritium retention via dust, co-deposition, and material migration will be studied in deuterium-deuterium (DD) and DT operations, and test blanket modules (TBMs) can ultimately contribute to our understanding and simulation capability. Tritium handling in all aspects, to varying degrees compared to a DEMO, will be accessible in ITER operations and the utmost advantage should be taken of this experience.

ITER may make a significant contribution in tritium inventory and control for PFCs if its schedule permits. Surrogate studies can be performed in the deuterium phase of ITER operation in preparation for deuterium-tritium. Deuterium retention in PFCs under ITER conditions will be measured and this can help benchmark existing computer codes and provide the impetus for new and improved computational tools. Studies of dust production, co-deposition and material migration will also be possible on ITER. Although beyond the 10-15 year scope of this report, the ITER test blanket module program will provide valuable data on tritium breeding efficiency, recovery and the operation of the tritium plant, providing a direct contribution on the DEMO design.

Theory/simulation development: Model tritium implantation, co-deposition, entrainment in dust and debris, and transport processes in real-time reactor systems. Tritium extraction from the coolant is an important issue for the tritium plant that ultimately affects PFC performance. Tritium can enter the PFC from the coolant as well as from the plasma, if the coolant inventory is not controlled. Existing modeling tools are 1D TMAP at INL, and various 2D and 3D complex geometry tritium transport codes. Expansion and improvement of these models and/or development of new simulation tools is needed to reach the level of accuracy and physics fidelity for tritium accounting.

III. 2. 4. Understand how bulk material modifications from fusion neutron-induced transmutation and helium production in PFCs affect the surface properties

All PFCs will see the unhindered 14 MeV neutron flux from the plasma, as well as the multi-scattered contribution from all directions with a wide range of neutron energies. The damage, transmutation and helium production in the vicinity of the plasma facing surface will be the highest anywhere in the fusion core. The impact of this damage, gas (helium and hydrogen), and transmuted atoms on the bulk material is the subject of much projection based on fission-neutron exposure, some high energy neutron exposure, and fast ion exposure data.

Although small D-D and D-T ion beam experiments are useful, DEMO fluence levels using fusion neutron-like spectra can only be approached with accelerators like the spallation neutron source, IFMIF, or MTS. The impact on bulk PFC properties is expected, such as the thermal conductivity, yield strength, ductility, creep and swelling. Helium and hydrogen retention, trapping and migration may affect recycling and surface erosion properties. Bulk physical properties may ultimately impact the plasma-facing surface region by affecting other temperature-dependent processes, and may have some coupling over the long exposure and surface evolution phase. This provides an unknown influence on PFCs that needs to be explored, since fusion neutrons provide a very powerful influence on material behavior in the fusion core.

For solids, this is accessed by irradiating the PFC material under relevant temperatures (and other parameters if possible, such as stress and hydrogen content), at different irradiation levels (fluence, DPA) and then exposed to plasma either in a linear plasma facility or confinement facility. These would be radioactive samples or assemblies, and would require the appropriate facility capabilities.

III. 2. 5. Identify and characterize the various processes that limit the lifetime of PFCs, including fusion-neutron irradiation, erosion, corrosion, thermo-mechanical cycling, and plasma-facing surface morphology evolution

Ultimately, the lifetime of PFCs prior to their replacement will be maximized through control of the plasma environment and the development of materials and designs that resist degradation to the greatest extent. Through detailed examination of the plasma material interactions and fusion neutron irradiation physics, and the incorporation of these phenomena into design constraints, the plasma facing components can be created with long service lives to provide their functions, under the multi-physics environment they will experience.

For solid PFCs the PMI, such as erosion and morphology evolution, can be accessed in linear plasma and toroidal confinement facilities. The thermo-mechanical and thermo-fluid features can be accessed directly in high heat flux facilities. Fusion neutron degradation requires an accelerator facility such as SNS, IFMIF, or MTS for high-energy neutrons.

For liquid PFCs the interaction of the liquid metal with the substrate material could be assessed in LM MHD flow loops used to study flow behavior and address chemical and metallurgical compatibility. There are difficulties in accessing these factors simultaneously and uncovering synergies that can exist under simultaneous phenomena. Linear plasma facilities can couple plasma particle exposure and heating, and also assess vapor shielding and particle recycling. Confinement facilities are limited in their operational capabilities but provide a more prototypical environment.

Combining neutron exposure with these other phenomena is not possible in-situ, but examining samples ex-situ is possible. The quantitative incorporation of these phenomena into the design and manufacturing of PFCs is needed to accurately predict replacement times; but proto-typical behavior will require a long-pulse DT confinement facility.

Development: The development of a qualification program similar to that established and executed for ITER divertor components is required, with the additional (or extended) features of higher fusion neutron fluence, high temperature operation, very long plasma exposures, very high cycles for transient loading if required, high heating and active coolant flow, and high heat/particle fluxes with LM free surface flow and high magnetic field. Significant basic material data are needed, both in solids and liquid metals, although the critical data are very different between these and can require completely different test stands or confinement devices to access. Overall the qualification flow is striving to reach fully integrated components, and fully prototypical loading and environmental conditions, although this access will be limited (e.g. we cannot produce the plasma/neutral particle energy spectrum that the first wall would see in an FNSF or DEMO on present facilities). Simulations of the plasma physics, PMI, and PFCs are absolutely required to project to the future environment.

III. 2. 6. Investigate the impact of advanced manufacturing on extending performance and lifetime limits

The new developments in solid component manufacturing (called advanced manufacturing) are expected to provide access to entirely new approaches for both material and overall part optimization for a given function and operating environment. This area is ideally suited to the complex multi-loading and multi-feature environment anticipated for PFCs. For solid PFCs, new alloys, eutectics and graded interfaces are now possible. Development of more ductile and radiation resistant refractory metal alloys with tailored physical properties is conceivable. For liquid PFCs, although less demanding for the plasma-facing surface, the detailed construction of the substrate to optimize liquid metal flow, heat transfer, mixing, wetting and capillary adhesion or other functions can also benefit from advanced manufacturing. The first wall, divertor, and special PFCs, although under some common loading features, see different detailed aspects (particle flux, plasma particle energies, plasma particle species, radiation, neutron wall loading, heat flux), while having significantly different functions and constraints (launching waves, minimizing negative impact on tritium breeding, or high thermal conductivity).

Theory/simulation development: Employ multi-scale and multi-physics modeling to develop advanced structural materials using advanced manufacturing (integrate materials-design-manufacturing aspects into one). The potential to prescribe a component's properties precisely over macroscopic scales is the promise of advanced manufacturing techniques. However, it can be seen that this involves an extremely large number of variables to choose from. The loading conditions and other constraints provide a tremendous combination of options, and this requires computational analysis to unravel. The development of multi-physics thermo-mechanical, CFD, LM MHD, material evolution, and PMI tools, and algorithms for deriving component geometry, materials, properties, and service life are needed. With this information, small mockups can be fabricated, quickly tested in the laboratory (e.g. small e-beam, laser thermal and universal mechanical testing) and considered for component scale-up. Additive manufacturing demands a paradigm shift in our work flow starting with analysis, then fabrication, testing and optimization of PFCs.

Chapter References:

[1] A. Hassanein and I. Konkashbaev, J. Nucl. Mater. **273**, 326 (1999).

Chapter IV

Priority Research Direction ‘B’ – Advanced Dissipative Divertors

IV. Priority Research Direction 'B' – Advanced Dissipative Divertors

<u>PRD-B</u>: Understand, develop and demonstrate innovative dissipative/detached divertor solutions for power exhaust and particle control, sufficient for extrapolation to steady-state reactor conditions

The divertor region is where the strongest interaction occurs between the hot boundary layer plasma (at ~10,000^o C or more), and the containing surface of the fusion device, which is made of ordinary matter. The divertor in future fusion devices will need to both handle an order of magnitude higher power density than present research tokamaks, and operate almost continuously, rather than the few hours per year typical of present devices. Viable divertor solutions are needed that can manipulate and stably control plasma conditions in the divertor so that the vast majority of the plasma power, which would otherwise concentrate and damage the target surfaces, is instead dissipated through the release of benign radiant heat that bathes a large surface area. The attainment of a detached plasma state, in which both the power and particles striking the target are reduced, may have additional advantages. The nearly continuous operation can damage the targets, e.g. by melting or erosion, and can also degrade the performance of the fusing plasma. Therefore, focused R&D is needed to address: *What are the physics mechanisms of divertor power dissipation and detachment? Can the power exhaust and erosion be acceptably controlled?* Promising innovative concepts have been identified that may dramatically enhance divertor performance and lead to more attractive divertor solutions. These involve manipulation of both the magnetic fields and the containing surfaces in the divertor, as well as the materials used for the target surfaces – solid, liquid and vapor. Therefore, the envisioned R&D also targets: *What are the effects of divertor magnetic topology, geometry and materials, including solid, liquid, vapor, on divertor solutions for future devices?*

IV. 1. Additional Background

The present knowledge base of tokamak divertor physics is not complete enough to specify a divertor "solution," i.e., a combination of magnetic topology, divertor geometry, materials, active control techniques, etc., that is *sufficient* for high-duty-cycle, high-power DT tokamaks, such as an FNSF or a DEMO. In fact, we do not know that a solution exists even in principle. The requirements are nevertheless clear: the integrity of the divertor targets must be maintained while achieving acceptable core plasma performance. Destruction of any target surface would be assured if the deposited power flux density on the target, $q_{\perp,t}$, exceeds thermal engineering limits, ~10 MW/m^2. Solid targets such as tungsten or graphite will also be destroyed under high-duty-cycle plasma operation if the net erosion rate is too high. It is thus imperative to control the divertor plasma temperature, T_t, below ~5 eV near the target to avoid unacceptable erosion. Furthermore, surface damage and modification by energetic helium implantation (e.g. tungsten fuzz) must also be avoided. Self-annealing liquid metals might be able to meet this challenge, but they introduce new elements to divertor/boundary/core physics for which an experience base needs development. Based on current projections, ITER may have ~4x the power exhaust flux density encountered in present experiments and it is expected that a DEMO may be ~4x higher than ITER. Thus divertor solutions are ultimately needed that can demonstrate order-of-magnitude improvements in power handling over present experience, while having acceptable divertor target-plate erosion and compatibility with maintaining good core confinement. These are the major research challenges facing the development of viable divertor solutions for future devices.

The primary external drivers of the divertor are the power entering the SOL, P_{SOL}, and the "upstream" density, n_u, the plasma density at the low-field side separatrix. These are imposed by core and pedestal plasma performance requirements and create the problem for the divertor. An acceptable divertor solution will be one that for a given P_{SOL} and n_u manipulates divertor geometry, magnetic configuration, with active radiation and particle control, so as to achieve desirable values of , $q_{\perp,t}$ and T_t, without degrading core plasma performance. A large body of knowledge has been assembled from tokamaks with "conventional" magnetic geometries and with solid target plates at a variety of inclination angles with respect to the magnetic field, $\theta_{\perp}$. The ITER 'vertical target plate divertor' was designed on this basis. It may be possible to avoid unacceptable target heat flux ($q_{\perp,t} \leq 10$ MW/m^2) and erosion by operating in a highly "dissipative divertor" regime[1] with $T_t < \sim 5$ eV, maintaining significant plasma fluxes to the target but achieving low net target erosion via prompt re-deposition processes[1,2], or with lower $T_t \sim 1$ eV by exploiting volumetric loss processes, so that ion impact energies are below sputtering/damage thresholds and plasma fluxes to the target plate are reduced, hence leading to a "detached divertor"[3].

A successful divertor solution for a DEMO etc., will be one that is capable of spreading the ~ 1 mm heat flux channel width (as measured at the outer midplane) to a size ~1 meter in the divertor fan. Significant progress has been made since ReNeW in the development of innovative divertor magnetic designs such as X-divertor[4], Super-X divertor[5], Snow-flake divertor[6], X-point target divertor[7], divertor shaping/baffling variations, e.g. deep slot V-shaped divertors[8] and liquid metal ideas such as the lithium vapor box divertor. This progress has been achieved by optimization and by introducing potentially new impactful physics: enhanced plasma turbulence to spread heat flux over larger surface areas; stabilization of the "detachment front" within the divertor volume via toroidal flux expansion and/or poloidal flux expansion with increased neutral-plasma interactions; reduction of peak heat flux via interaction with an X-point in the divertor volume; enhancement of non-coronal radiation via charge exchange and short impurity residence times; interaction with a high density vapor to safely extract energy and momentum, etc.

A number of promising advanced divertor strategies have emerged over the past decade, see Fig. IV-1.

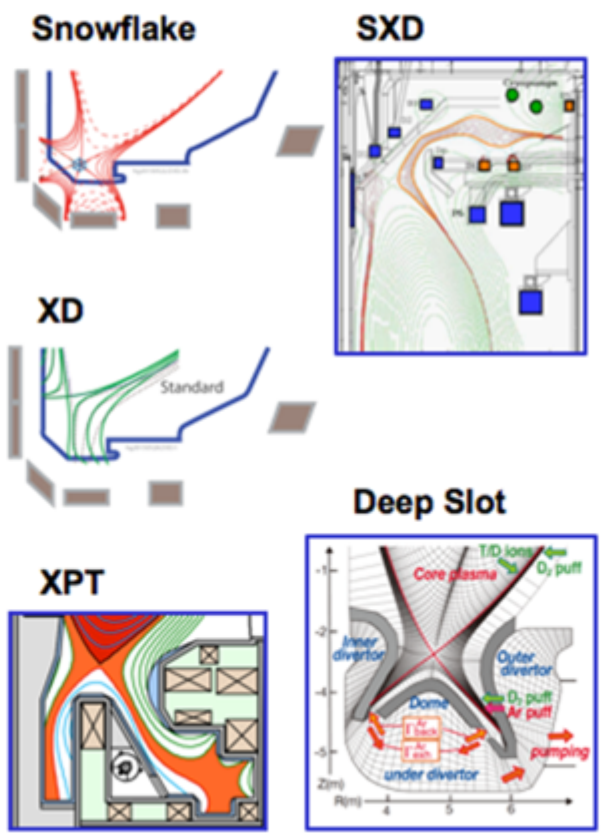

Figure IV-1: Advanced divertor configurations: Snowflake, X-Divertor, X-point target, Super-X Divertor, and deep slot divertor

Since the 2009 ReNeW report, significant theoretical and numerical advances have been made, advanced divertor experiments have begun, tokamaks have been designed with advanced divertors integrated into their design (e.g., MAST, HL-2M, ADX) and new divertor-focused experimental and theoretical programs are underway on DIII-D, NSTX-U, and C-Mod. Advanced divertor geometry proposals aim to reduce the power density on the divertor plates and enhance and control dispersal of power by divertor radiation and transport. The key features of the various approaches are:

1) Poloidal flux expansion. Reducing the poloidal component of the magnetic field increases the spacing between flux surfaces, and reduces the angle between the field line and the divertor target, reducing the perpendicular power density. There are mechanical alignment limits to how small an angle is beneficial. Flux expansion also leads to increased field-line length.
2) Increased field-line length provides a longer distance over which thermal resistance applies and over which the cross-field transport, radiative, and atomic processes can provide heat dispersal. In addition, regions of extreme length in the vicinity of (perhaps multiple) X-points are predicted to experience qualitatively different transport phenomena that increase the number of active divertor strike points or act as a virtual target.

3) Toroidal flux expansion involves divertor field lines that move to larger major radius and hence to smaller total magnetic field before striking the divertor plates. The flux tube expands, reducing the parallel power density. Additional predicted benefits include improved control of the detachment thermal front.
4) Enhanced neutral gas interactions aim to reduce solid target heat loads by raising the density of recycled neutrals and/or controlled impurities and the resulting losses. Specific baffle geometries include deep slots and at their most ambitious, gas-box divertors which replace the plate with a gas cushion.

These magnetic topologies/geometries are obtained by adding or modifying poloidal field coil locations/currents to create extra X-points: (1) at or near the main X-point (SFD), (2) intercepting the peak heat flux behind the divertor target (XD) or as a 'virtual target' in the divertor volume (XPT), and (3) bending the divertor leg toward larger major radius (SXD, XPT). Note, however, that limitations imposed on poloidal field coil locations and their maximum allowed currents affect the ability of engineering to create and actively control these geometries for any given device.

Proof-of-principle experiments have already been performed at low and moderate power flux levels showing that the control of divertor magnetic geometry/topology has great potential. Experiments with SFD and XD configurations (TCV, NSTX, DIII-D) demonstrate that these concepts can be created and that they behave roughly as anticipated. Results include surface heat flux reduction and an early onset of detachment due to flux expansion, as well as evidence for enhanced divertor radiation and enhanced cross-field heat transport. Further experiments are needed to assess overall performance improvements (e.g., heat flux reduction while maintaining core performance) relative to an ITER-like vertical target plate divertor, which employs flux expansion via target tilt angle rather than poloidal field reduction.

Deep slot V-shaped divertors (and variants) seek to greatly enhance the heat-flux handling over that of an ITER-like divertor primarily by physical (solid) geometry. These V-shaped divertors employ a highly optimized target shape and neutral baffling and they increase magnetic field line length to the divertor plate by moving the divertor plate farther away from the main plasma X-point in poloidal cross-section. A practical advantage is that they impose less severe engineering requirements on the poloidal magnetic field coil set compared to advanced magnetic divertors.

While solid divertor targets remain the divertor basis for future reactor designs, liquid PFCs offer a number of promising features and are therefore appropriate for investigation. Liquid metal divertor surfaces may be able to simultaneously address the divertor heat flux handling challenge, together with divertor target erosion and fuel retention concerns. This boundary condition may also substantially improve plasma confinement, as observed with lithium coated walls, e.g., in TFTR, LTX, NSTX, TJ-II, and FTU (see section II.4 and PRD E in Chapter 7). Candidate liquid metals presently under consideration include lithium, tin, and gallium and molten salts, such as FLiBe and FLiNaBe. Engineering PFC designs include liquid metal films flowing on solid surfaces, liquid metal pool targets, liquid metal droplet curtains, and capillary porous targets. However, the question posed by ReNeW "Can practical liquid surfaces be developed as an option for solid surface plasma facing components?" remains open, as little experimental progress has been made since 2009 and liquid metal divertors are still in their infancy. Critically, the use of liquid metals in the configurations typical of tokamaks and other fusion energy devices pose new challenges that are beyond the present state of understanding of materials and surface science. The various phase interfaces, as well as novel fluid dynamics questions associated with the different types of flowing liquid boundaries are multi-scale interdisciplinary problems. Thus, improved understanding and design of the "liquid boundary" approach to PFCs require linking the atomistic scale at the solid-liquid and liquid-plasma boundaries to the macroscopic motion of flow of liquid metals in the presence of electro-magnetic fields.

Steady-state plasma operation imposes another constraint, in addition to power handling and divertor target plate erosion control. It has been estimated in a number of studies that high duty cycle tokamaks starting with ITER will experience rates of net erosion and deposition of main chamber wall plasma facing component (PFC) material in the range of $10^2 - 10^5$ kg/year. Even if the net erosion (wear) problem can be solved by periodic in situ refurbishment, the deposition of such massive quantities of material has the potential to seriously interfere with tokamak operation. It will therefore be essential to manage material deposits, i.e., the management of PFC "slag" accumulation in the divertor and elsewhere. This requires detailed knowledge and perhaps methods to actively control material migration/deposition plus a technology that can accommodate PFC slag removal requirements, including periodic cleaning methods and/or the use of liquid metals.

In addition, damage from fusion neutrons imposes important constraints on all divertor magnetic geometry designs and choices of materials. The tritium Breeding Ratio (TBR) is another critical metric that must be considered when comparing different geometries. Although these neutron transport, material damage, and TBR considerations do not directly affect the plasma issues discussed in this PRD, they are critically important in evaluating and comparing reactor divertor design choices.

The key integrated performance questions include:

- Is it possible to achieve steady-state heat loads below 10 MW/m^2 and $T_t \sim$ a few eV, as well as net erosion rates of solid targets below ~1 mm per year under reactor relevant conditions, compatible with core performance and pumping?
- Can migration/accumulation, slag production and tritium retention be managed for very high duty cycle conditions?

- Are fast flowing or evaporative/radiative liquid target solutions compatible with high performance plasmas?

IV. 2. Main Scientific Questions

A central question for all advanced divertor concepts is: can they provide the order of magnitude improvement in power handling and nearly complete suppression of erosion needed for a DEMO while not compromising core/pedestal plasma performance? Present experiments have uncovered what appears to be an unavoidable tradeoff between maintaining good core confinement and protecting divertor targets from destruction (thermal load and erosion), prohibiting the application of existing divertor solutions to DEMO-class devices. The new physics ideas embodied in advanced divertors concepts have the potential to meet these challenges by keeping the "divertor detachment front" from impacting the pedestal and reducing core plasma performance, or by operating highly dissipative attached divertor regimes, perhaps facilitated by the use of liquid metals.

IV. 2. 1. *What are the physics mechanisms of divertor dissipation, detachment, stability and control?*

To achieve dissipative and detached divertor conditions and their stability, it is necessary to identify the basic physics so as to develop the means of control:

Heat losses

- Hydrogenic radiative and charge-exchange power losses: these are the most basic dissipative processes and the ones potentially least disturbing to the confined plasma. How can they be maximized?
- Low-Z intrinsic and extrinsic impurity radiative losses: what is the optimal low-Z radiator for a given divertor concept with regard to both the divertor and confined plasma requirements?
- Radiation trapping: this reduces the effective hydrogenic radiative cooling effect. How can it be minimized?

Momentum losses

- Momentum/pressure losses: these can achieve the positive effect of reduced target fluxes but they increase the required volumetric power losses to achieve low T_t and high n_t. How to manipulate and optimize these losses?
- Volume recombination: this can achieve the positive effects of reduced fluxes of particles, momentum and power to the target. However, this requires particularly low T_t, $< \sim 1$ eV, which may require stronger radiative loss and may be more disturbing to the confined plasma than warmer divertor conditions. How to optimize this process?

Heat and momentum spreading

- Role of cross-field drifts: these can play an important role in spreading heat between divertor legs, decreasing in/out detachment asymmetry owing to their presence in the main SOL. Power, momentum and particle transport by turbulence and collisional processes: this spreading of the target loads is highly desirable; how to maximize it?

IV. 2. 2. *What are the effects of divertor magnetic topology, geometry and materials, including solid and liquid?*

Choices of divertor magnetic topology, geometry and materials can have profound influences on divertor performance. What plasma and atomic physics aspects of dissipative and detached divertor regimes including those listed above are affected: (i) by divertor magnetic geometry for each of the advanced divertor options, specifically by the increased poloidal and/or toroidal flux expansion and by connection length? (ii) by changes to the solid structural shaping/baffling of the divertor? (iii) by the materials used — solid, liquid?

Magnetic configuration. Along with improved codes, theoretical developments since 2009 have proposed physics-based metrics to help characterize: (1) how the new geometries of advanced divertors affect the overall divertor detachment behavior and its impact on pedestal and core confinement; (2) the onset of new instabilities (e.g., churning modes) that can spread the exhaust power over a larger area, especially during ELMs. Do experiments find that these are appropriate metrics? Are there other simple measures (empirically and/or or theoretically derived) that can be used to project the overall behavior of an advanced divertor concept to an integrated reactor setting?

Gas dynamics – physical structure. The benefit of shaping the solid structure of the divertor has long been recognized. In particular, a vertical divertor target directs recycling fluxes toward the separatrix, thereby increasing dissipation in the spatial region carrying the highest power. Accordingly, ITER has adopted such a divertor structural shape. A number of computer simulations studies indicate further major benefits to yet more closed divertor baffling, which are referred to as “slot divertors”[8]. These types of designs reduce $q_{\perp,t}$ for given P_{SOL} and n_u. To some extent these simulation results have been supported by experiments in C-Mod, AUG and JET. However, diagnosis is difficult for slot divertors due to restricted line-of-sight access. As a result the amount of experimental information on slot divertors is at present too limited to adequately assess this option. How much further performance benefit might be obtained by optimizing slot divertor configurations?

Liquid metal divertor target. For liquid metal divertors the outstanding questions that need to be addressed in experiments are similar to the ones discussed for solid target divertors, namely: (1) compatibility of liquid metal targets with high-performance pedestal and high core confinement, including the pedestal structure and stability; (2) establishment of physics of steady-state divertor regimes with liquid metal targets, e.g. operating windows in terms of radiation, tolerable heat fluxes, impurity fluxes and screening, etc.; (3) assessment of interaction of liquid metal target concepts with transients, including ELMs. In addition, integrated tokamak tests should address combining optimized liquid target designs with optimized magnetic divertor geometries.

IV. 2. 3. *What are the physics mechanisms underlying near SOL heat flux width and its scaling?*

It appears that the near SOL heat flux width has now been reliably measured in multi-tokamak studies and a consistent scaling established, albeit for attached divertor operation. However, it is still unclear what mechanism controls this width: turbulence, neoclassical transport, stability considerations related to steep gradients near the separatrix, etc. Therefore, it remains to be seen whether the near SOL heat flux width scaling will still hold for detached conditions and whether it can be extrapolated to next-step devices. While considerable progress has been made in characterizing intermittency, the role of blob-filaments, convective transport, etc., reliable predictive tools for calculating heat and particle fluxes on plasma facing components at the divertor and first wall are not presently available. The competing and/or synergistic effects of neoclassical orbit widths and turbulence are not understood in the important narrow heat flux channel closest to the separatrix. The challenges also include large fluctuation levels, sonic flows and sheaths, kinetic effects on both electrons and ions, and the role of particle momentum and energy sources and sinks, neutral and atomic physics such as friction, ionization and radiation. Further, plasma instabilities, turbulence and anomalous transport are also affected by divertor magnetic configuration and X-point geometry. For example, it has been recently demonstrated in TCV ELMing H-mode experiments that a large fraction of the power flows to the secondary strike points vs the primary strike points, suggesting enhanced transport near the null point in near snowflake configurations, possibly caused by the churning mode[9].

IV. 2. 4. *How can we extrapolate to reactor regimes?*

It is imperative to operate in highly dissipative/detached divertor regimes to control divertor heat flux and erosion for high-duty-cycle, high-power, next-step fusion devices. Atomic, molecular and turbulence physics that control volumetric loss processes in detached/dissipative divertor conditions depend in a strongly non-linear way on both absolute density and temperature. Present tokamaks can access relevant values of density and temperature near the divertor target at moderate upstream heat flux, providing extremely valuable information on the fundamental underlying physics. But a much higher upstream heat flux, plasma pressure, as well as a wider range of geometries and materials need to be investigated to develop a viable divertor solution for DEMO. A dedicated tokamak divertor device will be needed to further advance understanding in this critical area for fusion development.

IV. 3. Action Plan

The first step is to extend our knowledge base and theoretical understandings of dissipative divertor physics to include novel magnetic topologies, geometries and materials, and to incorporate new physics that might arise. Enhanced diagnostics, theory and modeling activities are essential for all steps in this plan. Existing U.S. tokamaks can access reactor-relevant dissipative/detached divertor regimes while exploring novel geometries and materials; these capabilities should be exploited. Targeted collaborative research on overseas facilities can also contribute. The next step is to develop approaches that have the potential to handle extreme power flux densities, as anticipated for DEMO. Exciting possibilities have been identified. The final step is to explore and demonstrate divertor solutions for power exhaust, erosion suppression and particle control at near reactor-level conditions with key plasma/atomic physics parameters approximately matching that of a reactor throughout the divertor volume. These latter steps include the design, construction and operation of dedicated facility, a Divertor Test Tokamak.

IV. 3. 1. Advance physics understanding of advanced divertors (diagnostics, theory & modeling)

- Make high resolution measurements of plasma properties and dissipation processes in the divertor and in the SOL near the separatrix
- Develop fully predictive models of divertor dissipation/detachment and near SOL physics

The ability to project the behavior of the SOL and divertor in future devices requires detailed knowledge of the underlying physical processes – information that is lacking at present, due in part to the extreme complexity and richness of the physics encountered in these regions:

(a) All states of matter interact simultaneously: solid, liquid, gas, plasma.
(b) Plasma profiles in the boundary and divertor are inherently two-dimensional. In some cases, time-evolving phenomena involving three spatial dimensions must be described.
(c) Plasma turbulence in the boundary is extreme – fluctuation amplitudes approach, and often exceed, local time-averaged values, particularly in the far scrape-off layer
(d) Plasma turbulence and transport varies in strength significantly with location in the boundary plasma and divertor; the underlying drive mechanisms are highly varied.
(e) The SOL and divertor geometry is intricate, requiring an extensive set of high spatial and temporal resolution diagnostics.

Thorough diagnosis of the plasma properties is needed to identify the controlling physics. High spatial resolution measurements of time-averaged plasma quantities are needed – n_e, T_e, T_i, parallel flow velocity $v_{||}$, plasma potential Φ_p – at multiple points over the entire domain: main scrape-off layer on low-field and high-field side; divertor regions in inner and outer legs. It is also essential to unfold time-averaged distributions of impurities and their flow velocities, including multiple species and charge states, as well as radiated power distributions.

Plasma turbulence is, to a large degree, the physics that *defines* the boundary plasma and divertor. Understanding the scaling of the heat flux width at the outer midplane, transport in the far SOL and spreading of the heat footprint on the target plate will be attained only with the development of models that can accurately compute fluctuation amplitudes and transport. Detailed measurements of plasma fluctuations (e.g., n_e, T_e, Φ_p, poloidal magnetic field) – including frequency and wavenumber spectra, correlations, phase angles, structure velocities, etc. – at multiple locations in the boundary/divertor domain are essential.

Development of reliable reduced-physics interpretive models, and ultimately, fully predictive models based on next-generation computational tools are required. Edge plasma fluid transport codes (2D and 3D) presently provide important guidance and interpretation for experiments. However, these employ ad hoc prescriptions of cross-field transport, 'adjusted' to match measured time-averaged profiles. This approach cannot predict what will happen as input parameters are changed beyond those measured, including different plasma regimes, magnetic geometries/topologies, and mechanical baffling/shapes. Reduced, first-principles physics models that incorporate turbulence effects and comprehensive models that fully describe plasma/neutral dynamics are required. These include fluid turbulence models/codes and first-principles gyro-kinetic and fully kinetic codes. Research foci must address: (1) standard magnetic divertor topology, including closed and open magnetic flux surfaces, and private flux regions; (2) innovative divertor configurations for multiple X-points and magnetic flux expansion (poloidal and toroidal); (3) target plate shaping, baffling/pumping and cladding materials; (4) plasma-neutral and plasma-impurity atomic interaction processes that dissipate plasma momentum and heat; (5) atomic and molecular physics, including photon transport in plasmas that are optically thick to line radiation. This research will result in improved understanding and validated predictive capability for key phenomena such as: (1) control of detachment, its onset and thermal front position; (2) loss of detachment during ELMs; (3) divertor radiation/dissipation levels; (4) effects on pedestal and core confinement.

The recent insight on the scaling of the upstream heat flux width is a good example of how empirical observation, reduced physics models and first-principles model development can combine to reveal controlling physics. It is anticipated that further advances in boundary/divertor physics will continue successfully along these lines.

IV. 3. 2. Exploit and upgrade existing divertor experiments

- Enhance runtime, diagnostics and personnel resources for divertor physics
- Explore the power handling/performance limits of existing divertor configurations
- Upgrade divertor configurations and materials (solid and liquid) and explore power handling/performance limits

Atomic, molecular and plasma turbulence physics that control divertor volumetric power loss processes depend in a strongly non-linear way on plasma temperature and density, particularly under conditions of interest for a reactor in which divertor erosion can be suppressed, i.e., $T < \sim 5$ eV (see the Addendum at the end of this chapter). This non-linear behavior is akin to a phase transition (plasma-gas). Existing tokamaks can access these reactor-relevant parameters near their divertor target plates with moderate levels of upstream parallel heat flux, $q_{||u}$, compared to future tokamaks; in this case, lower $q_{||u}$ is compensated by reduced upstream density, n_u. Existing tokamaks should therefore be fully exploited to study the non-linear atomic and molecular physics of dissipative divertor plasmas. Enhanced runtime and personnel resources are needed. Enhanced edge diagnosis and modeling, as discussed above, is also essential. Present tokamaks should also be exploited to establish, more completely than has been done to date, the power handling limits of the divertor configurations readily accessible and their compatibility with maintaining good core plasma performance – employing a combination of optimized core/edge impurity seeding and active feedback control.

DIII-D, NSTX-U and C-Mod are very well suited for this task. They have complementary capabilities, allowing an important range of target-plate materials, geometries and magnetic geometries/topologies to be explored:

DIII-D

- Explore/compare divertor/SOL physics (low-Z target plates): horizontal target plate divertor vs. snowflake divertor vs. X-divertor
- Explore divertor power/performance limits and their compatibility with pedestal/core with core/edge/divertor seeding and feedback control
- Potential upgrades: deep-slot, high-Z heated divertor; SXD; replenishable low-Z wall coatings

NSTX-U

- Explore/compare divertor/SOL physics with low-Z target plates, high-Z target plates and lithium coatings; explore/compare: horizontal target plate vs. snowflake vs. X-divertor
- Explore divertor power/performance limits and their compatibility with pedestal/core with core/edge/divertor seeding and feedback control
- Explore lithium vapor shielding physics

C-Mod

- Explore/compare divertor/SOL physics (high-Z target plates, high heat flux density and poloidal field): vertical target plate vs. horizontal target plate vs. snowflake vs. X-divertor
- Explore divertor power/performance limits and their compatibility with pedestal/core with core/edge/divertor seeding and feedback control

IV. 3. 3. Leverage participation in overseas experiments

- Complement U.S. facilities and research program with targeted collaborations on divertor and near SOL physics topics
- Maximize U.S. benefits from ITER in divertor physics

Facilities abroad provide opportunities to strengthen the U.S. research portfolio in targeted areas. Participation in ITER is a key component. Specific examples include:

Advanced magnetic divertors: TCV is studying advanced divertors at low power densities and can address basic physics questions, including the effect of extending the outer divertor target to large major radii for increased detachment front stability and to explore X-point target configurations. MAST will study the super-X divertor configuration as it completes its upgrade. HL-2M will be implementing a 'tripod' divertor. Participation in planning and execution of such experiments could be valuable; U.S. facilities are not (yet) able to explore these configurations.

High-Z divertor PMI: ASDEX-Upgrade (tungsten first wall and divertor) and JET (ITER-like wall) are presently wrestling with the issue of maintaining core plasma performance with high-Z PFCs. WEST will explore tungsten divertor operation as it qualifies divertor components for ITER. C-Mod has substantial experience in this area; NSTX-U is planning to install high-Z PFCs; DIII-D is considering this also. There is a common interest in resolving these challenges for ITER and beyond.

Long-pulse material erosion/migration: As EAST and KSTAR increase performance levels, they will become platforms to investigate material erosion/migration issues for long-pulse operation. JT-60SA will enter this arena as well and EAST also has an aggressive lithium wall conditioning campaign. Ongoing fruitful collaborations with U.S. scientists can and should be continued. Although these machines will fall short of obtaining ITER or DEMO divertor conditions, particularly in fully non-inductive scenarios, they could provide valuable insights.

ITER divertor physics: In addition to concern about a projected ~1 mm heat flux width and the need to sort out the controlling SOL physics, ITER has additional concerns, including: loss of divertor detachment due to transient events or toroidal non-uniformities in divertor seeding; the effect of divertor seeding on divertor/core performance and the means for its optimization; the use of neon in place of nitrogen as a seeding gas, the latter having cryopump and tritium plant incompatibilities. U.S. experiments/researchers should participate in these areas. Since the projected performance of ITER's divertor relies primarily on modeling using the SOLPS code, ITER is seeking to validate this code with data from existing experiments. U.S. tokamaks should play a central role here as well. Reciprocally, when ITER attains plasma operation, it will be a definitive test of our understanding of heat flux width scalings and the performance of conventional divertors. Maintaining close collaboration/participation with ITER, such as through the ITPA, is essential.

IV. 3. 4. Develop a U.S.-led divertor test tokamak facility

- Understand the role of magnetic configuration, chamber geometry, and target materials in a dedicated US-led, Divertor Test Tokamak
- Discover and demonstrate new dissipative divertor solutions at reactor-level power densities
- Develop robust power and particle handling solutions directly applicable to steady-state fusion power systems

As discussed above, existing divertor tokamak experiments, properly diagnosed, can provide important insights into the complex physical processes at play in a dissipative divertor. Elements of advanced divertor schemes should be implemented (some already have been), providing proof-of-principle tests, with an eye towards potentially improving divertor performance in a reactor setting. However, for robust, power and particle handling solutions directly applicable to steady-state fusion power systems, a dedicated divertor facility will be needed.

The need for a dedicated Divertor Test Tokamak (DTT): As discussed in Section II.1, the heat flux density entering into the divertor of a DEMO is projected to be at least an order of magnitude higher than in present tokamaks. Dissipative divertor conditions, with target plate electron temperatures low enough to suppress erosion/damage, is therefore much "easier" to achieve in present experiments; one does not need to attain divertor power loss fractions as high as required for ITER, FNSF or a DEMO (see chapter end Addendum). This is beneficial in the short term; it allows present experiments to study dissipative divertor phenomena but beyond that, the wider range of reactor-relevant parameter space cannot be accessed. In a reactor, physical processes that enhance dissipation (e.g., non-coronal radiation, turbulence/transport, plasma/neutral/vapor interactions) must be pushed to unprecedented levels. Plasma momentum-loss mechanisms, such as those that arise via charge exchange and recombination, must also be greatly enhanced. Advanced divertors with extended geometries, gas-dynamic configurations and/or different materials, such as metal vapors, must be employed to attain these extreme regimes. Because plasma-atomic processes are highly non-linear, it will be valuable to more closely approach the absolute parameters expected for reactors.

The idea behind a DTT is to do just that – create a *dedicated* tokamak capable of producing reactor-level plasma parameters in its divertor – while at the same time having the divertor volume and flexibility to explore a variety of advanced divertor concepts: magnetic geometries, topologies, mechanical shapes, gas dynamic options and different target materials including liquid metals. An extensive suite of boundary/divertor/PMI diagnostics would be deployed to elucidate the science, challenge theories, validate models and develop first-principles understandings. A high field, short pulse, high-power-density tokamak constitutes a new possible option for a DTT[7], complementing earlier studies[10,11]. By matching upstream heat flux, pressure and exhaust channel width, the DTT can be configured so that plasma/atomic conditions closely match those in a reactor.

The European Fusion Development Agreement (EFDA) roadmap to a fusion reactor[12] noted that: "As the extrapolation from proof-of-principle devices to ITER/DEMO based on divertor/edge modeling alone is considered too large, a gap exists in this mission. Depending on the details of the most promising chosen concept, a dedicated test on specifically upgraded existing facilities or on a dedicated Divertor Tokamak Test (DTT) facility will be necessary. The DTT could be either a new or an upgraded facility, entirely devoted to the divertor problem, but with sufficient experimental flexibility to achieve the overall target." There is opportunity for a strong U.S.-EU collaboration in this area; however, as yet no plans are in place for the EU to establish such a facility nor are there plans for a DTT elsewhere in the world.

It is desirable that the DTT have the ability to test a full range of divertor concepts. This might be facilitated by employing a modular vacuum vessel in which the divertor region can be changed. For the case of liquid metal options, one could envision developing and testing basic concepts first in test stands, embodying the most promising approaches into divertor modules and then testing them in the DTT. Provision for heated divertor targets could also be made. A DTT could satisfy research needs in a number of other science areas as well. As noted in PRD-C, a dedicated DTT could explore the physics and test potential solutions for main-chamber wall components, including RF actuators. As noted in PRD-E, a dedicated DTT should provide integrated tests of divertor/pedestal/core compatibility with high confinement regimes and extend the science of pedestal/core physics. Through the process of experiment-driven science and discovery, a DTT would rapidly advance fundamental understanding, stimulate game-changing innovations, and facilitate U.S. world leadership in these most important science areas.

We recommend establishing within the FES strategic plan a national working group to examine design options for a DTT facility. This facility should be capable of producing reactor-level plasma parameters in its divertor – while at the same time having the divertor volume and flexibility to explore a variety of advanced divertor concepts: magnetic geometries, topologies, mechanical shapes, gas dynamic options, and different target materials including liquid metals. In our judgment, the development of this science and technology is the most critical issue for advancement to DEMO, and the country that leads here will be in a leading scientific and technological position for the future.

IV. 4. Addendum: Requirements for Obtaining Dissipating/Detached Divertors

How might an advanced divertor be designed so as to obtain dissipative/detached divertor conditions in future high-power high-duty-cycle DT tokamaks at the required power exhaust densities? For conditions approaching detachment, a two-point model analysis[18] provides a simplified point of reference and rough quantification.

Our current understanding of divertor physics is based on a large body of knowledge assembled from tokamaks with "conventional" magnetic geometries and with solid target plates at a variety of inclination angles with respect to the magnetic field, $\theta_\perp$. The ITER "vertical target plate divertor" was designed on this basis. It may be possible to avoid unacceptable solid target-plate erosion by operating in one of two regimes [18, 19]: (A) a "*detached divertor*", low temperature target regime, i.e., T_t ~ few eV, in which ion impact energies are below sputtering/damage thresholds and plasma fluxes to the target plate are reduced by volumetric loss processes (hence the term *detached* divertor) and (B) a "*dissipative divertor*" regime, operating with slightly higher T_t, maintaining significant plasma fluxes to the target but achieving low net target erosion via prompt redeposition processes.

To date, divertor scenarios have largely been designed around the detached divertor regime but the dissipative divertor scenario has several attractive features. This regime is achieved when the ionization mean free path for the sputtered impurity neutral $\lambda_{\text{ioniz}} <\sim \rho_{\text{DT}}$, where ρ_{DT} is the DT Larmor radius. In this case the strong electric field in the magnetic pre-sheath promptly returns the ionized impurities to the target [20]. A rough estimate for B ~ 5T, T < 10 eV and n > ~ 10^{21} m^{-3} yields $\lambda_{\text{ioniz}} < \sim 5\,\rho_{\text{DT}}$ for both high-Z and low-Z PFCs. Fortunately, such target plate conditions are also acceptable with regard to manageable exhaust ($q_{\perp,t} \propto sin\theta_\perp$ <,~ 10 MW/m^2), particularly for small $\theta_\perp$.

Target Conditions Consistent with Acceptable Power Loading

The target plasma power deposition density is $q_{\perp,t}^{\text{plasma}} = q_{\|t}\sin\theta_\perp = \gamma_{\text{sheath}} n_t k T_t c_{st} \sin\theta_\perp$, which for a sheath heat transmission coefficient $\gamma_{\text{sheath}} \approx 7.5 + \left(E_{\text{ei}}^{\text{surface-recomb}} + E_{\text{atom-atom}}^{\text{surface-recomb}}\right)/kT_t \approx 7.5\left(1 + 2/T_t\right)^1$ and $q_{\perp,t}^{\text{plasma}}$ = 5 MW/m^2 (which allows for some radiative and charge exchange neutral particle power loads to be added), then gives the maximum tolerable target plasma density $n_t^{\max}$, Fig. IV-2.

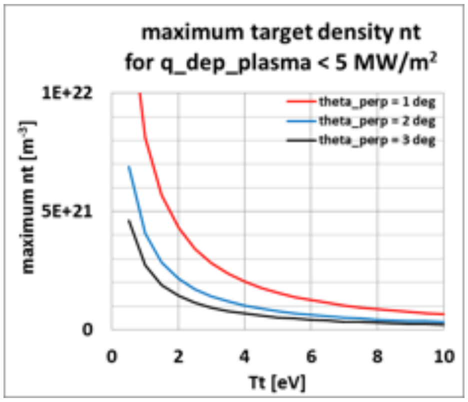

Figure IV-2: The maximum tolerable plasma density at the target, n_t, as a function of T_t assuming that a plasma power deposition density $q_{\perp,t}^{plasma}$ = 5 MW/m^2 can be handled. Various assumed $\theta_\perp$, the angle between the target surface and B.

Volumetric Power Dissipation Required to Accommodate ITER and DEMO heat fluxes

Target temperatures and densities at the end of a flux tube of length L is given by[3,13]:

$$T_t \propto \frac{q_{||u}^{10/7}(1-f_{power-loss})^2}{L^{4/7} n_u^2 R_{OSP}^2} \quad \text{and} \quad n_t \propto \frac{L^{6/7} n_u^3 R_{OSP}^2}{q_{||u}^{8/7}(1-f_{power-loss})^2}$$

(1)

where $q_{||u}$ [W/m^2] is the parallel power flux density entering the divertor, $f_{power-loss}$ is the fraction of $q_{||u}$ dissipated by radiation and charge exchange in the SOL and divertor, n_u is the upstream density (at the outside midplane separatrix), and R_{OSP} is the radius at the outer strike point. It can be helpful to think of $q_{||u}$ and n_u as being part of the "problem" while $f_{power-loss}$, L and R_{OSP} are part of the "solution." The values of $q_{||u}$ and n_u are imposed by core and pedestal plasma considerations and can't be altered by anything done beyond the separatrix.

For a specified level of power entering into the SOL, P_{SOL}: $q_{||u} = P_{SOL}/(4\pi R_{OMP}(B_\theta/B)_{OMP}\lambda_{q,OMP})$. Recent results from a multi-tokamak attached divertor H-mode database[14] give $\lambda_{q,OMP}[m] \approx 0.0008/B_{\theta,OMP}[T]$ and thus $q_{||u} \propto P_{SOL}B/R$. Future high-power tokamaks will have much higher P_{SOL} values than present tokamaks and also somewhat higher B/R values than most present tokamaks, C-Mod being an exception. For an ITER-like example of P_{SOL} = 100 MW and $\lambda_{q,OMP}$ = 1 mm, then $q_{u||}$ ~ 5 GW/m^2, *which is significantly higher (~ x5) than occurs in present tokamaks*. Reducing P_{SOL} by increasing core radiation has limits; experiments find that P_{SOL} must be maintained at or above the L-H power threshold in order to obtain good core plasma confinement[15].

How much volumetric power loss is required? We have $q_{||t} = (1 - f_{power-loss}) = q_{\perp,t}^{plasma} / \sin\theta_\perp$ thus:

$$f_{power-loss}\Big|^{minimum} = 1 - 5/q_{\|u}\left[MW/m^2\right]\sin\theta_\perp$$

(2)

Example: ITER-like $q_{\|u}$ = 5 GW/m^2, $\theta_\perp$ = 2.5^o, and $q_{\perp,t}^{plasma}$ = 5 MW/m^2. Then for a detached (dissipative) divertor option characterized, say, by T_t = 1 (5) eV and n_t ~ 2.5 (0.5) x10^{21} m^{-3}, see Fig. IV-2, one obtains from eqn. (2) that $f_{power\ \ loss}\Big|^{minimum}$ = 0.98 for either the detached or dissipative divertor option. Employing highly aligned, monolithic divertor structures, facilitated perhaps by demountable toroidal field coils, could allow $\theta_\perp \sim 1^o$, which would reduce the dissipation requirements to $f_{power\ \ loss}\Big|^{minimum}$ = 0.94. However, pushing to DEMO with 3-5 times the power flux as ITER, the situation becomes even more extreme, raising the divertor power dissipation level to essentially 100 percent. In addition to power dissipation, significant plasma pressure loss along magnetic field lines (momentum loss) would also be involved[3] – an effect that is not included in the analysis presented here.

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Chapter V

Priority Research Direction ‘C’ – First Wall Solutions and Actuators

V. Priority Research Direction 'C' – First Wall Solutions and Actuators

<u>PRD-C:</u> Understand, develop and demonstrate innovative boundary plasma solutions for main chamber wall components, including tools needed for controllable sustained operation, sufficient for extrapolation to steady-state reactor application

A relatively cool *boundary plasma* region surrounds the hot fusion core plasma in a tokamak and makes contact with the *main chamber walls*. Recent advances in understanding this region have emphasized its importance, both in terms of optimizing performance of the burning core of a reactor and preventing deleterious interactions with hardware components located on the walls. These components include sophisticated radio frequency wave actuators for plasma heating and current drive. Understanding the mechanisms that control the transport of plasma and eroded wall materials back and forth across the boundary region and devising possible means to control them is critically important. Targeted R&D is therefore needed in a number of areas. Physics-based understanding and quantitative models for both bulk plasma and impurity transport in the presence of intermittent turbulent structures are needed to assess the impact of plasma fluxes on the vessel walls and the fate of eroded materials. Similarly, RF-specific effects in the boundary region must be understood in order to minimize or eliminate deleterious interactions. Discovering how these processes couple and influence the core plasma, and learning how to control them in a reactor environment with new innovations, presents an exciting scientific challenge and a new frontier in fusion plasma physics.

V. 1. Additional Background

Particles, momentum and energy from the core and pedestal region cross the separatrix and emerge on the open field lines of the SOL. In the "near SOL," a narrow (mm to cm) layer adjacent to the separatrix, hot plasma is swept into the divertor by rapid parallel flow. However, it has long been recognized that a significant fraction of plasma undergoes cross-field transport that is strong enough to carry it radially into the "far SOL," the region between the near SOL and the main chamber walls. This far SOL plasma sometimes impinges directly on the main chamber PFCs.

PMI on main-chamber walls, including active components such as RF antennas/launchers and control actuators, are recognized as potentially serious issues for a fusion reactor and are the topic of this PRD. These issues have not been widely studied. Their operational impact in present day short pulse, carbon or boron-coated machines, has not been critical, but it has now become clear that the impact on long-pulse devices is a critical issue. Most tokamak boundary research to date has focused on the divertor, rather than the main wall. Our incomplete knowledge of main chamber PMI represents a significant gap in fusion research.

Transport in the main chamber SOL region (both outward and inward) couples the main core plasma to the wall, and is expected to have a profound effect on the performance of a fusion reactor. These integration issues are discussed in PRD E. Transport of plasma into the far SOL is thought to be largely due to turbulent processes, including intermittent convective transport by what are called "blob" and ELM filaments[1]. Plasma impacting the wall recycles as neutrals, similar to plasma recycling from divertor PFCs. Under some conditions, this main chamber recycling can dominate the particle balance in the machine[2]. Erosion of the first wall by resulting charge-exchange neutrals may play an important role in impurity production and first-wall lifetime. The presence of neutrals also necessitates complex modeling of the far SOL region, taking into account ionic species, neutral transport, charge exchange and radiation processes, in addition to plasma turbulence and transport. Understanding how these processes couple and influence the core plasma presents a major scientific challenge, and indeed a new frontier in fusion plasma physics.

The SOL plasma interacts not only with passive main chamber walls, but also with RF and other active components that are necessary for sustainment and control. These include ICRF waves for bulk heating and flow drive[3], LH waves for off-axis current drive, and RF waves that can be utilized for other applications, such as stabilization of additional core instabilities ("sawteeth" and neoclassical tearing modes), impurity control and perhaps even direct turbulence control. However, the presence of large amplitude electromagnetic waves generates large RF electric fields near plasma surfaces, which can lead to RF sheaths[4], enhanced impurity sputtering and localized power deposition when ICRF and LH frequency waves are deployed[5]. Plasma interactions such as sputtering and parasitic power dissipation on RF launcher and nearby surfaces can reduce component lifetime, even catastrophically damage components, and reduce global plasma performance through impurity contamination of the core plasma[6]. Finally, PMI and SOL plasma interactions may also be an issue for other actuators, such as internal magnetic coils used to control ELM transients, and coating or erosion of ECH mirrors, used for heating/current drive and control of MHD instabilities.

The erosion of material from active and passive wall components by charge-exchange neutrals or ions presents a wall lifetime issue that must be addressed[7]. In addition, eroded material, especially high-Z impurities, must be prevented from penetrating and residing in the core plasma. Thus, the counterpart to understanding main species plasma transport from the separatrix to the wall, is the transport of impurities in the reverse direction.

An important point is that materials that have eroded from the first wall can participate in repeated sputtering, ionization, transport (parallel and cross-field) and re-deposition resulting in long range migration and the build-up of complex layers of materials, sometime called slag. These slag deposits could potentially build up in locations that are far removed from the erosion sites, including the divertor, and critical recessed locations. The fusion program has little experience with such deposits because the rate of slag formation in present day low-duty-cycle machines is very small. In long-pulse, or steady-state devices, slag accumulation will have to be managed to prevent deleterious effects on device operation. These could include unwanted deposits on the divertor target plates, unacceptable dust levels and disruptions due to breakaway pieces of slag falling into the core plasma[8].

All of the preceding issues are expected to be exacerbated in long-pulse reactor-grade machines. On the other hand, exciting ideas have been proposed for controlling main-chamber PMI on wall structures and on active plasma components. Boronization is found to be essential to control high-Z impurities on AUG and C-Mod. Unfortunately, most such present day wall conditioning methods do not extrapolate readily to a reactor environment. This has led to a number of new ideas, e.g. replaceable reactor-compatible low-Z coatings, liquid metal walls, and reduced PMI contact on the low-field-side wall components — the latter idea via relocation of RF antenna/launch structures to the high-field side.

Thus, challenging scientific questions arise in the areas of far SOL transport, SOL interaction with RF and other active components, impurity erosion, impurity transport into the core plasma and long-range migration. Furthermore, integrated performance issues associated with extrapolation to reactor regimes need to be addressed. These plasma physics questions, which are distinct from but obviously related to PFC material science issues, are addressed in the following subsections.

V. 2. Main Scientific Questions

What governs far SOL transport, including blobs and transients, and main chamber recycling, and can we predict it quantitatively?

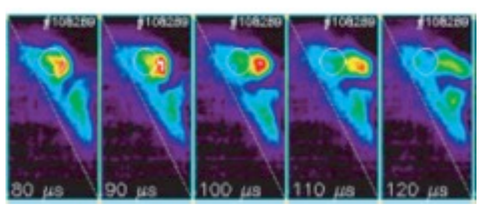

Figure V-1: Cross-sectional view in the R-Z plane of blob-filament propagation in NSTX[9].

Challenges in understanding the SOL have received attention in previous documents, including the 2014 Fusion Energy Sciences Advisory Committee Report on Strategic Planning and the 2009 ReNeW report. The 2014 FESAC Report notes that theory and simulation applied to the edge region to provide predictive understanding for ITER and beyond is not as developed as for other plasma phenomena. Understanding the physics of the SOL and developing solutions that control how this flux impinges on material surfaces is a high priority. The complexity of far SOL plasmas includes the interacting processes among ions, atoms, and molecules, their fluxes to/from the material surfaces, oblique incidence sheaths at the main chamber wall and RF launchers, sputtering in response to the incident plasma, emitted impurity transport, and the response of the PFCs to ELMs, disruptions, vertical displacement events (VDEs), runaway electrons and other possible high-power plasma transients. Fundamental physical mechanisms, such as intermittent convective transport by filamentary structures (blobs and ELMs), charge exchange, ionization, radiation and parallel transport have been identified. An example of blob transport observed in NSTX[9] is shown in Fig. V-1. However, understanding sufficient for calculating heat and particle fluxes on PFCs at the main chamber wall, or transport of impurities generated there, is presently unavailable. Integrating increased knowledge of the fundamental processes into a predictive understanding of the SOL presents a major and worthwhile scientific goal. The main scientific problem is to understand the nonlinear turbulence in the far scrape-off layer.

Progress has been made on the propagation of, and resulting cross-field transport by, intermittent blob-filaments in the SOL, including simulations with electromagnetic and 3D effects. Nevertheless, many challenges remain for achieving a quantitative description of the generation of these filamentary structures, presumed to be through turbulence of the edge plasma spreading into the SOL, perhaps in combination with SOL-driven instability. These challenges encompass understanding highly nonlinear coupled multiple physical processes interacting at multiple spatial and temporal scales. Specific elements include:

- The role of plasma shaping, magnetic geometry, and wall geometry;
- Large fluctuation levels due to eddy sizes that are comparable to ambient gradient scale lengths;
- Sonic flows and oblique incidence sheaths;
- Kinetic effects on both electrons and ions;
- The role of particle momentum and energy sources and sinks, from neutral and atomic physics such as friction, ionization and radiation; and importantly,
- The role of main chamber wall recycling in single and mixed-material, solid and liquid, and cold and hot materials.

Progress has been hindered in part by a lack of good diagnostic coverage and in general by a lack of resource allocation to these increasingly important topics.

Blob-filament transport in the SOL is thought to be important in L-mode and in the inter-ELM periods of H-mode discharges. These blob structures can convect hot ions to the wall resulting in enhanced sputtering. Closely related to blobs, but carrying even hotter and denser plasmas structures, are transient events such as H-mode ELMs. While it is generally believed that large ELMs will be unacceptable in a reactor due to divertor heat flux considerations, small ELMs may still be tolerable (depending on divertor physics discussed in PRD B) and even desirable to flush impurities from core plasmas; if so, their effect on the main chamber wall must be managed.

In summary, there are several scientific challenges in understanding the far SOL that make it a more complex problem than core transport. These include: (i) the intermittent nature of plasma processes in the far SOL (both spatial and temporal); (ii) strong plasma physics nonlinearities associated with order unity fluctuations; (iii) strong nonlinearities associated with atomic physics and wall processes. These challenges make far SOL physics a rich area for plasma physics discovery that is also critical for the mission of fusion research.

How do RF and other active main-chamber components interact with the SOL plasma, and how do we manage those interactions

The 2009 ReNeW report highlighted important scientific questions for RF antennas and launchers. These included:

(i) How can the predictive capability of plasma edge models, including material interaction, be enhanced?

(ii) Can these enhanced models incorporate the formation of ICRF sheaths produced by RF waves transiting between the antennas and absorption in the core plasma?

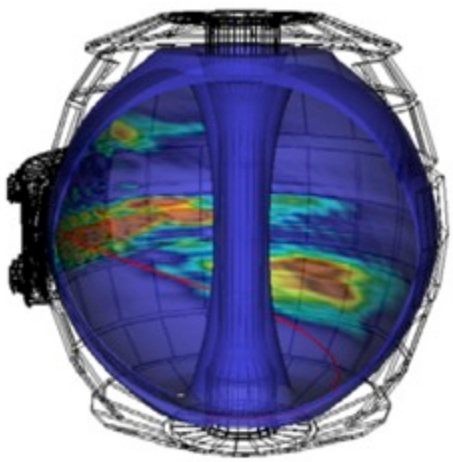

Figure V-2: AORSA simulation of absorbed ICRF power in the SOL for an NSTX discharge. The antenna is on the left of the torus[9].

RF launchers constitute part of the main chamber plasma-facing surface, and are the active components most often in close proximity to the LCFS. PMI degrades ICRF Faraday shields and LH grills, results in impurity influx, and affects power-handling limits. It has become increasingly clear that, in addition to the direct interaction between the SOL plasma and the RF launchers for ICRF and LH waves, the properties of the SOL plasma affect the propagation and absorption of the RF energy by the main core plasma. Also, RF waves modify the SOL through local (e.g. sheath and fast electron generation) and far-field (e.g. parametric decay and parasitic surface wave) effects. Since ReNeW, RF codes have been extended to calculate wave propagation and absorption in the SOL plasma[9] (see e.g. Fig. V-2), highlighting the importance of the SOL for understanding and predicting RF performance in the ICRF and LH regimes. Detailed measurements of the SOL characteristics are required to predict the location of possible RF interactions with the plasma-facing materials. Progress has been made in formulating the attributes of RF sheaths and incorporating them in RF and edge-plasma codes as an appropriate boundary condition. Additional progress is needed in predicting the effect of RF energy on the SOL parameters.

New concepts that have been put forward to minimize the interaction between the plasma and launcher components include inboard launch, field alignment through rotation of all antenna components, and helicon current drive.

Inboard launch takes advantage of the very low density, quiescent high field side SOL in double null operation. While the density is higher in single null, the plasma remains quiescent. For ICRF and LH, the absence of density blobs would permit the launchers to be closer to the separatrix (required for adequate ICRF coupling) with reduced plasma interaction. Furthermore, any fast electron filaments produced by edge RF interactions would convect away from the launchers. The helicon concept allows for a travelling wave launcher that requires only light coupling to the plasma, allowing it to be positioned in the far SOL. Success for this concept will rely on a detailed understanding of wave propagation in the SOL, however. Field-aligned antennas take advantage of the symmetry of the plasma dielectric to minimize undesired RF field components. On Alcator C-Mod the use of a field-aligned antenna was found to essentially eliminate impurities originating directly from the antenna structure.

Both Alcator and NSTX experiments have highlighted the importance of RF generated impurities away from the launcher structure generated by RF fields or changes in the electrical potential of the scrape-off plasma. Understanding and control of these effects remains a challenge.

In summary, RF-enhanced impurity production, particularly in ICRF regime, remains a challenge. The exact mechanisms by which the RF changes the SOL plasma, including the part that touches material surfaces has not been fully understood.

A second scientific challenge is associated with the way that ICRF and LH launchers interact with and modify nearby plasma. For example, while the field aligned ICRF antenna eliminates impurities directly from the antenna, it does not remove the RF-driven potential in the SOL and the enhanced flux of impurities away from the antenna. This may also be related to the observation on NSTX of enhanced interactions at material surfaces far from the antenna that are along field lines that pass in front of the launcher.

The preceding has emphasized SOL interactions with RF launchers and, more generally, the effects of RF in the boundary plasma. Arguably, understanding and mitigating such interactions are the most obvious and pressing needs. However, the SOL plasma may also have specific interactions with non-RF actuators. For example, mirrors employed to control the directionality of ECH could be both coated or eroded by the SOL plasma, severely limiting their performance lifetime. Similar concerns apply to diagnostic hardware. As a final example, lobes associated with ELM control coils, and 3D magnetic perturbations in the SOL in general, must be better diagnosed and understood. Emphasis to date has been on penetration of such effects into the pedestal region; however, their impact on main chamber SOL-relevant issues may be just as important.

What governs impurity erosion, transport into the core plasma, and long-range migration, and how do we control them?

The interaction between the SOL and the main-chamber walls and components, while not as intense as in the divertor, nevertheless involves scientific questions that are essential to the success of fusion energy.

First, the rate at which the wall material is eroded and thereby becomes an impurity available for entry into the plasma is uncertain. For the most part, this uncertainty arises mostly not from lack of understanding of the mechanisms of erosion such as sputtering, though some uncertainty arises in the specific fusion environment, for example high material temperatures and plasma surface modification. What dominates the uncertainty more are the characteristics of the impinging plasma and particles. Our limited understanding of the far-SOL plasma transport means the temperature and density at the wall are uncertain. Similarly our understanding and ability to quantify the effects of RF in enhancing sheath interactions with the wall and main-chamber components is still somewhat rudimentary. Moreover, the wall sputtering may be dominated not so much by direct plasma contact as by the effects of fairly energetic neutrals produced by charge-exchange at the plasma edge, and possibly by extremely energetic lost ions, for example fusion products and non-thermal tails arising from auxiliary heating.

Second, even if we were able to reliably predict the erosion and hence surface impurity production rate, major challenges remain in understanding the mechanisms by which these impurities subsequently migrate through the SOL, either reaching the core plasma or being scraped off and re-deposited elsewhere, for example in the divertor. There is substantial experimental evidence indicating that the core impurity influx in tokamaks, especially of high-Z metals, arises mostly from the main chamber rather than from the divertor. Moreover measurements show that the extent to which the impurities enter the core plasma (compromising plasma performance) varies greatly depending upon the poloidal position from which they arise. The understanding of and ability to predict the impurity penetration is handicapped the need for comprehensive spatially resolved experimental information in the SOL, and for the ability to perform controlled influx experiments from different locations. We also require thorough theoretical understanding, not yet available, of the entire SOL transport including plasma drifts, turbulence, and geometric effects.

Third, material eroded from the main chamber eventually finds its way back to surfaces where it re-deposits. Its final re-deposition position may be far from where it started; this is what is referred to as material "migration." Migrated material is generally a worrisome liability. It tends to build up in thin flaky layers that are poorly attached to the substrate. If the build-up is in the main chamber, there is a risk of flakes falling into and disrupting the plasma. Therefore, it is important to understand the processes that govern material migration in the main-chamber SOL, and hence the locations of re-deposition. Observing the build-up of thick deposits of migrated material requires long total plasma durations at relevant parameters. However, experimental investigations that track the migration of trace levels of impurities present important opportunities also in shorter-pulse devices.

At least two innovative approaches to main chamber impurity control by wall surface modification have been put forward. One is the use of liquid PFCs. While this approach has potential for transformative solutions of the erosion challenges, it faces its own tough problems. For example, tritium absorption in the large exposed surface poses a concern for reactors, especially for low-recycling PFCs. Also practical control of the liquid location and flow is a daunting engineering challenge. Results with capillary porous materials and thin slow-flowing films are encouraging but inconclusive. Another idea is the use of renewable solid surface coatings, perhaps composed of low-Z materials whose impact as core impurities is less severe than heavy metals. Such coatings still face challenges of material migration and removal, but might alleviate core impurity problems.

How can our understandings and solutions be extrapolated to reactor regimes?

How can we characterize and mitigate erosion/re-deposition, impurity influx, and tritium retention with tungsten walls at a reactor-relevant operating temperature? For a steady-state reactor, erosion and re-deposition of PFCs will continuously remake the wall and divertor surfaces. This issue is discussed further in connection with PRD D. Erosion, material migration, and re-deposition are not completely understood, and unlike a reactor, most present day tokamaks do not operate at sufficient edge density to promptly ionize and redeposit eroded material. Erosion can be mitigated by techniques that produce high edge radiation (impurity seeding), but not eliminated. Enhanced tritium retention, as well as modifications to thermal and mechanical properties of redeposited material, must be understood to adequately assess reactor PFC performance. Although results from JET with the ITER-like wall indicate sufficiently low tritium retention for ITER, the ILW experiments in JET differ significantly from reactor conditions, in that the duty factor for JET is orders of magnitude lower than that needed for a reactor, while the wall operating temperature is well below reactor needs.

Can we develop approaches to control the flow and mitigate migration of liquid metal PFCs under reactor conditions? Liquid metal PFCs are also subject to erosion and re-deposition, but with liquids there is no expected modification of mechanical or thermal properties of the surface itself. For hydrogen-retaining liquid metals (lithium), migration of eroded material, or ejection of liquid during MHD activity, into regions of the chamber or divertor which are not drained may lead to tritium accumulation issues. Inadequate control of flowing liquid metal can result in ejection of unacceptable amounts into the plasma, and a disruption. Localized heating can lead to excessive evaporative influx.

What are the impacts on recycling, impurity production, and PMI produced by operating tungsten PFCs above the ductile-to-brittle transition temperature (DBTT)? Most tokamaks now operate with fully recycling walls and divertor targets. For a reactor, the primary candidate material for the PFCs that comprise the wall and divertor is tungsten, operated above the DBTT at 600 – 800 °C. Although it is clear that hot walls are necessary to anneal neutron damage and avoid the mechanical issues associated with brittle materials, no tokamak wall or divertor has been operated in the requisite temperature range.

Can we understand the mechanism by which lithium PFCs produce enhanced *confinement, and develop effective tritium removal techniques for reactor applications?* An alternative to solid tungsten is liquid lithium. Other alternatives to tungsten and lithium include liquid metals such as tin, or continuously renewed carbon. The use of lithium PFCs has resulted in very low levels of core impurity accumulation, and also in enhanced confinement compared to ITER ELMy H-mode scaling. Lithium retains hydrogen to varying degrees depending on temperature, and has been demonstrated to reduce recycling. In detail, however, it is not clear that lithium wall conditioning or the use of lithium coated PFCs achieves better performance through a common mechanism, such as reduced recycling. As previously mentioned, lithium retention of tritium is also a concern.

Can we redesign or relocate RF launchers to effectively heat and drive current, while avoiding impurity generation and undesirable modifications to the SOL under reactor-relevant conditions? As discussed previously, a deeper understanding of RF interactions with the SOL in present day and future tokamaks is needed to develop solutions that can be expected to extrapolate to reactor conditions. Field-aligned ICRF antenna structures and high-field-side launch are proposed techniques that may significantly mitigate deleterious interactions.

V. 3. Action plan

V. 3. 1. Advance physics understanding of the main SOL through improved diagnostics, theory & modeling

A set of recommended actions to address gaps in our physics understanding of main chamber PMI issues is now discussed: (i) improved measurements in the upstream far SOL, (ii) global characterization of potentials and flows, and (iii) next-generation computational tools for SOL theory including RF processes.

(i) Make high spatial and temporal resolution upstream measurements in the far SOL

Order unity fluctuations in the far SOL are intermittent in both space and time. Filamentary structures (elongated along the magnetic field lines) and with cm-scale dimensions across the field lines have been inferred from edge diagnostics. Typical lifetimes are in the range of a few to a few tens of microseconds. While existing diagnostics have provided much valuable information, they leave significant gaps in our understanding of the far SOL. High spatial and temporal resolution measurements of basic plasma quantities including density, electron and ion temperature, and electrostatic and electromagnetic fluctuations are badly needed throughout the SOL and in the vicinity of the main chamber wall. Additional characterization of the plasma impurity content, main plasma and impurity transport fluxes and the neutral particle composition is also needed, though likely only at reduced resolution. Obtaining these fundamental measurements of the far SOL plasma is essential to progress in intuitive understanding, reduced model development and ultimately validation of more comprehensive numerical modeling tools. Plasma flux and energetics data will also support PRDs in wall-oriented and materials-oriented research.

(ii) Perform global characterization of potential & flow (intrinsic and RF-induced)
The importance of plasma potentials and flows in the SOL has become increasingly clear, both with and without the presence of RF waves. The plasma potential near surfaces controls available energies for sputtering. Thus, plasma potential characterization is important in the vicinity of the main chamber walls, limiters and other structures in the SOL, or in the presence of explicit 3D perturbations (such as from RF launchers or magnetic perturbation coils). The generation of RF-induced potentials is well-documented but a characterization of their 3D structure remains incomplete and poorly understood. These 3D structures can drive particle transport across the magnetic field that can affect SOL profiles (important for wave launching) and transport of RF impurity fluxes into the core. Measurement needs for these RF-induced structures are discussed in the following under the heading "Exploit and upgrade present experiments."

Characterization of intrinsic potentials and flows throughout the SOL is an important key to further advances. Global convective flow patterns impact particle balance, inner/outer divertor target interactions, and possibly global plasma confinement through SOL interaction with edge plasma rotation and E×B shear.

(iii) Develop divertor/SOL/RF theory and next-generation computational tools
Upgraded diagnostic capabilities will provide a solid basis for the development and validation of new theoretical models and a next generation of computation tools for understanding both individual and integrated aspects of divertor, SOL and RF physics. In this PRD we emphasize far SOL physics models, including wave propagation and RF sheath mitigation studies.

New theoretical models and computational tools are needed to address the diverse and coupled physics of SOL transport, neutral and atomic physics, and plasma material interactions. Needs include conceptual and reduced models, multi-physics components for comprehensive numerical simulations, and additional development of kinetic and integrated models. Development is needed to address individual and coupled issues discussed in the preceding section on, "Far SOL transport, including blobs and transients, and main chamber recycling."

Extension or modification of these models and separate or integrated models to address RF-specific effects in the SOL is also called for. Relevant physics includes RF launcher wave-coupling physics self-consistent with (RF-modified) SOL plasma parameters, RF sheaths, nonlinear plasma effects such as ponderomotive and parametric decay phenomena, parasitic power flow, and impurity generation and transport. Three-dimensional studies including realistic antenna and wall geometry and core-edge coupling will be required for quantitative predictive models.

Computational efforts, in both the intrinsic and RF cases, will benefit from an improved understanding of PMI effects on the surface itself, as discussed in connection with PRD D. For example, linear device measurements of main chamber sputtering coefficients and erosion rates would provide important information for tokamak modeling.

V. 3. 2. Exploit & upgrade existing experiments to advance main chamber SOL research

To the extent possible, the new ideas discussed here and elsewhere should be tested in existing U.S. experiments, with appropriate upgrades, focusing on developing fundamental understandings of the physical processes involved. With limited overall resources, this will require increased emphasis on boundary and divertor physics within the overall U.S. effort. We recommend that existing experiments be exploited and upgraded to specifically advance main chamber SOL research as follows:

(i) Enhanced runtime and personnel resources should be made available for main-chamber SOL and RF-SOL physics;
(ii) Enhanced diagnostics should be deployed;
(iii) The community should explore PFC material options including solid and liquid, high-Z and low-Z, and advanced designs of RF launchers.

Point (i) arises from the recent and growing understanding of the importance of main-chamber SOL physics as described in this document. Concerning (ii), some essential enhanced diagnostics are discussed in the preceding subsection. Questions that could be addressed in existing experiments include: quantifying the effects of recycling from the main chamber wall on the SOL and pedestal (especially using lithium); understanding how density pedestal structure relates to main chamber recycling; and utilizing 3D diagnostics like marker tiles to understand where far-field RF interactions with the wall actually occur, and more generally to understand other 3D effects.

For point (iii), PFC material options must address the issues of net erosion and slag management. The former involves replacement of large quantities of eroded wall materials (estimated to be hundreds to thousands of kg/yr) and suggests in situ replacement of solids or flow-through liquids. Both high-Z and low-Z wall coatings have been proposed. With the exception of Alcator C-Mod, recent tokamak operation has mostly been with carbon first walls. The changeover of AUG from carbon to tungsten and the changeover of JET from carbon to an ITER-like wall (beryllium first wall, tungsten divertor) has revealed challenges for obtaining good core plasma conditions (i.e., sufficiently high energy confinement time with low impurity contamination levels). Impurity seeding plus active control of core impurity accumulation, such as on-axis RF power deposition, is found necessary to obtain dissipative divertor operation and to mitigate high-Z impurity concentrations in the core. An acceptable choice of main chamber wall materials for devices beyond ITER remains a critical issue requiring extensive exploration on smaller and more flexible devices.

Detailed 3D measurements of the SOL plasma (density, temperature, potential), including the response to RF and the strength of RF wave fields, are required to make progress on understanding SOL physics and the interaction between RF fields, other actuators and the SOL. Increased diagnostic coverage on existing machines will be required to obtain this information. Given this information, improved RF codes that include the SOL, sheath boundary physics and realistic geometry (important for full characterization of the sheath) should allow for optimization of RF launchers. Detailed experiments to measure RF sheath driven erosion and comparison with modeling codes would allow the coupling to edge fluid models. Similarly, improved diagnostics and analysis are required to understand and mitigate unwanted interactions with other active components as discussed previously.

Installation of inboard RF launchers on existing devices is extremely difficult because of space and access considerations. In any case improved characterization of the high-field-side SOL in double null, and particularly what proximity to exact double null is required to produce the quiescent parameters desired, should be pursued on existing devices.

Existing U.S. facilities are uniquely positioned to address main-chamber SOL issues, developing basic physics understandings and testing some new ideas. Modest investments in runtime, diagnostics and upgrades would facilitate world-leading research in this area.

Alcator C-Mod

C-Mod's high-Z plasma facing components, high power density and exclusive use of RF for auxiliary heating and current drive provide a unique opportunity to explore RF wave physics (ICRF and LH) at reactor-level densities and magnetic fields. A recently developed field-aligned ICRF antenna has shown a remarkable reduction in RF-induced impurity sources at antenna surfaces, which should be explored. C-Mod's excellent SOL diagnostic set, including scanning probes on high-field-side SOL, allow direct characterization of the high-field-side SOL and measurements of RF waves at this location. A new in-situ, 1 MeV ion beam PMI diagnostic (AIMS) provides between-discharge characterization of wall conditions, which is essential for unfolding the physics of wall-conditioning and material migration. With current plans for C-Mod closure in 2016, it is imperative for the facility to be fully utilized to address main-chamber SOL issues in its remaining operations. Should C-Mod operations continue, valuable upgrades/experiments could include:

- Second field-aligned ICRF antenna (built but not installed)
- Passive tests of reduced PMI at inner-wall launchers
- Reduced power high-field-side launch LH
- Effect of lithium wall conditioning with high-Z PFCs

NSTX-U

The unique facility features of NSTX-Upgrade, namely, high power up to 12 MW NBI and up to 6 MW 30-MHz HHFW and a planned 1 MW, 28 GHz ECH/EBW system, enable RF and SOL physics studies with innovative PFC approaches. In the near-term, graphite PFCs will be tested with lithium coatings to understand the impact of low-recycling wall conditions on SOL and divertor physics. In the longer-term, high-Z PFCs (molybdenum and/or tungsten) are planned for the divertor and first wall. The high-Z PFCs would enable testing high-Z plasma facing components with static liquid lithium films, and eventually with flowing liquid lithium films, as well as general high-Z erosion and transport studies. The planned research will be supported by a SOL diagnostic suite that includes a beam emission spectroscopy system and gas puff imaging; Langmuir probes and fast cameras for edge/SOL transport and turbulence studies; a unique material analysis particle probe for in-situ surface science studies, and a new impurity laser blow-off system for impurity edge/SOL transport studies and benchmarking of edge turbulence codes.

DIII-D

The DIII-D boundary/PMI program aims at developing and testing solutions for coupling high performance core discharges with plasma-facing materials (PFMs) sufficient to provide the heat removal and erosion control necessary for next-step devices. DIII-D's excellent SOL diagnostic set, including scanning probes on the high-field-side SOL and DiMES materials evaluation system, allows detailed characterization of SOL and PMI physics. The main research thrusts focus on 1) quantifying the impact of high-Z PFMs on core plasma, and erosion/re-deposition and migration within a mixed-material PMI environment; 2) exploring PFMs at high temperature (~700 °C+) in the divertor and main chamber; 3) validating advanced materials relevant to fusion reactors, including both high-Z materials and low-Z coatings, in collaboration with the linear fusion devices and broad materials development community.

V. 3. 3. Participate in overseas experiments to advance main chamber SOL research

We recommend that the U.S. participate in overseas experiments in the areas described in this chapter. Potential benefits from international collaboration on far SOL transport include gaining access to international magnetic fusion capabilities not available in the U.S., in particular access to steady-state research in superconducting advanced tokamaks and stellarators. Some examples are:

- Long-pulse operation (e.g., EAST, KSTAR, JT60-SA, W7-X)
- Mix of first-wall materials (e.g. JET, ITER, EAST)
- High-field-side RF launch (e.g. WEST)
- Large size (ITER, JET, JT60-SA)
- Liquid PFCs (EAST)
- Specific areas of topical focus (e.g., far SOL transport in ASDEX-Upgrade)

U.S. fusion interests in international collaborations can be well served by forming focused U.S. physics teams that combine theory, modeling/simulations and experiments/diagnostics, and are charged with pursuing new initiatives, assuming leadership and developing fundamental understanding of far SOL transport relevant to unique international facilities. In some areas the United States can contribute needed expertise and gain additional knowledge and experimental data that existing U.S. facilities can't provide, such as knowledge and data on single and mixed-materials on the main chamber wall, new techniques for fueling and ELM control, and on RF and other control actuators.

An important goal for the world fusion program that will advance U.S. interests in main chamber research is the development of an international physics database and predictive capabilities for far SOL transport, including blobs, transients, and main-chamber recycling. Continued U.S. participation in ITPA could further this effort.

V. 3. 4. Develop a U.S.-led dedicated facility: a divertor test tokamak

Finally, there is a broad consensus within the U.S. boundary and divertor plasma physics community that a U.S.-led facility, with the flexibility to implement a wide range of advanced divertor magnetic and gas-dynamic configurations, as well as target and wall materials, should be pursued. We call this facility a Divertor Test Tokamak (DTT). The mission of this SOL-physics-targeted device would be to test and demonstrate divertor configurations and materials, and main-chamber wall/actuator solutions at power densities and SOL plasma conditions that are prototypical for a reactor.

Divertor-specific issues and needs that could be met by a DTT are discussed extensively in connection with PRD B. Importantly, a DTT would also provide an exciting opportunity to explore and develop main-chamber boundary-plasma solutions to many of the science issues relevant here to PRD C. The main tasks in this area are as follows:

(i) Explore main-chamber PFC material options
(ii) Explore innovative RF heating and current drive techniques compatible with the SOL
(iii) Develop main-chamber PFC solutions and RF actuators that would directly extrapolate to steady-state fusion power systems

The DTT should have the flexibility to implement a wide range of advanced wall materials for testing at power density and plasma conditions which directly extrapolate to a reactor. The DTT could also incorporate potential game-changing ideas for low-PMI, reactor-relevant RF current drive and heating actuators, such as high-field-side launch.

Valuable experience in main-chamber PMI physics will be gained from present-day experiments and ITER. Nevertheless, upgraded diagnostics in existing machines, and improved understanding through enhanced resources for data-taking, analysis, theory and modeling, cannot by themselves establish the knowledge base required to design a fusion reactor. Innovative divertor configurations and first wall components and materials must be explored experimentally in relevant regimes, and we believe this will be feasible and cost-effective only in a DTT.

A DTT would leverage unique U.S. scientific and technological expertise in a number of areas (e.g. advanced divertors, liquid metals, RF systems, and high-field magnet technology); the device would rapidly advance progress in divertor and main chamber SOL research, and give the United States a world-leadership position in this most important scientific topic. *We recommend establishing within the FES strategic* *plan a national working group to examine design options for a DTT facility.*

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Chapter VI

Priority Research Direction ‘D’ – Evolving Plasma Facing Surfaces

VI. Priority Research Direction 'D' – Evolving Plasma Facing Surfaces

<u>PRD-D:</u> Understand the science of evolving, reconstituted materials at reactor-relevant plasma conditions and how novel materials and manufacturing methods enable improved plasma performance

Plasma facing surfaces experience an evolving layer of material that is continuously re-constituted via erosion and re-deposition, leading to dynamic surface properties and plasma-surface interactions in fusion devices. The entailed actions include: 1) to understand the science of PMI on these dynamic surfaces at reactor-relevant conditions, and decipher the practical implications on heat and particle limits, and 2) to develop radiation tolerant materials that maintain material performance despite plasma (neutrons, T, He) induced material evolution, using both advanced manufacturing and modeling for tailoring of solid surfaces and evaluation of self-healing (liquid) structures.

VI. 1. Additional Background and Key Science Questions

Although progress has been made in the last half-decade in establishing an understanding of PMI, there remain critical knowledge gaps, particularly when it comes to predicting the behavior at the plasma-material interface under reactor-relevant plasma conditions in a future plasma-burning neutron-dominated environment. The plasma-material interface is considered to be one of the key scientific gaps in the realization of nuclear fusion power. At this interface, high particle and heat flux from the fusion plasma can limit the material's lifetime and reliability and therefore hinder operation of the fusion device. This region is critical to operation of a nuclear fusion reactor since material can be emitted both atomistically (e.g. through evaporation, sputtering, etc.) and/or macroscopically (i.e. during transients events, such as disruptions or edge localized modes). The environmental conditions at the plasma-material interface of a future nuclear fusion reactor interacting will be extreme. The incident plasma will carry heat fluxes of the order of 100's of MWm^{-2} and particle fluxes that can average 10^{24} $m^{-2}s^{-1}$. The fusion reactor wall would need to operate at high temperatures near 800 C and the incident energy of particles will vary from a few eV ions to MeV neutrons. Another challenge is the management of damage over the course of time. Operating at reactor-relevant conditions means the wall material would need to perform over the course of not just seconds or minutes (i.e. as in most advanced fusion devices today and in the near-future), but from months to years. **Therefore, the plasma-material interface would be a dynamic, evolving, reconstituted region of material that is constantly eroded and re-deposited a million times over, creating conditions that go well beyond our currently limited understanding of materials damage.**

Promising developments in advanced materials and additive manufacturing are providing alternatives to harness self-healing and adaptive materials properties that could make a significant impact in providing radiation-tolerant materials. Liquids as potential PFCs are another promising alternative. However, even with the promise of liquid surfaces, in which the issues of erosion/re-deposition are nearly absent, the long-term behavior of such surfaces and coupling with the edge plasma is not understood, particularly as it relates to power handling and impurity mixing. Advances in new materials for the PMI will require establishing robust testing facilities (i.e. both linear and toroidal) that can appropriately replicate conditions expected in future fusion nuclear reactors. In addition, sophisticated multi-scale in-situ diagnostics would be needed to validate multi-scale PMI modeling coupled to well-diagnosed single-effect science facilities to de-couple complex mechanisms inherent in plasma-based devices.

This PRD emphasizes two primary goals:

- To understand the science of plasma-material interactions on evolving, reconstituted surfaces at reactor-relevant conditions and decipher the practical implications on heat and particle limits, and
- To develop innovative radiation tolerant materials that maintain material performance despite plasma (neutrons, tritium, helium) induced material evolution through advanced manufacturing and modeling including surface modification and self-healing structures.

Building a *predictive* understanding of the evolving, reconstituted plasma-material interface under fusion reactor-relevant conditions

One of the grand challenges in establishing predictive modelling and theoretical capability of PMI is the requirement that such complex and diverse physics, which occur over a wide range of length (sub-nanometers to meters) and time (femtoseconds to years) scales, be addressed *simultaneously*. Predicting PMI behavior also remains a challenge, especially determining how best to couple current PMI computational models with experiments. The issue can be summed up as follows: plasma-surface interaction response codes serve as boundary conditions to erosion/re-deposition codes, which in turn link to edge plasma models, which then couple to core plasma performance codes. The limiting step in this approach largely depends on the sophistication and fidelity of surface response codes. Validating these codes with controlled, well-diagnosed laboratory experiments has been critical to increasing the reliability of the codes, and to aiding understanding of the physical mechanisms at the plasma-material interface. However, as these computational codes have limits, so do the experiments. One critical challenge to effectively validating PMI computational codes is the strong spatio-temporal coupling that exists when plasma interacts with a material's surface. Figure VI-1 below illustrates the spatio-temporal aspects of plasma-surface interaction scales, and the critical gaps between multi-scale modeling and experimental validation. Two axes of time and space depict the plasma-surface interaction (PSI) physical scales. The figure illustrates the complex coupling between the ballistic (i.e. collisional) mechanisms induced by charged particles from the edge plasma, and the diffusional mechanisms that dictate defect dynamics that ultimately determine surface and thermo-mechanical properties.

For example, pump-probe experiments, in which a pulsed particle beam (e.g. laser, ion or electron) induces a physical change to the material in question on a particular time scale can then be probed by another sequenced and synched particle, thus capturing time-dependent changes. The challenge is pulsing with the correct time resolution to appropriately diagnose the desired mechanism. However, not all pulsed measurements can be done at the appropriate spatial scales. Resolution from the order of a few nanometers to measurements that capture continuum properties in the scale of meters can be challenging, especially if coupled to time-dependent techniques. Therefore, understanding the limitations of both modelling and measurements, identifying validation opportunities, and identifying where computational codes can fill the gap for physical scales both in time and space inaccessible to experiments, is vital for progress toward a credible understanding of the PMI.

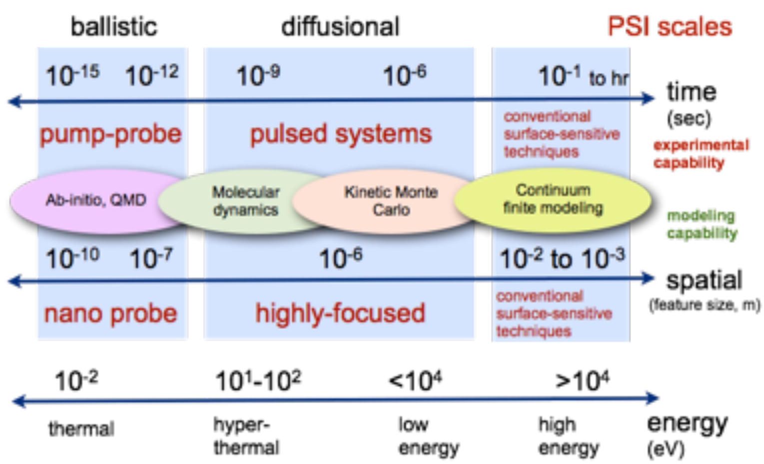

Figure VI-1: Schematic outlining the spatio-temporal physical scales involved in PSI and how experimental and computational tools access the same. For example, experimental tools could probe ballistic mechanisms with pump-probe type diagnosis. These could couple to QMD or MD type simulations tools. A third axis in the bottom depicts the energy scale relevant to PSI that one must address with the interaction of particles and the material surface.

Recent key advances in atomistic computational models and in-situ well-diagnosed *simulated* experiments that replicate conditions found at the fusion PMI are beginning to help unravel the mechanisms that produce plasma-driven modification of candidate materials and coatings, and their effect on plasma performance. However, the strong coupling of the plasma edge and material surface under reactor-relevant conditions limits our ability to extrapolate material performance attributes. Emphasis on the *effect of the emission of material to plasma edge performance* must be extended, and attention given to *how materials properties evolve and respond to the interaction with the plasma, particularly the evolving, reconstituted surfaces that have variable properties over time.*

Establishing an integrated materials design activity for a robust plasma-material interface

A scientific, multi-scale approach is needed, with substantial contributions from both the materials and design communities, to enable the development of analytical methods that will permit design of high-performance, high-reliability fusion reactor PFCs in the context of PMI solutions. To establish a robust design activity for future PFCs for the extreme conditions of nuclear fusion reactors, the appropriate testing facilities and prototypical environments must be available. One of the challenges in controlling the PFC lifetime under reactor conditions is maintaining an acceptable level of mass loss from the PFC surfaces over the course of operation. Therefore, minimizing gross erosion over large fluences (e.g. ~ year) sets requirements for T_e to a few eV and incident ion flux to about 10^{24} $m^{-2}s^{-1}$, which, depending on the material, can result in gross erosion yields that are below the sputter threshold limit of order 10^{-4}-10^{-5} atoms/ion. These limits are specified for the divertor regions where the maximum heat fluxes would be managed. If refractory metal PFCs are used in these regions (e.g. tungsten), fuzz formation would occur due to high fluence and high temperature conditions. One also expects that at the first wall even though the fluxes are five or six orders of magnitude lower, higher electron temperatures and therefore higher ion energies coupled to high wall temperatures would result in formation of surface morphology.

This is yet another challenge: currently, no charged-particle source can provide low-energy *and* extremely high-density (flux) of particles that characterize a fusion reactor plasma edge environment, thus posing some limits for single-effect facilities. Linear plasma devices can provide large fluxes and fluences of plasma onto material surfaces with the relevant incident particle energy and angle distributions. However, even these environments are very aggressive and challenging to diagnose, particularly when it comes to the study of reconstituted surfaces. There is already active work since ReNeW with development of advanced *in-situ* test stand facilities: IGNIS, DIONYSIS, etc. Future characterization must move toward improving the spatial and time scales to connect the multi-scale phenomena of PMI to elucidate the mesoscale (i.e. connecting nanoscale to macroscale effects) science from ion-induced defects at the atomic scale to the macroscopic deformation mechanics of the materials. The situation is similar for time scales from the prompt mechanisms governing ion-induced cascades to the long-temporal scales of defect and morphology evolution on the surface. Also challenging is understanding the various energy scales of particles reaching the plasma-material interface and sub-surface/bulk structure: from low-energy ions at the private flux region in a divertor of order 1-10 eV to high-energy MeV neutrons.

Still another daunting challenge to PMI is understanding the retention of hydrogen isotopes and their migration and permeation through the surface, sub-surface and bulk regions under the extreme conditions of a fusion reactor environment. Deciphering the mechanisms responsible for fuel management in the complex reconstituted surfaces at the PMI and its effect on plasma performance is one of the primary goals in this PRD. In addition, novel materials synthesis and discovery that mitigate issues related to modification by high-intensity plasma has received even less attention. Furthermore, sub-surface effects indirectly impacted by long-term irradiation mechanisms, and bulk effects from neutron-induced damage (as expected in future burning plasma devices) are also poorly understood, and are clearly an area where computational modeling and validation is critical to establishing an integrated understanding of the plasma-material interface.

Understanding the limits to both modeling and diagnosis at these physical scales of space and time, as well as the varied energy scales (e.g. from incident particle distribution), can help establish pathways for innovation and discovery of novel radiation-tolerant materials that can address the evolving reconstituted plasma-material interface and provide a viable solution to PMI under reactor-relevant fusion burning-plasma conditions.

There are five key scientific questions in this PRD that guide its proposed action plans:

1. What are the processes that dominate the spatial formation and destruction of reconstituted surfaces over time?
2. How can we simulate the complex experimental conditions and measure the in-situ evolution of reactor-relevant reconstituted surfaces?
3. What phenomena govern surface composition, morphology, and microstructure evolution of the reconstituted surfaces under reactor-relevant conditions?
4. What are the key neutron irradiation synergies with PMI, and can advanced materials address these?
5. How can the development of multi-scale models to predict the evolution of reconstituted surfaces during plasma exposure be accelerated?

The action plans derived from the scientific questions above are described below, first by summarizing key knowledge gaps, and second by providing guidance towards upgrades to existing facilities, leveraging international collaborations and producing new starts as appropriate.

Action plan 1: Understanding material migration: from microscopic to macroscopic erosion to transport and deposition

Understanding the basic erosion processes in a fusion device is critical for the development of viable fusion reactors. While much progress has occurred in the recent past owing to wall-material changes in several devices, and to the development of in-situ/in-vacuo surface diagnostic techniques, critical knowledge gaps remain in our understanding of PMI, particularly in material migration in magnetic fusion devices. To highlight the necessary research in this area, it is useful to categorize the necessary actions along the following lines (no implied prioritization).

Microscopic erosion: Erosion by plasma ions and/or charge-exchange (CX) neutrals is relatively well understood, although there are unknowns related to the particle fluxes on wall surfaces in a fusion device. Little effort has been devoted to the characterization of CX fluxes in the recent past, although they are expected to be the main erosion mechanism in future devices operating with a dissipative or detached divertor. Similarly, a better characterization of far-SOL plasma fluxes (both in terms of ion energy distribution, and magnitude) in reactor-relevant plasma conditions would improve the input used in large-scale material migration codes, which are currently being developed. This is an important topic also addressed in PRD-C. Particle fluxes and energies during ELMs also need to be studied, as very few measurements exist. Accurate measurements should be accomplished through the deployment of dedicated diagnostics (using Pd-MOS sensors, or laser-based techniques such as laser-induced fluorescence), coupled with in-situ erosion measurement techniques on existing devices. In addition, laboratory experiments will be necessary to further study the effect of plasma-induced material modifications (fuzz formation, void and bubble formation) on erosion mechanisms. Although both experimental and modeling work has begun to identify conditions for the formation of complex surface morphology under plasma exposure, the spatio-temporal dependence of this formation on defect dynamics in candidate PFC materials remains elusive. Furthermore, the emission mechanisms from these complex surface features at the plasma edge and their effect on plasma performance, during both quiescent and transient conditions, remains an open question. This lack of knowledge makes efforts towards extrapolation to fusion reactor conditions problematic.

Macroscopic erosion: An additional possible erosion mechanism for liquid surfaces, present whether through the use of a liquid PFM or because of accidental melting of a solid PFM, is the ejection of droplets from the liquid surface, which can occur through different instabilities or boiling of the surface. Several models (e.g. the HEIGHTS and MEMOS models) have been developed to study this issue and have been able to successfully explain the liquid motion in a tokamak environment. In addition some work has been done to understand droplet ejection dynamics[1]. However, droplet ejection remains an important issue, particularly under fusion reactor conditions, with additional needs that focus on criteria for melt-layer stability and a predictive capability about droplet size and initial velocity. This is particularly important for transient events such as ELMs and disruptions. Experiments with plasma guns are of limited interest since those guns have plasma pressures that are orders of magnitude higher than those expected in a fusion device, whereas the stored energy in current tokamaks is too low. The development of new, dedicated facilities combining pulsed energy/particle sources with relevant plasma conditions in terms of particle flux and energy (i.e. high stored energy), while having low gas pressures, is mandatory to progress in this area. For a non-melted solid surface, ejection of particles might occur during transient events, especially when cracking of the surface occurs, for example, because of the repetitive thermal shocking during ELMs. While negligible today, in future devices that operate with high duty cycles, this effect might become important for the lifetime of the divertor material. A facility capable of operating at high fluence with combined quiescent/transient plasma conditions would enable one to study and understand these effects under relevant timescales.

Given the expected scale of material erosion and migration in future devices, the accumulation of dust particles in the vessel might strongly impede successful plasma operations. Modeling of dust migration in the edge of a fusion plasma has made strong progress. But typically, no existing model can accurately describe the creation and mobilization of dust particles from surfaces, which is a function of the adhesion forces (which are ill-defined for fusion relevant surfaces and materials) and of plasma forces, which partly rely on the limited understanding of how plasmas charge particles. Effort should now be made to better understand the initial steps of dust charging and lift-off from the surface, which requires a combined experimental and modeling approach. On the experimental side, a new technique for controlled dust deposition on surfaces, and accurate accounting of plasma-induced mobilization has recently been developed. On the modeling side, effects such as dust particle adhesion onto surfaces, and a better description of dust charging in a magnetized plasma, should be included in existing codes. Accurately modeling dust creation and mobilization from surfaces would allow predictions of likely dust accumulation sites in future devices. However, predicting where dust will accumulate in a device depends on understanding the mobilization conditions of dust particles. Computational models that can predict both surface morphology evolution and mesoscale particle ejection is critical to understanding material migration.

Transport and deposition: Significant progress has recently been made in the understanding of material migration to the point where the global beryllium migration during the first JET ITER-like wall campaign (~2 years) could be reproduced by the code within a factor 3-10 of the experimentally measured value (after removal of tiles from JET) using the WALLDYN code. However, extrapolations to future devices such as ITER, which has a shaped and almost conformal first wall, require enhanced modeling capabilities to account for 3D structures. Indeed, shaping the first wall to protect leading edges implies that there are plasma-wetted shadowed areas on the first wall. Depending on the local plasma conditions, significant beryllium re-deposition could occur on the shadowed areas of the first wall, which could impact the capabilities for tritium removal. In addition, coupling of such a migration code to an existing or new fluid code would allow a more self-consistent approach, and ideally one would want to have an integrated edge-wall model that has been benchmarked against high-quality data. Here, the role of advanced in-situ PMI surface diagnostics is invaluable to allow this benchmark and provide shot-resolved data, instead of relying on campaign-averaged measurements that are prone to uncertainties, given the large number of plasma configurations run in today's experiments.

Upgrades to Existing Facilities: As mentioned above, a missing element for understanding material migration is a characterization of the particle fluxes (both CX neutrals and far-SOL ion fluxes) to the main chamber wall in existing devices. The former requires the development of diagnostic techniques capable of measuring neutral particle fluxes (and their energy distribution) at various poloidal and toroidal locations for a given plasma scenario, since the CX flux will be strongly affected by local gas injection, NBI, proximity of an antenna etc. Far-SOL flux measurement requires the deployment of electrostatic probes across the machine. In both cases, the measurements should serve as experimental validation for edge plasma codes.

Deployment of in-situ (or inter-shot) diagnostics on various devices is necessary to better understand the process of material migration. Such diagnostics should allow probing of a relatively large area inside the vessel to capture both local and global effects. However, such diagnostics alone are insufficient and should be coupled with detailed edge plasma characterizations that include measurements of flow velocities.

From the perspective of macroscopic surface evolution, optical techniques such as in-situ speckle interferometry and 3D-holography, if coupled with linear plasma devices, would enable tracking of net erosion in real-time.

International collaborations: The study of material migration requires important international partnerships to enable access to a variety of devices with different technical capabilities and operational limits. ITER, JET-ILW, ASDEX-U, in addition to the Asian devices that provide long-pulse and high magnetic field capabilities are among those with which fruitful collaborations could support this action plan. Critically important is the ability to work closely with scientists at these facilities to enable implementation of novel in-situ PMI diagnostics to address the knowledge gaps listed here.

New Starts: A facility capable of studying and characterizing droplet emission from melted (or liquid) surfaces under tokamak-relevant conditions is currently lacking. For example, existing plasma guns have plasma pressures that are orders of magnitude higher than what is expected during ITER ELMs. Ideally, such a facility could be coupled to an advanced linear plasma device.

Action Plan 2: Testing Experimental Conditions of Fusion Reactor-Relevant Reconstituted Surfaces to Elucidate the Plasma-Material Interface

Testing of reconstituted surfaces as they will occur in future fusion reactors will be a challenge. It will require the creation of reconstituted surfaces and the analysis of those surfaces during their evolution. The reconstituted surfaces are a result of the strong interaction of the plasma and surface under the exposure of intense particle and heat fluxes from the main plasma. Due to the strong coupling of the plasma and the material, the surface will evolve non-linearly, leading to multi-scale dimensional changes with completely new surface properties.

The incoming particle fluxes (hydrogen isotopes, helium, impurities and neutrons) will change the composition of the material surface due to their implantation, induced transmutation, preferential sputtering or segregation in case of liquids. As a result the surface morphology will change, leading to nano-bubbles and accumulating in the implantations zone (10-100nm); eventually, nano-structures protrude from a depth of 10-100nm and with fuzz thicknesses that can reach a few microns[2]. Blisters can occur due to accumulation of gases at the grain boundaries many micrometers (10-50μm)[3] below the surface, and cracking occurs along the grain boundaries due to stresses induced by strong transient heat and particle loads potentially leading to whole grain ejection[4,5]. In addition, the non-linear effects of erosion and re-deposition might lead to large surface structures of mm sizes, which are loosely bonded to the surface. These structures include includes multilayer films, formations such as cauliflower-like structures of light impurity elements[6]; re-solidified structures after melt-layer movement and splashing, and re-deposited macroscopic dust particles[7]. Material structures (grain sizes, crystal lattice) change due to recrystallization and amorphization induced by plasma and neutron damage. All those surface modifications have been found first in linear plasma devices. After detailed experimental investigations their occurrence has been confirmed in toroidal devices[8]. Albeit, pre-cursors of those surface modifications can be found in short-pulsed devices; the evolution of the surface restructuring is a continuous process with potentially non-linear effects on the plasma material interactions. Most data for the PMI processes are limited to maximum fluence of 10^{28} m^{-2}, well below the expected fluence in a fusion reactor.

The challenge of re-creating surfaces for study is correlated with how close we can mimic the conditions of a fusion reactor, which are characterized by the impinging ion fluxes > 10^{24} $m^{-2}s^{-1}$, ion fluence ~ 10^{31} ions/m^2, the neutron fluence and related damage up to 150 DPA, with a high He/DPA-ratio prototypical for 14 MeV fusion neutron irradiation conditions. In addition, PFMs and PFCs should be tested in the relevant temperature range anticipated in future fusion reactors, which for helium-cooled PFCs requires ambient temperatures of about T > 600^{o}C.

The analysis of the surface evolution requires in-situ, or at least in-vacuo, diagnosis of the surface morphology, together with the dimensional complexity, the elemental reconstitution due to material mixing, and the phase changes and potential transmutation products. Although the transmutation products are formed in the bulk of the PFM, they are also present at the surface and are expected to change the physical and chemical erosion yield. Non-destructive diagnostic techniques are preferred, since they allow intermittent diagnosis of progressive surface restructuring. New in-situ diagnostics might allow for capturing physics processes to an unprecedented extent that is not possible with the usual post mortem, campaign-averaged, surface analysis. When in-situ diagnosis is not possible, in-vacuo diagnostics utilized inter-pulse will avoid uncertainties arising from atmospheric exposure of samples before surface analysis.

Upgrades to existing facilities: The upgrades to existing linear and toroidal plasma facilities are mainly related to an increased portfolio of in-situ and in-vacuo diagnostics to document the evolution of the reconstituted surfaces in the multi-dimensional space of time and surface depth as described below in action plan 3. In addition, more extensive facility upgrades are mentioned below.

Upgrade of existing sample transfer stations in toroidal devices

Including in-vacuo x-ray photoelectron spectroscopy (XPS) or energy-dispersive x-ray (EDX) in toroidal devices would improve the characterization of the elemental surface composition and the chemical state of the surface composition, providing important information for the understanding of the PMI processes and its dynamics. For example, the processes would give information on the reactivity of the surface with plasma constituents, the pollution of the surface by impurities, and the evolution of coatings, passivation layers and diffusion barriers. Diagnostics with high spatial resolution would be able to map the surface reactivity as a function of the surface morphology evolution during very long pulses. Doing this in-vacuo would eliminate uncertainties from oxide and carbide formations occurring in ex-situ analysis.

Upgrade of an existing accelerator-based neutron source (e.g. SNS, MTS)
The upgrade would allow extension of neutron irradiation data of solids beyond the fission reactor neutron spectra in the absence of a true 14 MeV high flux neutron source. This upgrade would provide neutron damage closer to fusion reactor conditions with high helium/DPA ratios. Irradiation test stations should be designed such that they are well diagnosed, and enable well-controlled (e.g. material temperature) irradiation conditions.

Leveraging international facilities:

Utilizing long-pulse toroidal devices with refractory metal walls (EAST, WEST, JT-60SA, KSTAR)
With a long-pulse toroidal device the surface evolution of divertor PFMs due to deposition of impurities from the main chamber can be investigated. Such a device can also be used to study long-range material migration and might reveal potential disintegration of surface films/structures that will lead inevitably to dust production and mobilization. The effect of the released dust and impurities on the core confinement and stability can be investigated, leading ultimately to the development of plasma scenarios compatible with the plasma facing materials.

Utilizing high-flux linear plasma devices (Magnum-PSI)

Linear plasma devices like Magnum-PSI have proven to be able to provide useful information in testing materials exposed to high flux continuous plasmas and transient heat and plasma loads simultaneously. However, it should be mentioned that their capability is restricted in terms of the plasma temperatures achievable, and they are not able to test neutron-irradiated samples.

Utilizing accelerator-based neutron irradiation sources abroad (SINQ)
Using SINQ as a neutron source to irradiate PFMs closer to the neutron energies of a fusion reactor would allow the effect of high helium/DPA ratios on the microstructure evolution, albeit with the disadvantage of uncontrolled irradiation conditions (e.g. temperature control during irradiation).

New Starts:

A new advanced linear plasma device

As previously noted in ReNeW, a new linear plasma device would greatly expand the portfolio of PMI test stands for deciphering the physics of plasma material interactions at the fusion reactor level. A new device should have the capability of high ion fluxes of $\Gamma > 10^{23}$ m^{-2}s^{-1} (with the aim of 10^{24} m^{-2}s^{-1}); high parallel power fluxes of > 30 MW/m^2; inclined target; B > 1T; steady-state (up to 10^7 sec); > 600^o C surface temperature, and large plasma area ~100 cm^2. Such a new facility should allow exposure of liquid metal targets: gallium, tin, lithium to produce neutron-irradiated material samples with significant DPA, and should have independent control of T_e and T_i at the target. It is expected that the plasma parameters of such a device would cover the relevant n_e, T_e and T_i range of a reactor-relevant divertor plasma in front of the target. This range would enable testing of plasma- facing materials and components with high fluences under reactor-relevant heat and particle fluxes, which is not possible in existing linear devices and short-pulse tokamaks, or even in long-pulse tokamaks. Existing linear devices are limited to a daily fluence of 10^{28} m$^{-2,}$ and toroidal devices to an annual fluence of ~10^{27} m^{-2}. In addition, such a new device would be able to expose a-priori neutron-irradiated samples that have been damaged to significant DPA to very high particle and heat fluxes for up to 10^7 sec. The device would also allow an integrated test of PFCs addressing most of the reactor-relevant PMI physics up to end-of-life cycle, enabling the development of PFCs to a Technical Readiness Level of 6. Such a device (e.g. Fig. VI-2) would be able to investigate the effect of neutron irradiation on the lifetime of PFM for the first time. In addition, the device should be able to prove the high-erosion resistance of a plasma facing component over its lifetime; this would require investigations of the effect of the magnetic pre-sheath, together with the effect of the impurity transport close to the target and the ability to vary ne, Te and Ti in front of the target to the expected reactor conditions at high fluence, which is not possible in existing linear plasma devices or in existing toroidal devices.

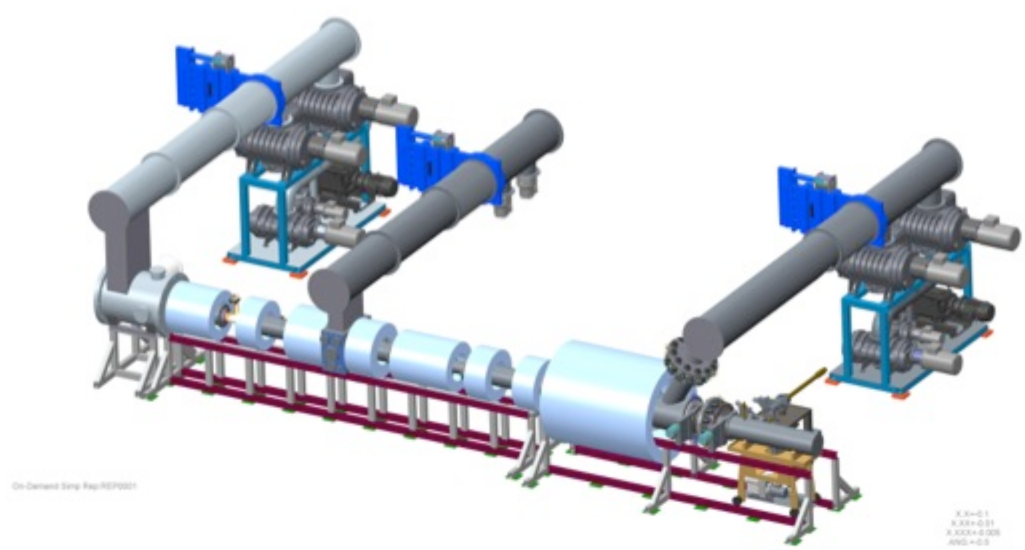

Figure VI-2: Schematic representation of advanced linear plasma device.

A flowing liquid module for a linear plasma device would allow the investigation of power and particle exhaust of a liquid metal target exposed to high-power, high-flux, high-fluence plasmas not possible in current devices. This module would allow for investigations of power exhaust, vapor shielding concepts, deuterium uptake and recycling for example.

A purpose-built, short-pulse tokamak with reactor-relevant SOL and edge plasmas fills a niche that would allow us to examine the effect of material migration on reconstituted surfaces on short time scales, but *under reactor relevant power exhaust conditions* and in an *integrated tokamak scenario*. This would complement high-power linear plasma devices that can achieve reactor-relevant fluences, fluxes, densities and temperatures, but would not be an integrated tokamak scenario (material migration, 3D geometry, gradients, etc), or an upcoming long-pulse tokamak that can achieve high fluences in an integrated tokamak scenario but not at reactor-relevant plasma conditions. For maximum impact, this new short-pulse tokamak should have a dedicated scientific goal and a program that focuses on power exhaust, edge plasma physics and PMI. Such a program will require reactor-relevant edge plasma conditions; the potential to actively heat components so that reactor-relevant wall temperatures can be achieved; a large suite of established and novel in-situ/in-vacuo material and edge diagnostics; the capability to be configured for both liquid and metal divertor surfaces; the ability to run in various advanced divertor magnetic configurations, and the ability to insert/swap sections of the wall and/or divertor to test the impact of new material selections on plasma confinement and stability.

Action Plan 3: Characterize and predict surface composition, morphology, and microstructure evolution of the PMI

Plasma-material interactions are influenced by surface composition, morphology, and structure. Characterizing and simulating how these properties evolve during high-flux plasma exposure poses a considerable scientific challenge. In this action plan, we identify research avenues that present considerable opportunities for progress, assuming parallel development of both solid and liquid metal science and technology.

Evolution of surface composition

Surface composition (including adsorbed impurities) affects fuel recycling and is coupled with recombination, reflection, and particle-impact desorption, as well as with diffusion and trapping of hydrogen isotopes. Surface composition can also drive surface morphology via self-organized instabilities that can be enhanced under high temperatures and fluxes. These effects must be well-quantified individually to project how an evolving surface interacts with the surrounding plasma. Further advances will require multi-effect experiments and models that consider simultaneous processes.

Material co-deposition with hydrogen isotopes alters the composition of the plasma-exposed surfaces. A fundamental question is: how do basic co-deposit properties (i.e.

properties of the reconstituted surface) differ from idealized bulk materials? Addressing this issue requires investigation of how co-deposits bind to hydrogen isotopes[9], and impurities, as well as the microstructure and sputtering behavior of these co-deposits. Predicting material evolution necessitates developing a physics-based understanding of: (a) inter-diffusion between co-deposited films and the substrate at elevated temperature[10]; (b) intermixing during plasma exposure; (c) implantation and sputtering of oxides; and (d) co-deposit response to transients. Beryllium co-deposition will be a key driver for tritium inventory in ITER, and should continue to merit considerable research focus. Re-deposition should also be considered in the context of other materials systems for a more advanced DEMO reactor.

From the perspective of new solid materials (e.g. tungsten alloys)[11], ensuring compatibility with a confinement device will require quantifying preferential sputtering, inter-diffusion of alloy species at elevated temperatures, surface morphology evolution, and assessing the stability of precipitate phases during transients.

Liquid surface evolution under high-flux plasma exposure also faces considerable challenges in the context of characterization. Predicting liquid surface composition evolution involves the additional complexity of taking into account the interfaces between the plasma, liquid metal, and substrate. Several competing processes govern composition, requiring a detailed physics-based understanding of: (a) impurity segregation and adsorption, (b) the creation of saturated liquid (e.g. such as lithium) plus hydrogen layers due to high-flux plasma bombardment, and (c) removal of surface atoms due to sputtering and evaporation. Liquid surface erosion can have nonlinear enhanced erosion and ion-induced segregation that can significantly affect both hydrogen trapping and hydrogen recycling[12]. Liquid surface stratification (atomic-scale layering) can impact hydrogen reflection as a function of temperature. Liquids at a variety of spatial scales also raise challenging questions such as how liquid percolates pores of nano-to-micro scales when interacting with energetic plasma exposure. Liquids also must address questions with regard to the hydrodynamics. For both thin films and flowing systems, it is essential to understand wetting behavior, flow and stability of the liquid metal surface. Specific to flowing systems, the key problems include the effect of turbulence on multi-component systems and impurity segregation.

Impact of large-scale surface structuring/morphology in fusion devices

Surface morphology and structure can strongly influence material response to plasma exposure, as illustrated in Fig. VI-3 [13,14,15]. A dramatic example is interconnected nano-tendril layer growth, or "fuzz", on refractory metal (e.g. tungsten) surfaces[16]. The exact underlying mechanism remains unresolved; further fundamental study of nm-sized subsurface helium bubble expansion and agglomeration is needed to provide further insight. This basic work should be coupled with more applied studies of how nano-tendril growth affects tritium retention, erosion yields, and dust-creation. Further work is needed to assess the impact of a large area of tungsten fuzz on operations and plasma performance[7].

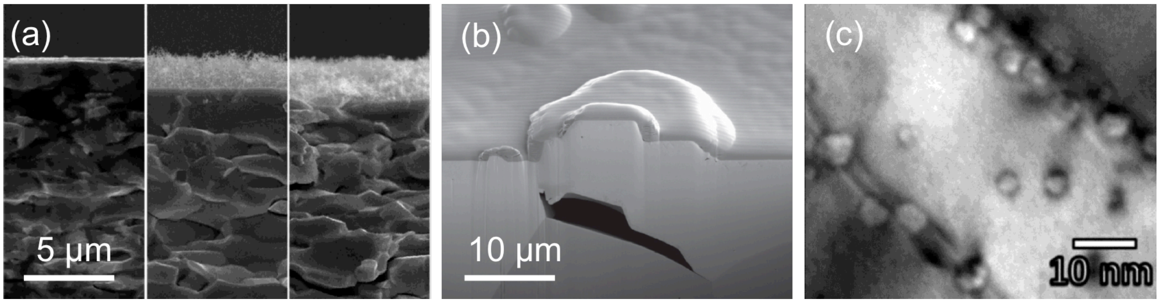

Figure VI-3: Examples of surface structure evolution: (a) image sequence showing the growth of tungsten nano-structure[13] (b) cross-section of a sub-surface cavity created by hydrogen precipitation[14] (c) high-pressure helium bubbles imaged with in-situ transmission electron microscopy during ion irradiation[15].

The formation of hydrogen and helium precipitates also affects surface morphology. Fundamental research is needed to identify (a) the physics underlying nucleation, (b) mechanisms of trapping of mobile species by bubbles/blisters and precipitation of high-pressure gas therein, and (c) how bubbles grow and interact with other defects. It is essential to take into account a full parameter space that includes material microstructure, temperature, intrinsic defects and displacement damage, as well as ion fluence and impurities. Accurate predictions of bubble/blister density and size distributions can then be applied to address how trapping and release of hydrogen affects tritium inventory, and how ejection of delaminated material from the blisters influences large-scale erosion and dust formation.

Repeated thermal transients can lead to surface cracking[17], which can provide shortcuts to the surface for outgassing, influence recycling properties, and eventually lead to material ejection. To address these issues, it is essential to determine how cracking depends on neutron embrittlement and the magnitude of transients.

Upgrades to Existing Facilities: Considerable progress toward addressing the above research needs is possible through enhancements to existing capabilities. Equipping single-effect devices (e.g. instruments using well-controlled ion beams or low-flux ECR plasmas in ultra-high vacuum) with advanced surface analysis tools (e.g. x-ray photoelectron, Auger electron, and high-resolution electron energy loss spectroscopies, ion scattering spectroscopies, etc.) will allow precise measurement of individual processes underlying fuel recycling and surface composition evolution. We envision that these measurements would be applied to model systems (single crystal surfaces or well-defined coatings) to provide direct comparison with quantum mechanical and classical atomistic models. More sophisticated single-effect science ion-beam facilities equipped with in-situ surface diagnosis under *high pressure* can study more complex dynamic mechanisms such as surface mixing, morphology evolution and phase-dependent erosion, *during bombardment from energetic species,* to provide surface-response code validation. Facilities that perform measurements during irradiation under realistic conditions closer to a fusion device can elucidate synergistic effects that are limited using *static* single-effect experiments.

To probe surface chemistry and composition evolution under more complex conditions, improved in-vacuo surface analysis/sample transfer stations can be added to existing linear plasma devices and tokamaks. Such capabilities would offer new insight into surface reactivity with plasma constituents, contamination due to impurity adsorption/implantation, as well as the evolution of coatings and passivation layers. Furthermore, diagnostics with high spatial resolution could map these properties over a large surface area and correlate with surface morphology.

Coupling existing plasma devices or tokamaks with high-energy ion beam analysis (e.g. nuclear reaction analysis or Rutherford backscattering) offers the possibility of measuring compositional depth profiles within the subsurface, including those of hydrogenic species. These techniques are nearly non-destructive, would eliminate uncertainty from atmospheric exposure, and could provide better time resolution than campaign averaged, postmortem ex-situ analysis.

A wide range of laser-based techniques can be incorporated into existing linear plasma and toroidal devices to elucidate surface composition and hydrogen retention. For example, in-situ laser-induced ablation spectroscopy would allow for real-time surface composition characterization, albeit in a destructive manner. Laser-induced breakdown spectroscopy provides quantitative, spatially resolved surface elemental characterization. From the perspective of quantifying trapping of insoluble gas species within PFC materials, laser-induced desorption spectroscopy and in-vacuum thermal desorption spectroscopy could provide considerable insight into trapping energetics.

For ex-situ characterization of structure of codeposits, near-surface bubble formation, and hydrogen blistering, sophisticated microscopy tools (e.g. focused ion beam profiling and transmission electron microscopy) are available and should be used to a greater extent to complement atomic-scale and meso-scale modeling.

Adding low-flux plasma devices to existing surface science facilities that study liquid surfaces enables examination of saturated lithium-hydride surfaces and impurity segregation kinetics. Dedicated microscopy systems configured to accept liquid metal materials are needed to characterize thin film wetting. For high-flux measurements, existing linear plasma devices can be upgraded to accommodate liquid metal targets, as well as incorporation of diagnostics for in-vacuo analysis of saturated lithium deuteride, formed during plasma-exposure (i.e. if one uses liquid lithium).

International Collaborations: JET provides valuable data for modelers to understand the physics of beryllium co-deposition in an integrated tokamak environment. It is crucial that the United States maintains a strong presence at JET through participation in the experimental plan, post-campaign material analysis, and modeling collaborations facilities to better understand the experimental results from when the ITER-like wall was installed in JET.

In addition to capabilities at domestic laboratories, the Institute for Plasma Physics at IPP-Garching (Germany) has considerable facilities for surface characterization and microscopy. Instrumentation for positron annihilation lifetime spectroscopy, and coincidence Doppler-broadening positron annihilation spectroscopy, is available at

JAEA/Tohoku University (Japan). This instrumentation provides detailed insight into defects present within plasma-exposed materials.

New Starts: Combining high-energy ion beam or x-ray analysis with a high-power plasma device to monitor surface composition evolution in-situ would provide access to physics processes not observable using conventional surface analysis tools. One approach involves developing sample exposure capabilities at major DOE user facilities that produce unique light and particle beams for materials analysis, including well-developed capabilities for ambient-pressure XPS and grazing-angle x-ray scattering. This could lead to measurements of key parameters and structures at unprecedented time and spatial scales. One intriguing possibility is the use of pulsed ion beams to enable a "pump" that could in principle be "probed" in the time scale of the modification[18]. This tool could transform our understanding of ion-induced damage in the context of the complex evolving, reconstituted materials under fusion reactor conditions. However, key limitations include low fluxes, and thus the absence of cumulative effects. Nevertheless, studying pre-irradiated materials could prove useful in overcoming the timescale knowledge gap with the prompt ion-induced effects discussed in this PRD.

In-situ and in-vacuo microscopy systems coupled to ion or plasma exposure

To determine the fundamental mechanisms governing surface structure evolution (e.g bubble nucleation, nanostructure growth), we foresee the need for new in-situ microscopy systems coupled with well-controlled ion beam or plasma exposure. In-vacuo scanning electron microscopy (SEM), for example, would enable observation of relatively large-scale morphology changes. Real-time microscopy tools (e.g. transmission electron microscopy) could enable visualization of dislocation loop punching and bubble diffusion at nm-scale resolution. Alternately, a small facility could be configured to examine atomic-scale surface response to low-flux plasma exposure (using scanning probes.) SEMs will give snap shots of the surface morphology evolution. In-vacuo SEMs will be powerful in the characterization of the surface evolution in very high fluence long pulses in linear plasma devices.

In-situ and in-vacuo laser-based spectroscopy PMI diagnostics in toroidal and linear devices

In-situ laser-induced ablation spectroscopy would be the only tool to allow for real-time material composition characterization albeit in a destructive manner and limited to depth resolutions of about 0.5 μm[19]. Laser-induced breakdown spectroscopy (LIBS) would allow for surface elemental characterization in a quantitative manner spatially resolved. Both laser-induced desorption spectroscopy (LIDS) and thermal desorption spectroscopy (TDS) are powerful tools to characterize the hydrogenic retention in PFCs and give information about the trap sites and damage level of solid PFCs. All those diagnostics would improve the time resolution of the elemental composition as well as the damage level evolution of solids. An important caveat with these techniques in the context of reconstituted materials and PMI is the fact that laser-induced removal of material could lead to inadvertent mixing of elements that could make interpretation of the compositions measured more complex. However, these techniques are a powerful tool to measure hydrogen retention at the reconstituted zone and towards the bulk in depths > 1-10 μm. More surface-sensitive *in-situ* laser-based diagnostics using ultrafast pulsing is a novel and emergent area of research for PMI diagnosis.

In-situ speckle interferometry or 3D-holography
Both in-situ speckle interferometry and 3D-holography are the only diagnostics to track macroscopic surface evolutions (net erosion) in real-time.

Action Plan 4: Advanced materials and neutron irradiation effects on PMI

Plasma-facing materials must survive and safely perform their intended function in an extremely hostile environment that includes high heat flux, plasma particle flux and volumetric damage associated with a flux of high-energy neutrons. The plasma strongly perturbs material surfaces through erosion and re-deposition, and hydrogen and helium implantation. The eroded material re-deposits continually as complex-bonded thin-films (i.e. some bonding can be conformal and others non-conformal) whose properties can change over time given their evolving surface morphology and composition. Interaction of fusion neutrons with materials produces residual point defect clusters, and both solid and gaseous transmutation products in the bulk. Intense heat loads lead to high material operating temperatures and significant thermal gradients that effectively couple bulk damage evolution with the physical processes governing near-surface material evolution. Consequently, it is essential to understand, predict and ultimately control these coupled degradation mechanisms in order to develop successful PFMs that will minimally affect plasma performance.

Tungsten is the leading candidate for a solid PFM. It possesses many desirable attributes, but it also has several properties that must be significantly improved for fusion applications. The most notable shortcomings of pure tungsten include a high DBTT, low recrystallization temperature, low fracture toughness, and poor oxidation resistance. The database on neutron irradiation effects in tungsten at fusion-relevant temperatures and doses is sparse, but existing information suggests that the mechanical and thermal properties of tungsten will degrade substantially in the fusion environment. Neutrons will penetrate and damage PFMs to distances on the order of several cm, with the greatest damage in the first ~10 cm. In contrast, the region affected by the impinging helium/tritium/deuterium flux is limited to the near surface over a length scale on the order of ~40 microns. The direct effects of coupled neutron damage and impinging particles are difficult to quantitatively assess given the high particle flux relative to neutrons and the high thermal vacancy concentration due to very high surface heat loads. Neutron irradiation of bulk material below the surface will cause several indirect effects, some of which are important for PMI, such as 1) buildup of point defect clusters, dislocation loops and solid transmutation products that will affect tungsten mechanical properties through hardening, and a decrease of the thermal conductivity, which will cause surface temperature to rise[20] and 2) decreased transport of implanted species away from the surface through increased trapping at radiation-produced sinks such as helium bubbles, dislocation loops and solid precipitates[21]. Indirect effects of neutrons will potentially cause the nature of PMI to change gradually over time as the dose increases[22,23].

A multi-task approach is needed to optimize the opportunity for successful development of tungsten-based PFCs. Three activities are recommended as described below, 1) an expanded experimental and modeling program to characterize radiation effects from bulk tungsten to the reconstituted zone at PMI-relevant temperatures to explore indirect effects of neutron irradiation, 2) an integrated design activity, and 3) an advanced manufacturing activity.

Expanded irradiation effects program

An expanded experimental and modeling program to characterize neutron irradiation effects in bulk tungsten at PMI-relevant temperatures is recommended. This effort contains both potential upgrades to existing facilities and new starts. In addition, such a program should also incorporate new methodologies that look explicitly at the impact of irradiated materials and their properties on surface-dominated properties of reconstituted materials. Reconstituted material properties may become significantly different from bulk properties, given the reconstituted layers will be thin films with high levels of intrinsic stress where the substrate under large doses can affect their surface behavior.

The peak neutron wall load for tungsten in a typical magnetically confined fusion power system is about 27 DPA/full power year[24]. There is little or no experimental data on the effects of neutron bombardment of this magnitude on the properties of bulk tungsten at PMI-relevant temperatures. There is less understanding on how these properties in turn affect PMI properties (e.g. erosion/re-deposition, recycling) of the reconstituted materials that evolve under reactor conditions. In the absence of a fusion-relevant neutron source, the radiation effects data must be collected using surrogate facilities such as fission reactors. It is recognized that attainment of high neutron doses will be challenging in these facilities and potentially confounded by atypical solid transmutation rates. Consequently, ion-beam irradiation studies should be considered to supplement fission reactor irradiations, to achieve higher doses than can be conveniently attained in those reactor experiments. However it is understood that ion-beam irradiations do not fully simulate neutron irradiations, since the ion-beam damage rate is orders of magnitude greater, the damage is highly localized and non-uniform, and solid transmutation effects cannot be assessed. All of these limitations could significantly affect the microstructure evolutionary path and highlight the need for a robust theory and modeling effort to interpret results (see next action plan section on computational modeling). An additional alternative to fission reactors and ion-beam facilities is to utilize spallation sources. Spallation sources provide a neutron energy spectrum that provides the capability to explore irradiation effects under conditions somewhat more representative of the fusion environment, but these sources generally cannot achieve high doses without sacrificing fusion relevancy.

Our panel concluded that indirect effects of neutron irradiation on PMI are a greater concern than direct effects because of the orders-of-magnitude disparity between neutron and particle fluxes at the surface. In addition, since surface temperatures will be much higher than in bulk tungsten, the effects of neutron irradiation on surface evolution might be less significant than in the bulk; this indicates the need for performance of higher-temperature neutron irradiation experiments that are more representative of actual service conditions. It will also likely be very challenging to simultaneously explore the effects of neutrons, impinging particles and thermal loads on PMI without the availability of a dedicated toroidal device. On the other hand, examination of the effects of neutrons on tungsten-based materials at temperatures greater than ~1025 K is needed and would provide valuable data on property evolution that is relevant to the gradually changing indirect effects on PMI that occur over long time. Furthermore, irradiated specimens could be used in a new or upgraded linear plasma device to explore the effects of re-deposited or reconstituted material on a previously neutron-damaged surface.

Upgrades to Existing Facilities:

Integrated design activity in irradiated materials science for PMI

A scientific, multi-scale, integrated design activity is needed to permit the development of analytical methods that will enable design of high-performance, high-reliability fusion reactor PFC components. Single-effects materials research to determine basic properties is essential to establish the feasibility of fusion, but by itself is not sufficient to ascertain the lifetime and reliability of in-vessel components. The limited investment in design studies that has been carried out to date has been helpful for guiding materials development, but a more robust design effort is needed to prepare for next-step plasma devices with much more demanding performance requirements. Design methods must be able to treat time-dependent material properties in components that are subjected to irradiation, complex stress states, and thermal gradients. Existing thermo-mechanical property data and high-temperature design methodologies are currently inadequate for design of a fusion reactor that must simultaneously meet stringent safety and economic attractiveness goals. Furthermore, these design methods must intrinsically couple bulk-material design properties with plasma-material interface properties. For example, integrating low-Z coatings with refractory bulk materials to minimize detrimental effects on plasma performance is an approach that requires a balanced design methodology between surface and bulk property requirements.

Methods for designing with inherently brittle materials are likely to be needed. This can be accomplished by development of new, mechanistically based, computational tools to replace simplistic, largely empirical high-temperature design and operating rules.

In these new tools, material and structural models must be integrated with appropriate material failure models. Guided by engineering design information, the integrated models must be informed by well-designed experiments, supported by high-quality material property databases, and benchmarked by germane component-structure level testing. By building mechanistic understanding into the design process, our ability to address unexpected events and recognize unexpected deformation processes will be greatly enhanced. Ultimately, these tools must be incorporated in design codes and regulatory requirements.

The resources needed to fully develop, test and code-qualify in-vessel components will be significant, but progress can be made by establishment of a coordinated research program involving universities, national laboratories and industry. Leveraging of U.S. investment in an integrated design activity may be possible by forging collaborations with foreign programs that have already invested in material and component development for ITER and next-step devices.

New Starts:

<u>Discovery of advanced PFC composite architectures and novel manufacturing</u>

It is apparent that pure unalloyed tungsten does not possess the requisite properties for successful fusion PFC applications [25, 26]. To address these deficiencies, exploration of composite architectures to dramatically improve resistance to crack propagation by incorporation of ductile phases or fiber reinforcements to the matrix, and implementation of advanced additive manufacturing methods are strongly recommended.

The inherently low fracture-resistance of pure tungsten, combined with thermal stresses associated with high heat flux loading and severe temperature gradients render tungsten highly susceptible to failure by crack initiation and growth. Neutron irradiation will further exacerbate these concerns. Recognition of these facts has stimulated research into various metallurgical approaches to develop other forms of tungsten to alleviate these concerns. For example, approaches to increase the fracture toughness of tungsten by alloying it with rhenium, or by employing severe plastic deformation to drastically refine the grain size, have produced moderate decreases in the DBTT. Such methods will likely prove to be impractical, however, because irradiation-induced segregation of rhenium promotes significant hardening of tungsten by precipitation of undesirable phases, and fabrication of large PFC structures by severe plastic deformation methods may be overly complex.

A ductile phase reinforced composite architecture is a very attractive alternative because as a crack grows in the brittle tungsten matrix, it leaves a bridging zone of ductile ligaments or fiber reinforcements behind the crack-tip that act in opposition to the applied loading, thus reducing the net stress intensity in that region. This approach offers significant promise because every constituent of the composite can be brittle yet the overall composite exhibits substantial fracture resistance. But while such concepts show promise, with a robust science basis for design and fabrication, specific application to tungsten-based PFCs is in its infancy. Limited work to date indicates that several manufacturing challenges must be overcome to demonstrate its efficacy for PFC applications.

The other technology that should be vigorously pursued is additive manufacturing. Additive manufacturing has been an active area of research outside fusion because it provides alternative methods for fabricating composites, functionally graded materials and geometrically complex components that incorporate coolant channels or embedded sensors. This technology enables near net shape fabrication and production of components with intricate geometries. Techniques such as electron beam melting can produce complex geometries a few atomic planes at a time, and ultrasonic additive manufacturing provides the capability for precise insertion of embedded sensors. Significant advantages of additive manufacturing over traditional fabrication methods include, 1) the potential for ultrafine-scale materials engineering and fabrication, 2) considerably reduced waste material, 3) rapid component prototyping and optimization, and 4) the possibility of fabricating components that would be very difficult or impossible to produce by conventional manufacturing techniques. These advanced manufacturing techniques could potentially be used to create unique engineering architectures that could be tailored for fusion-energy-specific applications. However, the improved ability to manufacture complex geometries that additive manufacturing allows may not permit attainment of optimal material performance. This is because current advanced manufacturing techniques may not be amenable to post-fabrication thermo-mechanical treatments that are sometimes required to produce optimal material properties and radiation resistance. The utility of advanced manufacturing needs to be thoroughly investigated to determine its viability for producing more radiation-tolerant materials microstructures extending from the near-plasma surface region through transitions to the coolant interface. These hierarchical gradient structures can be part of designer or intelligent materials that self-heal and adapt to their extreme environment. Further work is needed to understand not only the processing of these materials, but ultimately how their properties evolve under reactor conditions including appropriate testing methodologies. Figure VI-4 illustrates a holistic approach at complex materials design and testing for reactor-relevant fusion environments. The approach combines a program from single-effect, simple systems to establish fundamental understanding that translates to more complex geometries for plasma-facing materials and surfaces coupled to computational tools and *in-situ* diagnosis in prototypical and controlled plasma environments.

International Collaborations: Leveraging international partners such as the TITAN/PHENIX U.S.-Japan programs is an important approach to addressing the three primary activities discussed above. Additional international activities such as the Coordinated Research Projects managed by the IAEA (i.e. effort on irradiated tungsten and fusion PMI) can be an important platform to learn of new resources worldwide that can address the goals presented in this action plan.

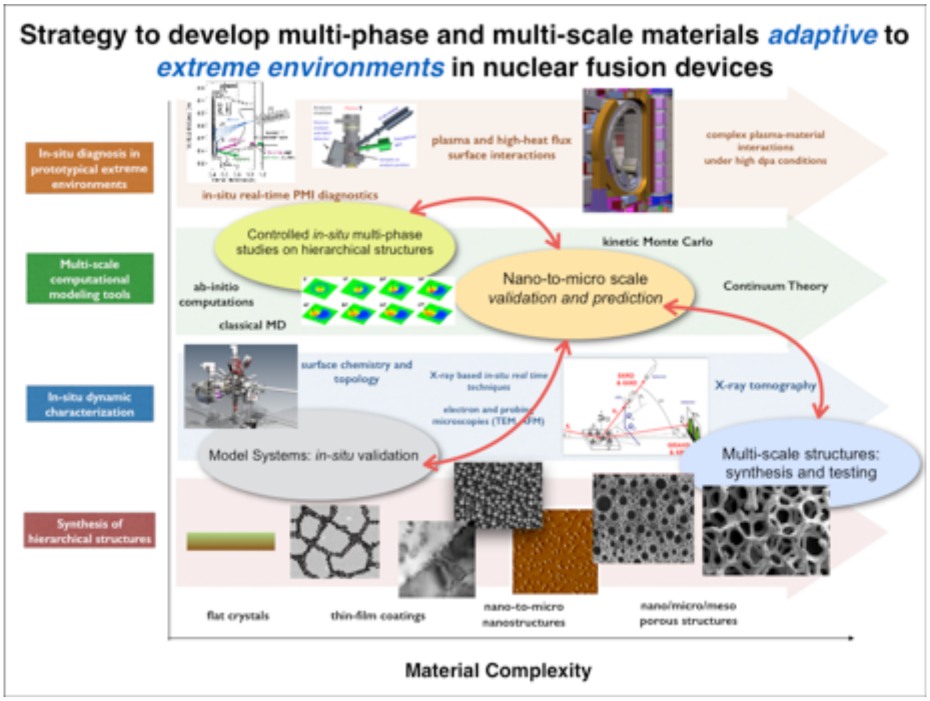

Figure VI-4: Illustration of the overall strategy in action plan #4 in this chapter, to develop multi-phase and multi-scale materials adaptive to extreme environments.

Action Plan 5: Modeling, Theory and Validation of the PMI

As noted earlier, the United States has promoted the advancement of multi-scale modeling activities within the SciDAC-3 (Scientific Discovery through Advanced Computing) program since the ReNeW report. These multiscale models attack the complex PMI problem from both a "bottom-up" atomistic-based approach, and a "top-down" continuum perspective, and focus on the hierarchical integration of kinetic processes of species reactions and diffusion in order to model increasing time scales. The simultaneous use of both an atomistic and continuum approaches to attack a wide range of issues minimizes the risks of using just a single-scale approach and furthers the prospects for scale bridging or multi-scale integration. Such issues include complex and inter-related surfaces, defect and impurity impingement, radiation damage, diffusion and evolution processes responsible for PFC surface, and bulk materials response. However, it is very important to note that these emerging modeling capabilities are very much in the early stages of implementation for PMI problems, as is the science of PMI, in general, highlighting the need for close integration of modeling with improved experimental capabilities. Closely coordinating modeling activities with experimental studies will both provide validation and guidance to the specific modeling activities, as well as to the design of experiments to resolve specific PMI issues. Such interactions increase the likelihood of successfully bridging the scales from the short-time, atomic-scale processes to the longer-term, micron-scale surface morphology changes.

Another critical challenge to both modeling and experimental PMI validation is understanding the evolving, reconstituted material under prototypical environmental fusion-reactor conditions, which will require major advances in computational modeling. For example, developing models for the evolving surface morphology at the PMI of reconstituted material, and simultaneously assessing the effect of neutron-induced damage as it couples to the sub-surface region for long doses, is a major feat.

The multi-scale approach involves atomistic simulations utilizing molecular dynamics (MD) as well as binary collision approximation simulations of non-planar, complex geometry surfaces with fractal features to describe the fast (i.e., time scale < 10 ns) dynamic processes of sputtering, re-deposition and surface evolution, as well as bulk defect and helium/hydrogen species evolution in mixed tungsten-helium–hydrogen–beryllium systems. Accelerated molecular dynamics (AMD) methods[27] can be used to identify key evolution mechanisms occurring on time scales up to seconds. AMD provides a unique approach that enables deterministic simulations of plasma ion flux at appropriate rates, and captures material evolution for durations up to and beyond the time scale of seconds that are needed to identify slower, rare-event processes that contribute to surface, defect and impurity evolution. The AMD approaches, complemented by techniques for activation energy barrier identification can determine activation energies and pre-factors that are used to define the reaction rates of individual mechanisms. First-principles electronic structure methods can be instrumental in providing interaction forces, basic thermodynamic and kinetic interactions and rates, and will be utilized where existing interatomic potentials are deemed inadequate, as is likely the case for the hydrogen–tungsten interactions, and for mixed materials surfaces of varying compositions, including impurities. Surface evolution phenomena, including re-deposition, fuzz growth and surface migration can be investigated using reduced-parameter continuum techniques with the goal of developing evolution models that reduce the dynamic complexity to the most pertinent and tractable variables.

Insight into mechanisms and rates of occurrence are the essential outcome of atomic-scale modeling, which can be coupled to reduced parameter models to effectively integrate across the length and time scales in a hierarchical multi-scale modeling paradigm. These insights (and corresponding rates) are then used as input in a sequential (hierarchical) fashion for micron-to-millimeter-scale models; such coarser-scale models may be in the form of either a kinetic Monte Carlo (KMC) simulation or a spatially dependent reaction-diffusion rate theory or cluster dynamics simulation to model the long-time morphological and chemical evolution of a plasma facing component at, near, and below the surface.

Upgrades to existing models and experimental validation:

The goal of this action plans is first, to improve model validation along spatial scales: from both a "bottom-up" atomistic-based approach to a "top-down" continuum perspective. Second, to improve model validation along temporal scales: hierarchical integration of kinetic processes of species reactions and diffusion in order to model increasing time scales. As mentioned earlier, the prevailing approach to integrate such multi-scale modeling techniques is through a hierarchical, information-passing paradigm. The atomistic-based materials modeling approaches naturally link to particle-in-cell, kinetic sheath models for interfacing across the plasma-surface boundary to provide the incident ion energy and momentum as a function of the plasma environment and surface morphology. Continuum PMI models likely will initially interface with continuum-level fluid models of the plasma scrape-off-layer, but could also be linked to a particle-in-cell model to provide a more spatially-dependent description of the incident particle and thermal flux distributions. The bulk material below the surface close to the near-surface region can be modeled using the same set of hierarchical techniques, and a similar approach to scale-bridging can be used. The bulk is where radiation damage processes lead to the nucleation and growth of extended defect clusters, gas bubbles and local chemical segregation. Research should incorporate multi-constituent models to address range of anticipated impurities and further the surface to boundary plasma coupling, including sheath effects.

The fidelity of the modeling predictions of long-time behavior, whether using continuum approaches or discrete-particle KMC methods, is determined by the extent to which the most important kinetic processes and rates are accurately predicted and incorporated into the physical reaction-diffusion models. In such a hierarchical modeling approach, independent of the choice of time scale, the use of uncertainty quantification (UQ) techniques can provide important insight for identifying important parameters in process/rate prediction. Such parameters result from either the intrinsic error of interatomic potentials used in the atomistic simulations, or from the inherent uncertainty in environmental conditions in the plasma. The passage of these uncertainties through the multi-scale modeling hierarchy will be important in assessing the impact on predicted PFC behavior. Furthermore, the UQ will be used to evaluate the extent to which the coupled first- and second-order kinetics influence observed behavior (e.g., how the mobility of a vacancy cluster can influence the resulting size and number density of gas bubbles that act as trapping/retention sites for permeating hydrogen). Such UQ studies of the parameter sensitivities from non-linear coupling among the reacting species can prioritize additional atomic-scale simulation studies.

Recent advances in computational modeling of PMI in tungsten and beryllium exposed to helium and hydrogen plasma conditions have been promising, as noted earlier within this document, and the modeling activities are rapidly maturing to the point of strong interaction with experiments to resolve critical PMI issues. However, it is important to note that the modeling activities are very much in their infancy. Correspondingly, an enhancement of modeling activities, strongly coordinated with experimental characterization and the in-situ/in-vacuo development of diagnostics and materials characterization, should be adopted to expedite the resolution of key PMI challenges associated with minimizing erosion and tritium retention, managing extreme energy exhaust and heat fluxes, and avoiding deleterious degradation effects from the fusion nuclear environment. Among the most pressing future needs within the field are the continued evaluation of the effect of the implantation rate and surface temperature on the surface morphological response of PMIs exposed to low-energy plasma bombardment, and the analysis of synergistic interactions between helium and hydrogen that are expected to influence the amount of tritium retention and fuel (hydrogen isotope) saturation, in addition to the role of impurities and mixed material formation. The continued development of modeling capability will both fuel advances in scientific understanding, as well as provide key tools required to evaluate advanced PFC and divertor concepts to drive innovation in PMI science towards a fusion nuclear science facility and demo reactor.

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Chapter VII

Priority Research Direction ‘E’ – Core-Edge Plasma Compatibility

VII. Priority Research Direction 'E' – Core-Edge Plasma Compatibility

<u>PRD-E:</u> Understand the mechanisms by which boundary solutions and plasma-facing materials influence pedestal and core performance, and explore routes to maximize fusion performance

Conditions at the plasma boundary, both the divertor dedicated to handling the heat flux and the main chamber that comprises most of the surfaces, are known to affect the performance of the hot core plasma where fusion takes place. This interaction takes place largely in the outer 10 percent of the plasma, where self-organized transport barriers can occur. Parameters at the top of this "pedestal" provide a key boundary condition that largely determines the core profiles and fusion output power. The optimal conditions for handling high boundary heat fluxes differ from those for a self-sustained core. Physics in this outer region is complex and multi-scale; the understanding required to quantitatively predict the influence of boundary solutions is incomplete. It is proposed to address this critical gap through a coordinated program of experiments and modeling, primarily on U.S. experiments with targeted upgrades, and supplemented by international devices. Research would be extended to more reactor-relevant conditions in a U.S.-led Divertor Test Tokamak facility, which would develop optimized core-boundary solutions for future fusion devices.

VII. 1. Additional Background

The science challenges, research status and knowledge gaps regarding integration of boundary solutions, including both the divertor and main chamber and their plasma-facing materials, with attractive core scenarios are reviewed in Section II-4. To summarize the points most critical for this Priority Research Direction, the coupling by which the boundary impacts the core plasma primarily occurs in the region at and just inside the last closed flux surface, roughly the outer 5-10 percent of the plasma. In discharges with an edge transport barrier, which is typical and indeed needed for high

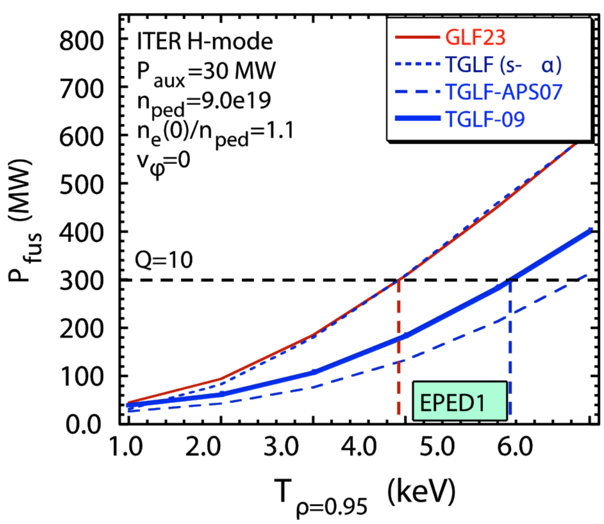

Figure VII-1: Predicted fusion power in baseline ITER scenario, vs assumed pedestal temperature, for several transport models; all show strong sensitivity[1]. EPED1 represents predictions from a leading pedestal model[2].

confinement, this region is called the "pedestal" as it raises core profiles. It has been shown both theoretically and experimentally that conditions at the top of this pedestal strongly influence the profiles of the whole plasma, due to transport physics which tends to limit temperature gradient scale lengths, so-called "profile stiffness." An example of the projected influence on ITER fusion power[1] is shown in Fig. VII-1.

Boundary solutions, and fluxes from the confined plasma, combine to set conditions at the last closed flux surface, such as densities of electrons, main species ions and impurities (n_e, n_i, n_{imp}), electron and ion temperatures (T_e, T_i), fuel recycling coefficient and edge turbulence. We know experimentally that these conditions influence the pedestal temperature, density and pressure. Certain separatrix conditions, such as too cold a temperature or too high a density, tend to degrade pedestal pressures and thus core performance. However, our present predictive capability is insufficient to quantitatively predict these effects or the corresponding operational limits (e.g. in density, impurity radiation).

The goal of this proposed priority research is, through experiments and modeling, to improve our knowledge of pedestal transport and profiles sufficiently to predict the impact of materials and scrape-off layer conditions on the core performance. This will define mutual constraints on boundary and core parameters. We will then use this knowledge to develop and optimize compatible core-boundary solutions, without large transients.

The physics in the pedestal region depends on both plasma physics, which can generally be expressed by local dimensionless plasma parameters and governs transport and stability, and on atomic physics, which depends on absolute plasma parameters such as T_e and n_e. These determine, for example, the ionization profiles setting the edge particle source, and the radiation profiles. It is not possible to match completely and simultaneously all of the expected parameters for fusion burning plasmas in a single, smaller-scale device. For this reason, a combination of experiments, matching key sets of parameters and studying subsets of important physics effects, and improved models validated by these experiments, will be required. Models of the pedestal are being developed and simulations are improving to include a more complete set of physical effects. This development is a priority topic in the Integrated Simulation for MFES Workshop.

In addition to defining boundary constraints and core performance in future fusion devices, a validated predictive capability for transport in the pedestal will aid development and understanding of techniques which may control transport, avoid the instabilities which can cause large transient events, and improve core-boundary compatibility of other actuators needed to sustain and control fusion device. If left unresolved, issues of core-edge integration have the potential to lead to serious performance degradation in future burning plasma devices, including ITER. Recognizing the need and readiness to make progress in this vital area, all three 2015 FES Workshops (i.e. Control of Transients, and also Integrated Simulation, in addition to this report) are recommending that pedestal transport be prioritized. The overall impact of a successful research effort would be to optimize integrated performance for extrapolation to more attractive fusion reactors.

Key scientific questions to be addressed are described below, followed by a proposed action plan to address them and achieve the above goals.

VII. 2. Main Scientific questions

What physics sets the profiles of plasma temperature and density in the edge transport barrier or "pedestal"?

The physics setting profiles of electron and ion temperature and density in the edge transport barrier region is particularly complex, compared to either the core plasma or the scrape-off-layer. Turbulence within the barrier is lower than in the core, though a residual turbulence level remains. Heat fluxes through the regions are comparable; consequently, local pedestal gradients steepen. As summarized in II.4, there has been great recent progress in predicting pressure limits at the top of the pedestal[2]; the range indicated by EPED1 model in Figure VII-1 is much smaller than the factor-of-several uncertainties a decade ago. However, predictions of temperature and density profiles are not as advanced; pedestal density is typically an input in present models. These profiles have short scale lengths that can approach plasma scales such as the ion gyroradius, violating standard assumptions in core gyrokinetic theory and models. Many complex plasma physics effects can be important, some involving average orbits and others requiring full kinetic distribution functions. Inward pinches as well as outward diffusion are possible, and there can be significant poloidal variation, in particular large influences from the X-point region near the divertor. Hence measurements and modelling of distribution functions, at multiple locations, are required. The steep gradient region typically spans the separatrix, extending a short distance into the near-SOL; gradients in the near-SOL and pedestal are highly correlated. Changes in the boundary can thus impact the pedestal, and vice versa. As gradients steepen, eventually micro-instabilities become unstable and start increasing transport, often limiting temperature and density even before the onset of a large-scale ELM. This is in fact desirable and even necessary in order to eventually avoid ELMs. It is unclear in general which classes of instabilities dominate; this may well vary with experimental parameters and even across the barrier, given order of magnitude variations in density and temperature across this narrow (few mm to few cm) region. Improved measurements and modeling of turbulence are thus required, to help understand the cause and magnitudes of thermal and particle transport.

Prediction of density profiles is particularly complex. While heat is largely deposited in the core and transported towards the pedestal, the particle source, i.e. the ionization of recycled neutrals from the divertor and wall, is largely in the SOL and pedestal itself. This source term depends on the parameters of each experiment. For most present experiments, ionization profiles extend across much of the pedestal, while for large burning plasma devices most ionization will be in the SOL. This may cause major changes in fueling rates and dynamics. Present experiments with the largest neutral opacity are Alcator C-Mod, which operates at high n_e, and JET, which is the largest in size. Inter-machine studies have confirmed that such differences in particle source profiles indeed change the relative profiles of pedestal temperature and density[3,4]. A further complication is that neutral densities can vary significantly with poloidal and perhaps even toroidal location; hence, a comprehensive suite of diagnostics will be required to interpret experiments and validate models. Charge exchange from neutrals can also cause significant thermal transport.

While the focus of current and proposed research is on transport in tokamaks, the configuration of ITER and the present leading candidate for fusion reactors, we note that edge transport barriers have also been observed in stellarators, and much of the underlying physics should be common to other configurations.

How do low vs. high retention and recycling and retention of fuel influence the pedestal region?

The impact of particle fueling and transport is particularly notable when changing the fraction of fuel particles that re-enter the plasma as cold neutrals via recycling. Low recycling, e.g. operation with lithium, has led to substantial improvements in pedestal performance on NSTX, increasing confinement 50-100 percent. Full wall lithium coatings on LTX have resulted in an improvement of confinement above H-mode expectations by as much as a factor of 3-4. The reduction of edge neutrals in LTX is associated with a broadening of the electron temperature profile, and steepening of the edge temperature gradient. These results are broadly consistent with predictions for low recycling walls[5], and point to a potential for considerable improvements in confinement performance. Enhanced performance has also been observed in experiments with carbon PFCs that are conditioned to temporarily reduce particle recycling. Notable examples include "supershots" on TFTR, which led to its maximum D-T fusion production[6], and the VH-mode on DIII-D[7]. Further experimental and theoretical research in this area is clearly warranted.

In general, high recycling seems to increase edge density gradients, while lower edge neutral fueling can lead to decreased density gradients in the outer pedestal. The behavior in the outer half of the pedestal appears to be key. In the NSTX lithium experiments, it has been observed that the electron temperature profile remains fixed in this region even as the density is reduced by lithium pumping. This moves the peak pressure gradient to smaller radius, which is stabilizing to peeling-ballooning modes. This may allow the pedestal to broaden, leading to higher pressure limits. This picture is supported by recent lithium injection experiments at DIII-D, where a flattening of the profiles in the outer pedestal was also observed to correspond to wider and higher pressure pedestals, although in this case the flattening appears to be due to an increase in fluctuations rather than to changes in recycling. It is possible that, once more fully understood, such improvements in performance might translate to burning plasmas, potentially enabling substantial margins in fusion Q, and reductions in cost for some next-step devices. Successful exploitation depends also on resolving technical challenges of lithium PFCs, the subject of PRDs A and D.

Even with solid, high Z plasma facing components, increasing surface temperature, as will be required for efficient fusion power production and to reduce tritium retention to the low levels needed for plant safety, is likely to have similarly important effects on the pedestal. There is currently no predictive or experimental capability to study these temperature effects on integrated plasma scenarios.

How are impurities transported in the pedestal and what is their effect?

Particle transport of impurities is particularly important, and challenging. While ideal fusion plasmas would be composed nearly entirely of hydrogenic fuel, in practice small amounts of material can be released from the PFCs, including actuators, via PMI. The probability of these contaminants entering the confined plasma as opposed to being screened depends on location (eg divertor vs high- or low-field-side main chamber), and other effects as discussed in PRD C. Gaseous impurities may be deliberately injected for diagnostic purposes, or in order to dissipate a fraction of the plasma energy before it reaches PFCs and reduce their heat flux. Helium "ash" is, of course, a byproduct of the fusion reaction which needs to be transported from the plasma core and exhausted from the system.

Once inside the separatrix, the transport of impurities is complex. It occurs due to a combination of turbulent fluctuations, which mainly cause outward diffusion, and neoclassical effects due to ion orbits. The latter can provide an inward "pinch", moving particles up the plasma gradient, as well as outward transport, and is strongly dependent on the atomic number Z of the impurity. Depending on these effects, high-Z impurities can either accumulate in the core, cooling and diluting the plasma, or they can be readily "flushed" into the boundary layer for exhaust. There has been considerable experimental and theoretical progress on core impurity transport[8,9], driven by concerns for ITER and results such as those on AUG and JET. Transport analysis across the pedestal is even more challenging, due to small spatial scales and the incomplete understanding of transport mechanisms; improvements in both diagnostics and modeling will be required. ELMs, both periodic and continuous, clearly play a large role in exhausting impurities. Indeed, some form of instability is likely required to ensure that impurities do not accumulate in the core.

Impurity profiles can also affect the electron and main ion transport, edge stability and pedestal profiles. As one example, adding small quantities of nitrogen to JET plasmas with an ITER-like wall and reduced performance (see Section II.4.2) resulted in higher pedestals, perhaps replacing other low-Z impurities which had previously been present when JET used carbon PFCs. Such effects, both deleterious and positive, remain to be understood so that they can be controlled or optimized.

How is pedestal transport modified by edge transient control techniques and in regimes without large transients?

Instabilities in the pedestal region are an important, often dominant, source of transport in the pedestal. In regimes with large ELMs, the outward flux at an ELM crash can be the primary mechanism to expel particles, even when time-averaged over the periods between ELMs. However, it is recognized that large ELMs are intolerable for ITER and DEMO, and even small ELMs, such as those targeted by the ELM mitigation strategy for ITER, are unlikely to be allowed in DEMO, due to the damage accumulated over many repetitive events. The active ELM control techniques currently available (e.g., pellet pacing or resonant magnetic perturbations; see the report of the FES Control of Transients Workshop) typically reduce the pedestal pressure. Further, large ELMs cannot be relied upon as the primary means of particle transport.

In general, each method of actively mitigating or avoiding ELMs requires some alternate means of driving particle transport to flush impurities. Truly low turbulence, ELM-free regimes tend to steadily increase the main plasma and impurity density, leading to transient phases which ultimately collapse. Fortunately, there is a class of ELM-free scenarios that employ turbulence-enhanced particle transport. A feature common to several of these, e.g. resonant magnetic perturbations (RMPs), and to regimes with continuous fluctuations such as quiescent H-mode (QH), enhanced D_α (EDA) H-mode, and I-mode is that particle transport resulting from perturbations appears to be larger than thermal transport, allowing steady density with high temperatures. Our pedestal experiments and modeling will need to include, and even focus on, such regimes. RMP physics and transport involve 3-D effects, which will require significant extension of models, and have much in common with stellarator physics.

While these aspects of ELM control challenge the integration of edge solution with the core, it should be noted that further studies are required using test-stands able to better replicate ELM heat and particle loads to determine material damage thresholds, as detailed in PRD A. Combined with efforts to develop materials resilient to many repetitive transients, the results could affect the requirements for ELM mitigation schemes and hence the degree of this challenge.

What are the limits to robust pedestal operation, and how do they constrain divertor solutions?

In addition to predicting pedestal profiles in conditions which are optimal in terms of core performance (e.g. moderate density, low radiation), we need to understand and predict the impacts of effects which can degrade the pedestal. These effects will ultimately set the constraints on tolerable separatrix conditions and thus on development of boundary solutions. As discussed in section II.4, optimal conditions for the boundary and core are often different. For example, the power handling is easier with higher density and strong radiation, but the pedestal pressure is higher at moderate density and low radiation. Mutually compatible conditions need to be determined, which can only occur if impacts are well understood.

A number of possible mechanisms may lead to degradation of the pedestal, particularly with dissipative or detached divertor regimes, which, as discussed in PRD B, are the main options for avoiding erosion in fusion plasmas. Some of these include, i) increased pedestal turbulence and transport and reduced stability, due to an increase in the normalized collision frequency, ii) excessive neutral flux resulting in charge-exchange losses and modification of the electric field profile that sets up the transport barrier in the first place, iii) SOL conditions effectively propagating inward at detachment onset or approach to the density limit, iii) reduction of power across the separatrix, cooling the pedestal and potentially causing back transition to L-mode.

Experiments will need to be designed which separate these potential effects, have parameters which are as relevant as possible to key burning plasma conditions, and importantly, have sufficient diagnostics to isolate causes of pedestal degradation. For example, assessing effects of neutrals would require CX and neutral diagnostics, and studying changes in microstability would need measurements of turbulence, and ideally the edge bootstrap current. Models would then be tested to see if they correctly predict the trends and onset of degradation, and to identify the dominant mechanisms. Validated simulations can be used with greater confidence to predict the limits and solutions for future devices.

How can the pedestal and divertor be integrated to optimize performance of burning plasmas?

The ultimate goal of pedestal research is not only to understand and predict pedestal profiles and to avoid degradation, but to optimize and improve them. If pedestal pressure can be increased beyond the typical limits of large ELM regimes on which present global confinement scalings are primarily based[10], then energy confinement would be increased. Studies have shown that even a small (10-20 percent) improvement

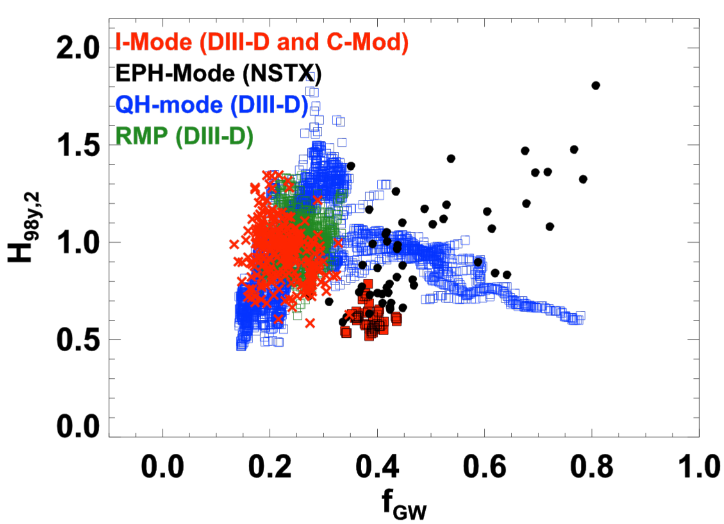

Figure VII-2: Normalized confinement factor H_{98y2} vs fraction of the empirical density limit, for several regimes without ELMs on US tokamaks Alcator C-Mod, DIII-D and NSTX.

would lead to more attractive fusion devices[11]. Pedestal improvements could also provide margin to compensate for any deleterious effects on confinement[12]. A number of examples illustrate that such improvements are possible. Device shaping has been shown to increase pedestal stability limits[13]. Several quiescent regimes (eq. QH-mode, I-mode, Enhanced H-mode, RMP H-mode) have exceeded the H_{98y2} scaling, either transiently or in stationary conditions[14]. Figure VII-2 shows examples from a 2013 FES Joint Research Target report on this topic. Pedestal modeling has been used to identify a route to enhanced stability at high density and shaping; initial experiments to explore this "super H-mode" are promising[15] (Figure VII-3). As noted above, low recycling can broaden the pedestal, and advanced divertor geometries may potentially also have positive effects.

Active means of controlling and optimizing pedestals would also be extremely valuable, increasing flexibility in operation of fusion devices. A general issue with transport barriers is that turbulent transport can become too low, leading to particle accumulation, MHD instability or loss of control. The observation that particle transport can be separated from thermal transport offers the prospect that it may be possible to actively and independently drive the two transport channels. Externally imposed magnetic perturbations have clearly increased particle transport in a number of devices. Other ideas which have been proposed but not fully explored include driving naturally present instabilities with external antennas, or using localized RF waves (See Section II.4). Such active controls could potentially ease the impact of PFC materials on the core plasma performance. For example, experiments with the JET ITER-like wall relied only on large or medium-sized ELMs to flush tungsten from the plasma. When tungsten concentrations increased, higher gas fueling was required to increase ELM frequencies, with negative effects on the plasma scenario. A direct means of driving outward tungsten flux could provide needed flexibility to optimize the core performance.

A further consideration in optimizing burning plasma scenarios is the imperative to avoid sudden losses of current, known as disruptions. These could have catastrophic effects on PFCs. Disruptions are recognized as one of the key challenges for tokamak fusion reactors, and along with ELMs are the subject of the FES Control of Transients Workshop. As discussed in that report, it is important to recognize that

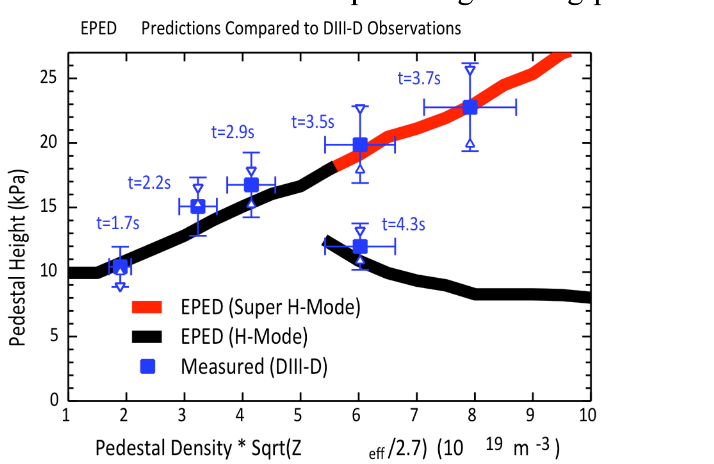

Figure VII-3: Pedestal height for 'Super H' mode regime (red curve) is much higher than for standard H-mode (black), at high density and strong shaping. Initial experiments (blue points) agree well with predictions from the EPED model.

disruptions can be caused not only by exceeding stability limits in the core, but by exceeding material limits in the PFCs; sufficient margin must be allowed in both limits, and improved methods of predicting and controlling PMI-related events are needed.

VII. 3. Action plan

VII. 3. 1. Advance understanding and complete predictive capability of the pedestal using enhanced diagnostics, theory & modeling and validation experiments on existing U.S. facilities.

We propose to make significant progress on most of the key scientific questions detailed above by means of a coordinated campaign exploiting existing U.S. experiments. This would be closely coupled to an effort to improve models of the pedestal, a focus of the Simulation Workshop. To make a major impact, this campaign would need to enhance present efforts in a number of key areas:

1. Diagnostic enhancements. As discussed above, increased capabilities are needed to measure a number of key parameters. Examples include: i) 2-D measurements of neutral ionization profiles, to give the source of particles, ii) main ion average temperature, and also its distribution function, iii) fluctuations in density, temperature and potential, iv) impurity density profiles, v) radial electric field profile, vi) edge current density. The spatial resolution of present diagnostics should be improved to fully resolve the sharp ~cm scale gradients in the H-mode pedestal region. Some of these enhancements are also required for PRDs B and C.

2. Coordinated experimental scans in multiple devices. Developing models must be able to reproduce key trends with plasma parameters, not just profiles from individual discharges. The United States is fortunate to have devices which together can span a large range in relevant parameters, such as collisionality and neutral opacity, materials (currently molybdenum, carbon and lithium), divertor and confinement regimes. These devices can access different divertor configurations. While it is not possible to simultaneously match all relevant parameters of burning plasmas, experiments should aim to access key parameters in single-effect scans. Key examples should include:
    a. Density scans, scanning neutral ionization lengths towards smaller values, normalized to the pedestal width and density to a larger fraction of the disruptive limit n/n_G.
    b. Collisionality (power and current) scans, towards low pedestal collisionality. Experiments on multiple devices will help to break correlations between density-related normalized parameters that can each affect profiles.
    c. Confinement regimes. Experiments should not be limited to H-modes with large ELMs, but should include cases with ELM mitigation and naturally ELM-free regimes such as QH-mode and I-mode. As noted,

cases without large ELMs are the most reactor-relevant and can have quite different transport.

d. Impurity seeding, to progressively decrease the separatrix temperature and obtain divertor detachment. Cases in which degradation of the pedestal pressure is observed will be particularly valuable in understanding the limiting phenomena and constraints on divertor solutions.

e. Variation of magnetic configuration (lower vs upper and double null, and more complex configurations such as "snowflake" and "X-divertors") to understand influence of X-points and drifts.

In each case it will be important to obtain comprehensive sets of measurements using the expanded diagnostic set in (1).

3. Enhanced effort in modeling and validation. Interpreting these experiments, and using them to validate current and improved models, will require a large analysis effort. Preparing a full set of data for even a single discharge, even with the limited diagnostic sets on present day devices, as input and comparisons to simulations requires substantial time. Analyzing the multiple scans in (2), and more complete diagnostic sets in (1) (e.g. inference of the 2-D neutral profiles) will need additional personnel, including diagnostics experts, modelers, and "analysts" dedicated to objective comparisons between data and simulations.

   Validation will be closely coupled to parallel developments in simulation capability, which will be needed to resolve the science issues. The Plasma Boundary Panel of the FES Integrated Simulation for MFE Science Workshop is developing a broad strategy which will progressively improve simulations of the structure and evolution of the coupled pedestal-SOL system and ultimately incorporate transient events, detached divertor plasmas, and RF antennas. The validation effort will serve to identify the parameters and regimes that are well explained, and the new physical effects which need to be included. The improving models in turn should be used to guide optimization of the pedestal.

This campaign (1-3 above) could begin immediately, provided increased resources of diagnostics, experimental run time and analysts are prioritized at U.S. facilities.

VII. 3. 2. Extend pedestal research with targeted experiments on international tokamaks

The validation experiments outlined above can be primarily carried out on U.S. facilities. These have the most complete diagnostic sets, ready ability to add further targeted diagnostics, and greatest control over run priorities. However, it is likely that gaps will emerge which could be supplemented by targeted experiments on overseas devices. In the near term (five years) the greatest opportunities for such experiments seem to be in E.U. tokamaks. The large size of JET gives shorter normalized ionization lengths, allowing important tests of fueling in scenarios where much of the ionization is outside the separatrix. It also offers ITER-like plasma-facing materials, and as noted in Section II.4, has encountered unexpected effects of these materials on the pedestal and core scenarios. U.S. expertise with pedestal physics and metal PFCs could potentially help to resolve these issues, which are crucial for the planned JET DT campaign, and provide us with knowledge for future ITER participation. ASDEX Upgrade has parameters comparable to DIII-D but has tungsten PFCs, offering a useful point of comparison. The program is also making studies of strong impurity seeding a priority, providing useful data on the limits to radiation and the mechanisms behind eventual degradation of core performance. MAST-U, when it resumes operation in 2017, will complement NSTX-U and provide the first information with a Super-X divertor. We recommend that support and priority for EU collaborations in the PMI and pedestal areas, which is currently relatively limited, be increased.

In the medium term (five to ten years), as diagnostics capabilities mature, the Asian tokamaks EAST and KSTAR will offer other opportunities for collaborative experiments. The new JT60-SA tokamak (scheduled for completion in 2019) will have size comparable to JET, and hence greater neutral opacity. However, none of these devices has been designed with PMI issues as its primary mission, or has the flexibility to test all of the boundary solutions under active consideration. Power densities and parameters also are not significantly beyond those on present U.S. devices. Thus, we believe that the most complete tests and extension of the predictive capability would be enabled by a new and dedicated facility, described below.

Once ITER begins experiments in about 10 years, it will provide important new information. U.S. research on core-edge integration will be valuable in optimizing scenarios and ensuring success of the ITER Q=10 and steady-state missions. Recent JET ILW results highlight that we must learn how to optimize these integrated solutions. It is important that we maintain a strong program to enable these contributions and position our scientists for a leading role in this crucial area for fusion. ITER results will in turn provide unique information on pedestal and SOL profiles in parameter spaces (opaque SOL, low collisionality, high fraction of density limit) which are not accessible in smaller-scale experiments, and validate the enhanced predictive capability. ITER will assess the capability of its selected divertor configuration, in combination with impurity seeding, to handle high heat fluxes. However, it will not have the flexibility to test other configurations, which as noted in PRD B will likely be needed for DEMO.

VII. 3. 3. Explore pedestal optimization and compatibility with boundary solutions using upgraded divertors, plasma-facing materials and actuators in existing experiments,

Some of the scientific questions described above will require significant upgrades to present U.S. tokamaks to explore experimentally. An example is the effect of low recyling due to lithium PFCs. This requires that a large portion of a device be covered with low recycling materials. The small LTX tokamak already has lithium PFCs, and as noted, has demonstrated enhanced global confinement[16]. The addition of core, and especially pedestal, profile diagnostics, and of core auxiliary heating and fueling to avoid the necessity of gas puffing, would provide better tests and understanding of the benefits of low recycling. An upgrade to allow diverted operation would also be valuable. Tests of lithium PFCs in a larger, higher confinement device are envisaged on NSTX-U. NSTX had considerable, largely positive, experience using modest amounts of lithium via evaporative coatings; effects on the pedestal profile and ELMs are well documented. Studies of the impact of reduced recycling on pedestal performance will be extended on NSTX-U with higher plasma parameters, and progressively more extensive use of lithium. This includes more uniform application of lithium coatings via additional evaporators, testing coatings of high-Z materials (as opposed to carbon that has been explored in other devices), and eventually testing flowing lithium systems as needed for steady-state.

Both NSTX-U and DIII-D tokamaks are planning near term tests of high-Z PFCs in localized regions. This will give information on local impurity sources and transport. Addition of systems to inject high-Z impurities would also be helpful in answering questions about impurity transport in the pedestal and core. In the medium term, conversion of all PFCs to reactor-relevant high-Z could be considered and would be valuable from a PMI perspective. However, this would represent a major change with implications for all aspects of tokamak operation and scenarios. In NSTX-U, it is envisaged that high-Z PFCs could be a substrate for liquid lithium, which would provide a valuable comparison of these two options.

Exploration of advanced divertor geometries such as “snowflake” and X-divertors is ongoing on DIII-D and NSTX, and planned soon on Alcator C-Mod, using existing coil sets. Full optimization of such solutions, however, is likely to require modification to coil sets and/or changes to divertor-plate geometry which would represent a major upgrade. Designs, feasibility and likely impact of options for divertor modifications should be explored using improved and validated SOL and pedestal models. However, it is likely that options will be highly constrained in any existing device by the available divertor volume and restrictions of magnet sets and power supplies. Any changes in PFC materials or geometry should be made using carefully planned comparison experiments. Ideally, predictions of integrated effects would be made in advance to test our improving understanding.

Other upgrades that could be useful in optimizing core-boundary solutions include the installation and testing of actuators to actively control the transport of particles and heat in the pedestal. Since many of the techniques and regimes under consideration to control or avoid ELMs preferentially increase particle transport over thermal transport, ELM mitigation tools such as RMP coils may prove useful in this regard. Other ideas proposed, discussed in Section II.4, include active stimulation of edge modes and direct modification of the pedestal via waves. Ideas for sustainment and other actuators which promise greater compatibility with PMI solutions should also be tested, to the extent feasible, on current devices. Examples of this are ICRH and LHCD with high-field-side launch. Space restrictions in present devices make this highly challenging, however.

VII. 3. 4. Extend research to more relevant conditions and develop optimized core-boundary solutions in a U.S.-led Divertor Test Tokamak facility

Research to understand the interactions between boundary solutions and pedestal and core solutions, and to develop and demonstrate an attractive solution to this challenge, would be greatly enhanced through experiments on a Divertor Test Tokamak, i.e. a toroidal facility explicitly designed and dedicated to address PMI challenges. As detailed in PRDs B and C, such a facility would provide reactor-level parameters in the divertor and be designed with much more space allocated to the divertors, and with highly flexible coil sets, allowing a greater range of proposed divertor configurations with extended volumes for heat dissipation, to be tested. This facility would also plan from the outset to test different materials, both solid and liquid, in the divertor and main chamber. It should be designed to allow controlled and elevated PFC temperatures, which is difficult to add to an existing device, enabling the first explorations of the effect of material temperature and associated changes in fuel retention on integrated solutions.

While a small-scale DTT can be designed to match the key dimensionless and dimensional parameters expected in the divertor region of a burning plasma, it is not possible to exactly match all parameters in the pedestal and core plasmas. As discussed above, ITER will eventually provide unique information in that regard. Nevertheless, it should be a goal to access regimes that approximate more closely than is currently possible the conditions expected in future fusion devices, and to test boundary solutions in conditions which extrapolate to burning plasmas. This would test key components of the physics understanding and predictive capability developed in the above campaigns on existing facilities, and more fully address the key science issues. Desired features include for example:

- Low fueling within the pedestal, enabled by high plasma density, to test effect on density profiles.
- High heat flux, allowing tests of conditions where a large fraction of heat must be dissipated by radiation.
- High confinement regimes with low normalized collision frequency pedestals and without large ELMs.
- Improved actuators for sustainment and pedestal optimization.

Since PMI and core-edge integration would be the primary mission of a DTT, diagnostics in the boundary and pedestal would need to be extensive, including and going beyond those proposed above for current devices. As discussed in PRDs B and C, the experimental program would systematically explore a range of divertor and main chamber solutions that had been found promising in present devices, and include some that had not been possible to test. While there is risk that some will prove deleterious to core plasma scenarios, such risks are acceptable in a dedicated program. Each potential solution must be pushed to the limits of its power and particle handling (e.g. complete detachment), allowing a full assessment of its potential and of the physical mechanisms underlying degradation of the pedestal and core plasma when they ultimately occur.

While a device with the mission described here was included in the European fusion roadmap[17], there is no present plan for such a facility in the EU or elsewhere. The United States has an opportunity to lead in solving the PMI challenge, including core-edge integration. Planning and design should proceed as a national effort, starting now and incorporating further information from current experiments as it becomes available. An exciting experimental program could commence in as early as five years.

VII. 3. 5. Summary of expected outcomes

If fully implemented the research proposed in this chapter, and in accompanying Priority Research Directions from this Workshop, would result in at least one, and potentially several, robust solutions to the critical challenge of PMI. We expect to develop, in a timely and cost effective manner, a strong predictive capability not only for divertor and SOL physics (PRDs B and C) but for the mechanisms by which the boundary materials and parameters influence the pedestal and core plasma. This capability, validated by experimental demonstrations in relevant regimes and conditions, will allow the U.S. program to select with confidence appropriate boundary solutions for future burning plasma experiments, fusion nuclear science facilities or pilot plant, and ultimately a fusion DEMO. This would reduce the risk and cost for these large, expensive nuclear facilities.

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Chapter VIII

Crosscutting Research Opportunities

VIII. Crosscutting PMI Research Opportunities

VIII. 1. Introduction

The broad nature of the PMI topic encompasses numerous research challenges, many of which involve multifaceted, multidisciplinary science aspects. During the course of the PMI community workshop and subsequent discussions, several recurring research topics emerged as overarching and compelling research opportunities that transcended the scope associated with individual priority research directions.

The crosscutting panel identified a total of four high-importance crosscutting research opportunities that would advance PMI science across multiple Priority Research Directions:

1. Enhanced exploitation of existing machines for PMI issues;
2. Examine long-pulse PMI science issues under reactor-relevant conditions of high accumulated plasma and neutron fluences;
3. Understand the science of liquid surfaces at reactor-relevant plasma conditions and examine the feasibility of liquid PFC solutions; and
4. Develop integrated plasma-material solutions in a purpose-built Divertor Test Tokamak.

The contextual relationships between the five PRDs and the four crosscutting research activities are shown in Figure VIII-1.

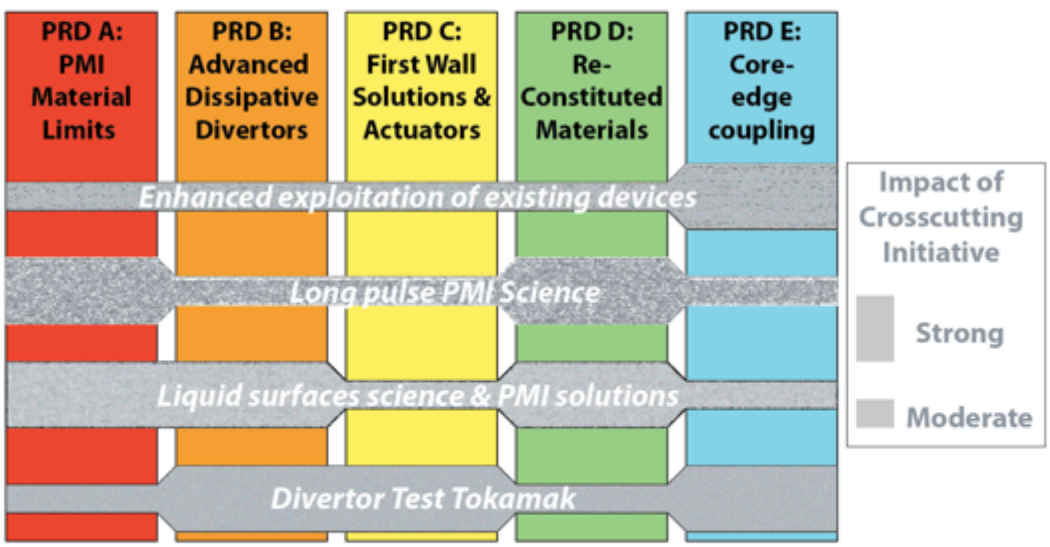

Figure VIII-1: Schematic scientific relationships between the four crosscutting initiatives and the five priority research directions. Shorthand descriptive titles are listed for the PRDs (vertical bars) and the crosscutting activities (horizontal bars).

These crosscutting activities collectively represent a new opportunity for a national program with world leadership in assessment and solution of fusion's critical boundary/PMI issues. The first three of the crosscutting research opportunities listed above involve relatively independent activities that should significantly improve our understanding of PMI issues during the coming decade. The final listed crosscutting research opportunity involves integration of this newly obtained knowledge to obtain comprehensive PMI understanding in a purpose-built tokamak. These crosscutting research opportunities are discussed in more detail in the following.

VIII. 2. Crosscutting research activities

CC-1: Enhanced exploitation of existing machines for PMI issues

The fusion community has invested substantial resources in its world-class facilities, which includes both single and multi-purpose facilities (commonly referred to as "test stands" e.g. PISCES, TPE), as well as toroidal confinement facilities (e.g. DIII-D, NSTX-U, Alcator C-Mod, MST). While the cutting-edge discovery science from these devices is impressive and impactful, additional resources toward full utilization of these facilities are needed to develop deeper understandings of relevant PMI phenomena. Furthermore, one overarching science aspect that arose from the workshop discussions is that it would be highly beneficial to assemble multidisciplinary research teams to address multiple aspects of the complex PMI issue on existing or modified confinement facilities. This multidisciplinary activity would build on past knowledge that has been primarily achieved by focusing on either plasma physics or materials science aspects, rather than a holistic perspective.

While deliberating on potential activities within each PRD, a range of actions and their impacts via enhanced utilization of the range of facilities described above, including modest upgrades, were identified. The most common prospective actions in multiple PRDs involved our toroidal confinement facilities because they enable integrated PMI and fusion physics tests, thereby motivating their inclusion as a stand-alone crosscutting research opportunity. Three general classes of actions are outlined below: comprehensive diagnosis, targeted facility upgrades, and enhanced resources for PMI studies, including both manpower and run time.

Comprehensive diagnosis for model development and validation: in all PMI areas, from material surfaces to the top of the pedestal, the need for substantially more measurements, including higher spatial and temporal resolution, is described in the PRD chapters. The boundary layer plasma containing regions of open and closed magnetic field lines is at least a 2-D space (along and across magnetic flux surfaces) and time problem, with parameters varying by orders of magnitude. Moreover, the presence of 3-D magnetic error fields, either intrinsic or imposed for amelioration of transient events, extends the problem to full 3-D. Additionally, the plasma transport rates on the open field lines, typically on a sound-speed time scale, distorts the typical Maxwell-Boltzmann distribution function (referred to as "kinetic effects"). Resolution of these kinetic effects requires measurement of the plasma velocity components, both along and across magnetic flux surfaces, in judicious locations. Also the zone of intense PMI occurs in a few mm of the material surfaces; in these locations, measurements of the properties of the other states of matter: solids, liquids, and gases are needed. Finally, the properties of the constantly evolving material surface require measurement; these are presently done either between discharges in-situ, or in separate facilities altogether, requiring in-vacuo transfer of the test samples. All of these activities are needed for continued PMI model development and validation. At present we dedicate the resources to measure only a small fraction, ~1-10 percent, of these desired plasma and materials properties.

Targeted facility upgrades: As we project the power exhaust challenges for a reactor, it appears that new PMI solutions will be needed for power and particle exhaust, and that their compatibility with attractive core plasma scenarios must be assessed. These solutions include development and testing of innovative divertor topologies, new plasma-facing materials (both solids and liquids, in the presence of gaseous cushions to ameliorate the PMI), and new ways to manage the intense PMI on internal components used for plasma diagnosis and control. Several of these solutions could be done in the DIII-D, NSTX-U, and Alcator C-Mod facilities, albeit at short pulse lengths < 10 sec. The long-pulse issues and the use of a dedicated, flexible divertor test facility are each discussed in separate crosscutting opportunities in subsequent sections of this report.

Enhanced resources for PMI studies – more people and run time: The resources currently dedicated to PMI studies on existing major fusion plasma facilities and single-purpose facilities are not commensurate with either the needs or the scientific opportunities associated with this important fusion science issue. For meaningful progress, an increase in the people focusing on boundary physics, along with dedicated experimental time for boundary physics studies, is necessary. A portion of this experimental time should be dedicated to coordinated multi-disciplinary experiments across facilities, as each facility can access unique dimensional operational regimes, with some overlap in dimensionless quantities. This not should be misconstrued as more "business as usual": extending studies across devices while examining simultaneously plasma and materials behavior will lead to seminal discovery science and deeper insight into fundamental PMI phenomena.

Conducting these research lines will provide the foundation for in-depth PMI model validation and development. This will require both an enhanced analysis effort, as well as increased emphasis on model-data comparison and subsequent model upgrades. An additional parallel FES workshop, the "Integrated Simulation for MFE Science Workshop", is developing a strategy that will progressively improve simulations of the structure and evolution of the coupled edge and boundary plasma system, including models for in-vessel components used, e.g. for heating and current drive, and transient events originating from specific plasma operational regimes. The validation effort will strive to identify "what is known and what is not known" to guide model development. Taken together, this set of actions will help develop the predictive capability that is eventually sought.

<u>CC-2:</u> Examine long-pulse PMI science issues under reactor-relevant conditions of high accumulated plasma and neutron fluences

The development of a fusion reactor or a fusion nuclear science facility will require mastering the science of PMI and the development of PFCs that exhibit unprecedented erosion resistance and/or self-healing capability during prolonged exposure (>10^6 sec) to high particle/heat fluxes and intense D-T fusion neutrons. The lifetime of PFCs, particularly in the divertor, could impact the availability of a fusion reactor, and hence its economic viability. In addition, PMI impacts the performance of the core fusion plasma during long pulses, for example through the release of impurities originating from dust expulsion of disintegrating solid surfaces, leading to dilution of the plasma fuel and radiative power losses. Even before the lifetime of a PFC is reached, stringently controlled in-vessel inventory limits of dust and tritium could suspend reactor operation if exceeded. An improved understanding of the degradation mechanisms associated with long-pulse PMI is needed to identify potential PFC materials and operational regimes.

Extended plasma exposures of divertor PFCs to reactor-relevant ion fluences in the range of 10^{30}-10^{31} m^{-2} will require acceptable net physical and chemical erosion yields (e.g. $Y\sim10^{-6}$ for tungsten). The strong coupling of plasma and surface in those high-density divertor plasmas will likely change the evolution of the surfaces in a non-linear way. Large-scale surface restructuring due to erosion and re-deposition and morphology changes will take place. In addition, at high temperatures the growth of interconnected nano-tendril layers, or "fuzz", on refractory metals (e.g. tungsten) will occur. For liquid metals the complexity of the surfaces is increased not only due to the plasma impact but also due to potentially strong electromagnetic forces interacting with the liquid metal. The incoming particle fluxes (hydrogen isotopes, helium, impurities and neutrons) will change the composition of the material surface due to their implantation, induced transmutation, preferential sputtering or segregation; the relative magnitudes of these phenomena are expected to be different for solid and liquid PFCs. Finally, long-range material migration from the main chamber of the fusion device will in some places lead to net deposition, adding to the complexity of these re-constituted surfaces.

The stability of these reconstituted surfaces and nanostructures could determine large-scale erosion processes, which should be avoided since they can have detrimental effects on the plasma core performance, or even cause disruptions, due to instantaneous release of a large particle source. The erosion of the deposited surfaces will be altered due to the morphology of the surface and the composition of the surface layers. Loosely bound deposition layers or other surface morphology changes will produce increased surface temperatures due to thermal conductivity degradation, which will change erosion yields and might lead to melting. The surface area will increase by the increased roughness, possibly leading to effectively lower ion flux densities to the surface, which could in itself have an effect on the chemical erosion yield of, e.g. carbon. Moreover, it is unclear how the plasma will interact with surface topologies, which are not in direct line-of-sight in those complex 3D-nanostructures. Neutron irradiation of material samples to high DPA (as would appear after prolonged operation in a fusion reactor) might change the PMI processes due to direct and indirect effects. Currently, most PMI investigations have been performed on samples with no or rather low (~1 DPA) neutron damage levels whereas in a fusion reactor, materials will experience much higher damage levels (50 DPA or greater). Understanding the fundamental processes leading to the formation of the complex PMI surface architectures (including the influence of neutron irradiation) might open routes to control the surface morphology changes by acting on the plasma parameters or their composition in front of the surface.

Neutron irradiation can produce pronounced radiation induced solute segregation and precipitation in solids, in addition to the important changes in physical (e.g., thermal conductivity) and mechanical (e.g., fracture toughness) properties. This will change the erosion yields of the materials significantly. The microstructure of solids is also changed by the neutron irradiation such that the trapping of tritium in the solids becomes larger (particularly at intermediate to high temperatures where cavity formation occurs). High helium/DPA-ratios representative of the fusion neutron spectrum, in turn will promote cavity formation, thereby having a direct effect on tritium retention. These radiation-induced microstructural changes in the material would be expected to influence surface topography evolution (e.g., fuzz formation, etc.) and a variety of other effects. Furthermore, the microstructure changes, the hydrogen/helium embrittlement due to neutron irradiation could even lead to macroscopic erosion processes by dust production and leading to a reduced lifetime of the PFCs.

High-flux, high-power linear plasma devices are best suited to investigate the evolution of the surfaces in high-fluence plasmas. Advanced linear plasma devices will in addition offer the possibility to study the synergistic effects between neutron irradiation damage at high doses and PMI. The current data on deuterium ion exposures (as a surrogate for tritium) are limited to a fluence of $\sim 10^{28}$ m^{-2} and low DPA levels. Increasing the knowledge base to significantly higher fluence and DPA requires a new high flux steady-state linear plasma device.

Long-pulse toroidal devices will provide additional information on long-range material migration and deposition in the main chamber and divertor. In front of RF antennas the erosion of PFCs might be exacerbated due to RF sheaths. Due to the connection of magnetic field lines from the RF antenna to other PFC surfaces (e.g. in the divertor), toroidal- and poloidal-localized modifications in sputtering may occur at PFC locations far removed from the RF antennas. Long-pulse tokamaks, as well as linear plasma devices, will give very valuable information on the effect of RF sheath parameters on PFC erosion. Our international partners have chosen to develop long-pulse capabilities, e.g. in EAST, WEST, KSTAR, JT-60SA and W7-X; targeted international collaborations in these areas appear to be the most efficient way to access these particular physics issues. For example, collaboration on “fuzz” growth on tungsten at elevated temperatures in long pulse may be accessible on EAST and WEST.

Very long-pulse operation in tokamaks might also reveal some of the potential disintegration of the surface films/structures (e.g. due to unipolar arcing, delamination) in solid PFCs, which will lead inevitably to dust production and possible mobilization. The effect of dust on confinement and stability of the core plasma can be explored and documented. As a result, improved plasma scenarios can be developed to optimize core-edge integration with acceptable impurity release and transport into the plasma core.

While most of this PMI workshop deals with steady-state heat and particle flux exhaust, transients such as edge-localized modes (ELMs) and disruptions cannot be neglected. In the FES workshop process, a separate workshop focused on control of transients operated independently from this one. Repeated thermal transients induced by plasma transients and their relatively shallow power deposition can lead to surface cracking in solid PFCs. Cracks can provide shortcuts to the surface for outgassing, influence recycling properties, and, if sufficiently dense and interconnected, lead to material ejection. To address these issues, it is essential to determine how cracking depends on neutron embrittlement and the magnitude of transients. Even refractory metals, our leading solid PFC candidates, might melt during transients. As a consequence, melt-layer movement and droplet formation are important research topics to be studied for both solid and liquid PFCs during transients. Data on the material response to transients of up to 10^8 ELMs needs to be examined to minimize the impact on confinement and stability.

Simulating ELM-like transients as they will occur in a fusion reactor is a challenge. Existing toroidal devices are unable to deliver the anticipated power loads; however, dedicated pulsed power test stands (e-beams, lasers, plasma guns) can access the energy flux density. Linear plasma devices have proven to be able to provide useful information in testing materials exposed to steady-state plasmas and transient heat and plasma loads simultaneously (e.g., Magnum-PSI in the Netherlands). Simultaneous testing of periodic ELM-like heat flux in conjunction with steady state reactor-relevant plasma and heat flux on neutron irradiated material samples requires a new linear plasma facility.

CC-3: Science and feasibility of liquid PFCs

The daunting PFC heat and particle flux environment, combined with pronounced radiation damage degradation concerns, might ultimately prove to require operational conditions for solid PFCs that are unsuitable for high-power, high-duty-cycle fusion reactors. Liquid surface PFC options are attracting increasing attention as an alternative to conventional solid PFCs. The range of potential liquid PFCs include chemically active liquid metals such as lithium (which also provides potential benefits in terms of impurity gettering, low hydrogen fuel recycling, and low-Z to avoid core radiative collapse from high-Z impurity accumulation), relatively inert liquid metals such as tin-lithium alloys, and low electrical conductivity fluids such as liquid salts. As noted in the PRD discussion sections, a variety of potential liquid PFC concepts may be envisioned including quasi-stagnant and rapidly flowing liquids and evaporative thin films. Promising scoping results have been recently obtained using lithium in several tokamaks, where in many cases improved plasma performance was observed.

Overall, the scientific understanding of potential feasibility issues and overall technological maturity of liquid PFC concepts is relatively limited. As a consequence, substantial improvement in scientific understanding may be achieved by judicious utilization of analytical and computational modeling along with scoping experiments on small- to mid-scale test frames, and in existing toroidal machines. Some of the key issues to be investigated are described in the following paragraphs.

Although many of the liquid PFCs offer the prospect of significantly improved divertor performance, the heat and particle flux limits that might be achievable in liquid PFCs under steady state and transient conditions need to be quantified. Further, as noted in PRD E, the compatibility of liquid PFCs with high-performance pedestal and high core confinement needs to be examined. It is possible that some liquid PFCs might actually lead to improved plasma performance, but this needs to be verified.

As described in several PRDs, significant additional research is needed to understand the temperature dependencies of liquid PFCs with respect to evaporation rates, vapor pressures compatible with good core plasma performance, and retention/recycling of hydrogenic and other impurity species. These effects may depend sensitively on incident plasma parameters, liquid layer thickness and flow conditions, and surface conditions of the underlying substrates including surface roughness and wetting properties. Further, currents in the boundary scrape-off-layer from thermo-electric and transient/ halo effects will interact electromagnetically with liquid metals, and magnetohydrodynamic effects could also be potentially important for rapidly flowing liquid metal systems.

As discussed in PRDs A and C, additional research is needed to develop several self-consistent reactor concepts incorporating liquid PFCs, including proposed details of liquid and substrate materials, operating temperatures, flow rates, pumping, and retention/impurity handling requirements. To the extent possible, such integrated systems should be prototyped in existing toroidal facilities and any new toroidal facilities dedicated to divertor and/or liquid PFC testing. Any integrated testing in toroidal systems must first be prototyped in dedicated test stands and supported with surface and material science laboratory facilities. Substantial computational capabilities are also required to understand the potentially complex interacting effects of chemistry, erosion/re-deposition, evaporation, radiation, retention, and MHD and electromagnetic effects including free-surface flowing liquid metals.

As discussed in PRD D, utilization of liquids as plasma-facing materials could introduce new material migration mechanisms in addition to well-known particle sputtering, reionization and deposition sequences. On a global scale, many of the solid PFC redeposition issues would appear to be more tractable for liquid PFCs (e.g., mixed solid layer stratification with potential for entrapment of hydrogen isotopes). However, droplet formation could lead to relatively pronounced PMI processes in spatially localized regions.

Tritium transport and retention mechanisms need to be explored for liquid PFCs. For example, although low hydrogen fuel recycling may improve plasma performance, if the amount of hydrogen isotopes in the liquid is too high then safety issues associated with tritium inventory and potential release to public may arise (under prolonged normal operational and accident scenarios). Further, the relatively low tritium burn-up fraction in envisioned reactors could require substantial tritium reprocessing/recovery from recirculating lithium if the lithium is operated in a temperature regime in which retention is high.

A major limitation of high-Z solid metallic PFCs is sensitivity to off-normal events. For example, transient high heat flux from ELMs can lead to local melting followed by re-solidification and formation of non-uniform surface features more prone to local melting during subsequent thermal transients. Liquid PFCs have the potential to avoid this problem entirely by already operating in the liquid state. However, a variety of engineering considerations need to be resolved for utilization of liquid PFCs. For example, sufficiently thick and uniform liquid coverage must be provided to protect underlying substrates and (for low-Z liquids) shield the plasma from high-Z substrate impurities. The liquid must also be circulated and reprocessed at a sufficiently high rate to remove any entrained fuel gasses and impurities. High mass flow rates are likely required for thicker liquid layers or vapor boxes used for divertor power exhaust. The transport of flowing and potentially chemically reactive liquids into and out of high-temperature magnetized vacuum chambers could be a daunting engineering challenge, and has not been demonstrated in any representative device or test stand. Further, potentially deleterious materials effects including stress-induced liquid metal embrittlement of the substrate should be investigated and ameliorated using dedicated test facilities prior to any large-scale deployment of hot flowing liquid metals in toroidal facilities.

Liquids offer the potential to provide a replenishable and more resilient PFC for fusion systems. Compared to the much larger world-wide effort investigating high-Z solids, liquids are comparatively unexplored. The liquid PFC research area is therefore ripe for new scientific discoveries, could be an area of U.S. leadership in the world program, and if successfully developed, could significantly improve the attractiveness of future fusion devices.

CC-4: Develop integrated plasma-material interaction solutions in a purpose-built divertor test tokamak

The scientific rationale for a new facility dedicated to PMI research has been documented in numerous community reports over the last decade. The 2009 ReNeW report called for action to "Develop design options for a new facility with a DEMO-relevant boundary, to assess core-edge interaction issues and solutions." European researchers, in their 2012 fusion roadmap study, concluded, "Since the extrapolation [of plasma exhaust solutions] from proof-of-principle devices to ITER/DEMO based on modelling alone is considered too large, a dedicated test on specifically upgraded existing facilities or on a dedicated Divertor Tokamak Test facility will be necessary."

The present study has further advanced the case for the United States to take the initiative in moving forward with this critically needed DTT facility. Five, largely independent, Priority Research Directions (PRDs) have been carefully developed, based on detailed analyses of the critical issues, the key scientific questions, and the needed research actions. The PRDs generally call for augmentation of the program resources devoted to PMI research, including enhanced exploitation of existing U.S. facilities to fully exhaust their potential to advance the field, as well as collaboration on existing overseas facilities. Still, all five PRDs foresee that a new facility will be needed in order to provide greater dedicated volume and design flexibility for testing advanced divertor solutions, and to produce heat and particle flux densities closer to fusion power plant conditions, than are available even with upgrades of existing facilities. The DTT is thus a crosscutting opportunity for PMI research. Here we highlight the overarching scientific need for the DTT based on descriptions from the accompanying PRDs, while simultaneously pointing out its generic features.

As implied by the name, divertor science lies at the heart of the DTT; yet the PRDs show that it is the inherent integration of the complex boundary plasma system that truly drives the need for a DTT. Starting with the divertor, as pointed out in PRD-B, advances in divertor magnetic geometries, heat dissipation mechanisms, and materials will all be needed to function effectively under the extremes of DEMO conditions. A number of promising candidate divertor concepts have been identified, and future innovations may emerge over the coming decade. To examine the feasibility of these concepts, a DTT facility must provide the flexibility in its design to accommodate and effectively test multiple divertor configurations. This design requirement is driven by that fact that many of the divertor concepts require a larger fraction of the device's volume than is currently devoted to the divertor in presently existing devices. Importantly, the exploration of innovative divertor concepts has important implications for science issues outside the divertor volume. The entire scrape-off layer is inextricably linked to its interactions with not only the divertor, but also the main-chamber wall and its associated actuators, for example RF launching structures. As noted in PRD-C, a DTT would support the development of main-chamber and RF actuator solutions for power plants, using design flexibility for first-wall material and the allocation of dedicated space for innovative actuators. PRD-E observes that the PMI solutions explored in PRD-B and PRD-C must be made compatible with a high-performance core plasma pedestal. Because the pedestal spans the region from the core plasma to the SOL, pedestal science is inherently coupled to boundary and PMI, and requires a confined plasma for proper assessment. The DTT, with its enhanced and modified divertor/first wall, provides a key opportunity to explore pedestal physics in an integrated environment. Therefore, one readily sees a central theme arising from PRD's B, C and E: a DTT is needed with significant dedicated volume and geometric flexibility to address the coupled boundary and pedestal plasma issues.

Another theme arising from the PRDs is the strong desire for a DTT to access a wider range of reactor-relevant conditions than presently available, both in terms of plasma parameters and plasma-facing material choices. In order to assess innovative dissipative/detached divertor solutions, PRD-B indicates that the DTT must provide divertor target loading and plasma conditions that more closely approach the absolute parameter range expected in power plants, while also accommodating interchange of a variety of both solid and liquid divertor target materials. This requirement from PRD-B is strongly linked to the theme of PRD-D in assessing the science of evolving, reconstituted materials. The plasma facing materials are constantly and vigorously remade by the intense interactions with the plasma, posing serious challenges in the extrapolation to ~year long DEMO pulses. A DTT with reactor-relevant SOL and divertor plasmas fills a niche for PRD-D by examining the effect of material migration on reconstituted surfaces on shorter timescales but under reactor relevant power exhaust conditions in an integrated tokamak scenario. The DTT is complementary to high-power linear plasma devices that can achieve local reactor-like conditions, but which do not have integrated tokamak migration patterns, plasma gradients, etc. The effect of plasma-facing material in the divertor/first wall is strongly linked to the science issues called out in PRD-D and E. For instance, it is already known that the choice of plasma-facing material strongly can affect the pedestal, the science topic of PRD-E. With respect to PRD-C, a DTT would support the development of main-chamber and RF actuator solutions for power plants in an environment with reactor-like SOL plasma conditions.

All the PRDs also emphasize that a DTT, while critically important for empirical demonstrations of integrated PMI solutions, must simultaneously advance the underlying science. For example, PRD-E calls for an intense campaign to develop better predictive capability for pedestal physics through enhancements in diagnostics, modeling, and utilization of both existing devices and a DTT. With such tools, the DTT provides a key opportunity to test and expand predictive capabilities under conditions closer to those expected in a reactor, for example with an advanced, large-volume dissipative divertor or with liquid plasma-facing surfaces. The advancement of divertor and SOL physics, dealt with in PRD-B and C, are inherently built into the mission of a DTT. With respect to understanding reconstituted surfaces, PRD-D states that while the DTT should achieve reactor-relevant edge plasma conditions and reactor-relevant material temperatures, likely using actively heated components, it must also deploy a large suite of in-situ/in-vacuo material diagnostics for both solid and liquid surfaces.

Taken together, this examination of the crosscutting opportunities underscores the richness of PMI science, in terms of the variety of physical processes and the broad range of parameters and materials that exist in the region bound by solid or liquid walls on one side and high-temperature plasma on the other. A new DTT facility, with a design optimized for innovative PMI research, would extend PMI science into new regimes and provide world-leading opportunities for scientific discovery in this research area so critical to fusion's viability

The next steps in making a DTT become a reality are to develop design concepts for implementation and formally document the mission-need case. In qualitative terms, the general facility requirements are already clear from this study, namely flexibility in geometry/materials, and edge/divertor parameters approaching those expected in fusion reactors. In addition, the facility is understood to be a tokamak, since only tokamak PMI research has advanced to a stage of both readiness and need for such a specialized facility. Several conceptual designs for a DTT have been put forward during this and previous studies, and all have been extensively discussed. While selection of a particular design is clearly premature, those designs already put forward generally indicate technical and scientific readiness. In summary, the scientific motivation, basic requirements, and plausible approaches already exist and provide the starting point for a focused DTT design activity.

The task for the design activity is to develop the high-level requirements and a set of design options at a pre-conceptual level to permit tradeoffs to be made among the different possibilities. Power density and geometric flexibility have been emphasized in this study, but other attributes important to PMI research, such as pulse length, component operating temperatures, and plasma heating and sustainment methods, must be established and rigorously justified. The mission-need case is already well developed, but additional information such as international perspectives, management approach, and rough cost and schedule estimates may be required to support a formal DOE determination of mission need sufficient to move ahead with conceptual design and project planning. Developing this information is a task for the design team.

In summary, the need for a dedicated facility to allow this field to continue moving forward in the next decade is compelling and provides a world-leadership opportunity for the U.S. Fusion Energy Sciences program. A U.S. DTT would attract users and collaborators from around the world. The research program supported by a DTT would influence the direction of international fusion reactor development and provide unique opportunities to advance the science of plasma and materials.

Chapter References:

(None)

Chapter IX

Appendix

IX. 1. Appendices – Charge letter from D.o.E. Fusion Energy Sciences (p.1)

Department of Energy
Washington, DC 20585

February 9, 2015

Dear Colleagues,

The Fusion Energy Sciences (FES) program is planning to hold a series of technical workshops this year in order to seek community engagement and input for future program planning activities. This letter describes the workshops, their objectives, and some of the organizational arrangements.

I had initially mentioned such workshops in my talk at the University Fusion Association Evening Session at the 56th Annual American Physical Society Division of Plasma Physics Meeting in October and also in my presentation at the Fusion Power Associates Annual Meeting in December. Subsequently we had a discussion in December with community leaders about these workshops, which was very helpful.

In addition, Congress has indicated its interest in scientific workshops for the FES program with the following language in the FY 2015 Appropriations Act: *"The Office of Science is further directed to seek community engagement on the strategic planning and priorities report through a series of scientific workshops on research topics that would benefit from a review of recent progress, would have potential for broadening connections between the fusion energy sciences portfolio and related fields, and would identify scientific research opportunities. The Department is directed to submit to the Committees on Appropriations of the House of Representatives and the Senate not later than 180 days after enactment of this Act a report on its community engagement efforts."*

The workshops are being planned in four areas. These are listed in the table below, along with the names of the chairs and co-chairs and the federal points of contact:

Workshop	Chair / Co-Chair	Federal POC
Integrated Simulations for Magnetic Fusion Energy Sciences	Paul Bonoli (MIT) / Lois Curfman McInnes (ANL)	John Mandrekas (FES), Randall Laviolette (ASCR)
Plasma-Materials Interactions	Rajesh Maingi (PPPL) / Steve Zinkle (U Tennessee)	Peter Pappano (FES)
Transients	Chuck Greenfield (GA) / Raffi Nazikian (PPPL)	Mark Foster (FES)
Plasma Science Frontiers	Fred Skiff (U Iowa) / Jonathan Wurtele (UC Berkeley)	Sean Finnegan (FES)

The first three of these workshops correspond to critical areas identified in the 2014 FESAC Strategic Planning and Program Priorities report as areas where increased emphasis would be beneficial as the fusion program moves further into the burning plasma science era:

- Developing an experimentally validated integrated predictive simulation capability that will reduce risk in the design and operation of next-step devices as well as enhance the value of participation in ITER,
- Understanding and controlling deleterious transient events that can disrupt plasma operation and damage fusion devices, and
- Addressing the extreme harshness of the burning plasma environment at the plasma-materials interface and finding solutions.

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IX. 1. Appendices – Charge letter from D.o.E. Fusion Energy Sciences (p.2)

These three areas are very challenging scientifically and also offer opportunities to build upon U.S. strengths and potential partnerships with other Office of Science programs.

The fourth workshop area is that of Plasma Science Frontiers, which is comprised of the sub-areas of General Plasma Science, High Energy Density Laboratory Plasma, and Exploratory Magnetized Plasma. Given the FES stewardship of plasma science and the fact that Plasma Science Frontiers is a new category in the restructured FES budget, there is high value to holding a workshop in this area. Furthermore, given the very broad and diverse nature of this scientific area and the fact that two of the sub-areas have not yet had the benefit of a research needs type of workshop, the plan is to hold a series of two workshops in this area: the first one to identify compelling scientific challenges at the frontiers of plasma physics, and a second workshop to identify research tools and capabilities that exist presently, as well as the general requirements necessary to address these challenges in the next decade.

The objectives of the workshops being planned will depend on their specific topical areas. In general, the objectives will likely include elements from among the following: (1) review of progress and an update about new developments since the last time organized community input was obtained, (2) identification of gaps and challenges, along with specific parameters that would need to be achieved for addressing such gaps, (3) discussion of near- and long-term research tasks, such as experiments that could be performed on existing facilities, (4) descriptions of upgrades to existing facilities and diagnostic capabilities that would enable or enhance the research tasks, (5) identification of linkages to associated research areas, (6) descriptions and analysis of potential new activities for addressing the gaps and challenges, and (7) identification of areas for which modeling and simulation could be impactful.

Enclosed with this letter are four "one pagers" that describe the background, objectives, and organization for each of the planned workshops.

Let me express our sincere appreciation to those who have agreed to assume leadership roles as chairs and co-chairs. We recognize that organizing these types of workshops requires a lot of time and effort, and it is our intention to help them in any way that we can. Each workshop has an FES point-of-contact person and, in the case of the integrated simulations workshop, we are pleased to partner with the Advanced Scientific Computing Research (ASCR) program within the Office of Science, which has provided an additional point-of-contact person.

We are counting on your assistance in making these workshops successful.

If you have any questions about the workshops, please feel free to contact any of the POCs.

Sincerely,

Edmund J. Synakowski
Associate Director of Science
for Fusion Energy Sciences
Office of Science

Enclosures

IX. 1. Appendices – Charge letter from D.o.E. Fusion Energy Sciences (p.3)

Workshop on Integrated Simulations for Magnetic Fusion Energy Sciences
Chair: Paul Bonoli (MIT), Co-Chair: Lois Curfman McInnes (ANL)

Background
Motivated by the opportunities afforded by the extraordinary advances in high-performance computing, the Fusion Energy Sciences (FES) program, in partnership with the Advanced Scientific Computing Research (ASCR) program, has supported a number of world-leading multi-institutional and interdisciplinary efforts in the last decade under the Scientific Discovery through Advanced Computing (SciDAC) program, addressing grand challenge problems in fusion energy sciences. While most of these efforts treat phenomena in relative isolation by taking advantage of scale separation, FES and ASCR have also supported efforts that have taken the first steps toward integration, recognized as the next necessary undertaking for developing credible predictive capability. FES and ASCR solicited community input several times during the last decade, including the 2007 Fusion Simulation Project workshop, a two-year fusion simulation planning study, the 2009 ASCR-led workshop on Scientific Grand Challenges and the Role of Computing at the Extreme Scale, and others. In addition, experimentally validated integrated simulation has been among the top recommendations of several Fusion Energy Sciences Advisory Committee (FESAC) and other community studies, including the Greenwald report, the ReNeW report, and the recent FESAC report on strategic planning.

Objective
The main goal of this workshop will be to review recent progress and identify gaps and challenges in fusion theory and computation directly relevant to the leading scientific opportunities for integrated simulations identified by previous community studies. In addition, the workshop should reassess these opportunities and adjust or broaden them appropriately, taking into consideration recent progress and using the criteria of urgency, leadership computing benefit, readiness for progress within a ten-year time frame, and world-leading potential.

The leading scientific opportunities identified by previous studies have been remarkably robust and consistent: the prediction, avoidance, and mitigation of major disruptions and the physics of the plasma boundary, with Whole Device Modeling as the long-term goal. These scientific priorities are also consistent with the findings of the recent FESAC report on strategic planning. The workshop will achieve its objectives by considering recent advances, including research tools and capabilities developed by the eight FES SciDAC Centers and computational expertise at the ASCR SciDAC Institutes; advances and associated challenges in emerging extreme-scale computing hardware; recent progress in verification and validation and uncertainty quantification; "big data" issues; and the emerging needs of ITER. Crosscutting issues such as the status of measurement capabilities that are relevant to the experimental validation mission will also be addressed through coordination with the other workshops in this series.

Organization
The workshop will follow the format of the successful Office of Science Basic Research Needs series of workshops. FES and ASCR will select the chair and co-chair(s), who will define the various workshop panels and sub-panels (including any crosscutting panels), and select the panel leads. The chair, co-chair(s), and panel leads will make up the Executive Group of the workshop. The panel leads will select the panelists and any sub-panel leads. The workshop report will be written by the chair, the co-chair(s), the panel leads, and by select panelists designated as writers. Input from the entire community will be solicited during the preparation for the workshop, and participation will be open, but the total number of attendees will be limited to preserve the "working meeting" character of the workshop. A substantial amount of work via teleconferences and other means will be done prior to the workshop to allow the preparation of a draft report during the last day of the workshop.

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Workshop on Plasma-Materials Interactions (PMI)
Chair: Rajesh Maingi (PPPL), Co-Chair: Steve Zinkle (U Tennessee)

Background
Because of the importance of PMI science and the number of potential approaches to address the most relevant scientific issues, a multi-day workshop is planned in order to allow the research community to update and reassess the most critical scientific PMI questions that need to be answered (cf. MFE ReNeW report), and how best to answer these scientific questions.

The research needs for this issue are extraordinary, inviting innovation and vision for the development of solutions, while also achieving world-class scientific understanding. The recent Fusion Energy Sciences Action Committee (FESAC) strategic priorities report noted the importance of plasma-materials interactions science and proposed near term initiatives utilizing a linear divertor simulator, computation, and existing domestic and international toroidal facilities to help study this crucial area of fusion research. Previous community reports have also noted the need for a dedicated toroidal device to study PMI (cf. Research Needs for Magnetic Fusion Energy Sciences 2009, Opportunities for Fusion Materials Science and Technology Research Now and in the ITER Era 2012, Fusion Nuclear Science Pathways Assessment 2012, Prioritization of Proposed Scientific User Facilities for the Office of Science 2013), as well as calling for a linear device to unfold the science of plasma-materials interactions and boundary layer physics.

Objective
The goal of this multi-day workshop will be to engage the community of scientific experts working in the fields of materials, plasma-materials interactions, and boundary/edge plasmas and identify:

1. Compelling scientific questions in PMI that must be addressed in order to advance the field and achieve new scientific understanding and,
2. Options for addressing these scientific questions, including but not limited to new facilities, upgrades of existing facilities, validated computation, and international partnerships.

The community shall reassess the current state of knowledge and urgent scientific issues encompassed by the PMI thrusts from MFE ReNeW:

- Unfold the physics of boundary layer plasmas (Thrust 9)
- Decode and advance the science and technology of plasma-surface interactions (Thrust 10)
- Improve power handling through engineering innovation (Thrust 11)
- Demonstrate an integrated solution for plasma material interfaces compatible with an optimized core plasma (Thrust 12)
- Develop the materials science and technology needed to harness fusion power (Thrust 14)

Organization
The workshop to be held in the Spring of 2015 will be set up following the format of the successful Office of Science Basic Research Needs series of workshops and will serve as the primary means for broad community input. FES has selected the chair and co-chair, who will define the various workshop panels and sub-panels (including any crosscutting panels) and select the panel leads. The chair, co-chair, and panel leads make up the Executive Group of the workshop. The Executive Group selects the panelists and any sub-panel leads. The final report will be written by the chair, the co-chair, the panel leads, and by select panelists designated as writers. Participation in the workshop will be open, with a final report deadline at the end of June. A substantial amount of work via teleconferences and other means will be done prior to and after the workshop.

IX. 1. Appendices – Charge letter from D.o.E. Fusion Energy Sciences (p.5)

Workshop on Transients
Chair: Charles Greenfield (GA), Co-Chair: Raffi Nazikian (PPPL)

Background
It is well known that transient events such as disruptions and Edge Localized Modes can have deleterious effects on tokamak plasmas, with the potential to cause damage to plasma facing components and first wall structures, as well as degrading plasma performance. Although these events are generally tolerated in present tokamaks, they are predicted to have more severe impacts on ITER and future burning plasma devices. If not prevented or mitigated, these events will have unacceptable impacts on the operational availability of these devices and shorten the lifetime of the in-vessel components. It is critical to develop the means to minimize these events and their consequences when they do occur.

The fusion community, through the comprehensive ReNeW process (*Research Needs for Magnetic Fusion Energy Sciences*, 2009), developed a proposed research thrust in this area – "Control transient events in burning plasmas". Subsequent Fusion Energy Sciences Advisory Committee (FESAC) reports (*Report of the FESAC Subcommittee on the Priorities of the Magnetic Fusion Energy Science Program*, 2013 and the *Report on Strategic Planning: Priorities Assessment and Budget Scenarios*, 2014) have endorsed this as one of the highest priority magnetic fusion research topics. Several workshops have already been held to examine in more detail the underlying physics issues and specific aspects of the ITER disruption mitigation system, and the U.S. Burning Plasma Organization (USBPO) currently has an active task force coordinating research on this topic.

Objective
Building on the ReNeW effort, other workshop results, and the ongoing USBPO disruptions task force plans, this workshop will review recent progress and identify the remaining science and technology challenges that must be addressed to demonstrate that magnetically confined tokamak plasmas with the characteristics desired for a fusion power plant can be robustly produced, sustained, and controlled without deleterious effects on the device's materials and structure. Based on thorough understanding of the remaining science and technology challenges, the workshop will identify specific research opportunities that can address these challenges in the next decade. These may include both domestic research and international partnerships and will be informed by the requirements of ITER and future burning plasma devices.

Organization
The workshop will be set up following the format of the successful Office of Science Basic Research Needs series of workshops. Fusion Energy Sciences will select the chair and co-chair(s) who will define the various workshop panels and sub-panels (including any crosscutting panels) and select the panel leads. The chair, co-chair(s), and panel leads make up the Executive Group of the workshop. The panel leads select the panelists and (if necessary) any sub-panel leads. The workshop report will be written by the chair, the co-chair(s), the panel leads, and any panelists designated as writers. A multi-day workshop will be held that will allow for a vigorous discussion of the scientific and technical issues and opportunities in this area. A substantial amount of work via teleconferences and other means will be done prior to the workshop to allow the preparation of a draft report during the last day of the workshop. Input from the entire community will be solicited during the preparation for the workshop, and participation will be open, but the total number of attendees will be limited to preserve the "working meeting" character of the workshop.

Since transient events will also be a subject of interest to the integrated simulations' effort, the activities of this workshop should be coordinated as appropriate with related activities of the integrated simulations workshop, including sharing participants and possibly establishing cross-cutting panels.

IX. 1. Appendices – Charge letter from D.o.E. Fusion Energy Sciences (p.6)

<u>Workshop on Plasma Science Frontiers</u>
<u>Chair: Fred Skiff (U Iowa), Co-Chair: Jonathan Wurtele (UC Berkeley)</u>

Background
The reorganization of the Fusion Energy Sciences (FES) budget structure in FY 2015 brings together three program elements at the frontiers of plasma science–viz., general plasma science, high energy density laboratory plasmas, and exploratory magnetized plasma. These three activities support a rich and diverse portfolio of plasma science, sharing many common intellectual threads with the potential for broadening connections between the fusion energy sciences portfolio and related fields.

Objective
The Plasma Science Frontiers (PSF) activities in FES seek to engage the community of scientific experts working in the fields of general plasma science, high energy density laboratory plasmas, and exploratory magnetized plasma in a series of two community-led workshops to identify:

1. Compelling scientific challenges at the frontiers of plasma physics, and
2. Research tools and capabilities that exist presently, as well as the general requirements necessary to address these challenges in the next decade.

The report(s) generated from these workshops will inform FES in planning and executing its strategic vision for the FES stewardship of the PSF activities, taking into consideration the recommendations from the Fusion Energy Sciences Advisory Committee [1] and the National Research Council [2].

Organization
The first workshop, "Scientific Frontiers," will focus on identifying the grand scientific challenges in plasma science. The starting point will be the six critical plasma processes that were identified in the 2007 National Research Council plasma science report [2] as being not well understood: explosive instabilities, magnetic self-organization, turbulence and transport, correlations in plasmas, multiphase plasma dynamics, and particle acceleration and energetic particles. The goal of the workshop will be to bring together input received from across the community (via one-page white papers) on updates to the state of the art and where the frontiers are since the 2007 report.

The second workshop, "Research Needs," will focus on identifying the research needs required to address scientific challenges at the forefront of plasma physics. It will specifically address existing experimental tools and capabilities, as well as future performance requirements at the intermediate scale and computational hardware and software needs.

Both workshops will follow the format of the successful Office of Science Basic Research Needs series of workshops. FES will select the chair and co-chair(s), who will define the various workshop panels and sub-panels (including any crosscutting panels) and select the panel leads. The chair, co-chair(s), and panel leads will make up the Executive Group of the workshop. The panel leads will select the panelists and any sub-panel leads. The workshop report will be written by the chair, the co-chair(s), the panel leads, and any panelists designated as writers. Input from the entire community will be solicited during the preparation for the workshop, and participation will be open, but the total number of attendees will be limited to preserve the "working meeting" character of the workshop. A substantial amount of work via teleconferences and other means will be done prior to the workshop to allow the preparation of a draft report during the last day of the workshop.

[1] "Report on Strategic Planning: Priorities Assessment and budget scenarios" (2014)
[2] "Plasma Science: Advancing Knowledge in the National Interest" (2007)

IX. 2. Appendices – List of Panel Members

SOL & divertor physics (ReNeW Thrust #9):

- **Leader/Deputy: H.Y. Guo (GA), B. LaBombard (MIT)**
- Panelists: R.J. Goldston (PPPL), I.H. Hutchinson (MIT), S.I. Krashenninikov (UCSD), J.R. Myra (Lodestar), V.A. Soukhanovskii (LLNL), P.C. Stangeby (U. Toronto), P.M. Valanju (U. Texas), X.Q. Xu (LLNL)

Advancing PMI science and innovation (ReNeW Thrust #10 and part of #14):

- **Leader/Deputy: J.P. Allain (UIUC), R.P. Doerner (UCSD)**
- Panelists: M.A. Jaworski (PPPL), R. Kolasinski (SNLL), R.J. Kurtz (PNNL), J. Rapp (ORNL), G. de Temmerman (ITER Organization), B.D. Wirth (UT-K), G. Wright (MIT)

Engineering innovations for plasma exhaust challenges (ReNeW Thrust #11)

- **Leader/Deputy: C.E. Kessel (PPPL), D.L. Youchison (SNLA)**
- Panelists: J. Blanchard (UW-M), R.W. Callis (GA), R. Ellis (PPPL), R. Majeski (PPPL), N.B. Morley (UCLA), D.N. Ruzic (UI-UC), M.S. Tillack (UCSD), S.J. Wukitch (MIT), M. Yoda (GIT)

Compatibility of boundary solutions with attractive core scenarios (ReNeW Thrust #12)

- **Leader/Deputy: A.E. Hubbard (MIT), A.W. Leonard (GA)**
- Panelists: J.M. Canik (ORNL), M. Kotschenreuther (UT-A), R. Majeski (PPPL), P.B. Snyder (GA), J.L. Terry (MIT), E.A. Unterberg (ORNL), J.R. Wilson (PPPL)

Crosscutting group to facilitate discussions, identify high leverage opportunities

- **Leader: S. Zinkle (UT-K)**
- Panelists: D.N. Hill (LLNL), D.L. Hillis (ORNL), R. Maingi (PPPL), J.E. Menard (PPPL), G.H. Neilson (PPPL), D.G. Whyte (MIT)

IX. 3. Appendices – List of PMI Workshop Participants

Ahn, J.W.	ORNL	**Kaita, R.**	PPPL	**Scime, E.**	WVU
Allain, J.P	UIUC	**Katoh, Y.**	ORNL	**Shimada, M.**	INL
Anders, A.	LBNL	**Koel, B.E.**	PU	**Singh, J.**	PSU
Andrucyzk, D.	UIUC	**Kolasinski, R.**	SNLL	**Skinner, C.H.**	PPPL
Blanchard, J.	UW-M	**Kotschenreuther, M.**	UT-A	**Snead, L.**	Consultant
Browning, P.	PSU	**Krasheninnikov, S.**	UCSD	**Soukhanovskii, V.A.**	LLNL
Callis, R.	GA	**Krstic, P.**	SUNY	**Stangeby, P.C.**	U-Toronto
Canik, J.M.	ORNL	**Kurtz, R.**	PNNL	**Stevens, E.**	DoE Observer
Caughman, J.B.	ORNL	**LaBombard, B.**	MIT	**Stotler, D.**	PPPL
Chang, C.S.	PPPL	**Leonard, A.W.**	GA	**Terry, J.L.**	MIT
Coburn, J.	NCSU	**Lisgo, S.**	ITER	**Tillack, M.S**	UCSD
Curreli, D.	UIUC	**Lumsdaine, A.**	ORNL	**Tynan, G.R.**	UCSD
deTemmermen, G.	ITER	**Lunsford, R.**	PPPL	**Unterberg, E.A.**	ORNL
Diallo, A.	PPPL	**Maingi, R.**	PPPL	**Valanju, P.**	UT-A
Doerner, R.	UCSD	**Majeski, R.**	PPPL	**VanDam, J.**	DoE Observer
Donovan, D.C.	UT-K	**Mansfield, D.**	PPPL	**Volpe, F.**	Columbia
Ellis, R.	PPPL	**Marmar, E.**	MIT	**Wang, Y.Q.**	LANL
Foster, M.	DoE Observer	**Menard, J.E.**	PPPL	**Whyte, D.G.**	MIT
Garrison, L.	ORNL	**Morley, N.**	UCLA	**Wilson, J.R.**	PPPL
Glenzer, S.H.	Stanford	**Myra, J.R.**	Lodestar	**Wirth, B.D.**	UT-K
Goldston, R.J.	PPPL	**Nardella, G.**	DoE Observer	**Wright, G.**	MIT
Goulding, R.H.	ORNL	**Neilson, G.H.**	PPPL	**Wukitch, S.**	MIT
Guo, H.Y.	GA	**Nygren, R.E.**	SNLA	**Xu, X.Q.**	LLNL
Hill, D.N.	LLNL	**Parish, C.M.**	ORNL	**Yoda, M.**	GIT
Hillis, D.L.	ORNL	**Pellin, M.J.**	ANL	**Youchison, D.L.**	SNLA
Hu, X.	ORNL	**Rapp, J.**	ORNL	**Zakharov, L.E.**	PPPL
Hutchinson, I.H.	MIT	**Reinke, M.**	Consultant	**Zinkle, S.J.**	UT-K
Jaworski, M.	PPPL	**Schenkel, T.**	LBNL		

IX. 4. Appendices – PMI Workshop agenda (p.1)

Epitome - Monday 5/4/2015

Entrance at PPPL guard booth **8:00**
Registration and badging in PPPL lobby **8:15**
Refreshments - Melvin B. Gottlieb (MBG) Auditorium **8:30**

Introduction Session – Auditorium (Chair: Maingi) *
Prager, Foster - Welcome **9:00**
R. Maingi/S. Zinkle – Goals, process, timeline, logistics **9:10**
H. Guo/B. LaBombard – SOL and divertor physics – ReNeW Thrust 9 **9:20**
J.P. Allain/R. Doerner – Plasma-materials interactions – ReNeW Thrust 10 **9:30**
C. Kessel/D. Youchison – Engineering Innovations – ReNeW Thrust 11 **9:40**
A. Hubbard/T. Leonard – Core/edge integration issues – ReNeW Thrust 12 **9:50**

Coffee break **10:00**

Plenary Session – Auditorium (Chair: Maingi) *
G. de Temmermen – PWI Research Needs for ITER **10:30**
I. Nunes – Experience with ILW in JET **11:00**

Lunch break **11:30**

Parallel Sessions: (Led by thrust leaders and deputies) **1:00**
Thrust 9: talks and structured discussion (B331 – Director's Conference Room)
Thrust 10: talks and structured discussion (B318)
Thrust 11: talks and structured discussion (B252) *
Thrust 12: talks and structured discussion (A104 – Visualization Wall)

Coffee break **3:00**

Parallel Sessions: (Led by thrust leaders and deputies) **3:30**
Thrust 9: talks and structured discussion (B331 – Director's Conference Room)
Thrust 10: talks and structured discussion (B318)
Thrust 11: talks and structured discussion (B252) *
Thrust 12: talks and structured discussion (A104 – Visualization Wall)

Joint Panel Session – Auditorium (Chair: Zinkle) * **5:30**
D. Whyte – Achieving/exploring reactor-level PMI simulation in small-scale devices
R. Nygren - A New Vision for Materials, In-vessel Components and Diagnostics for the Plasma Edge
J. Rapp - Integrated PMI R&D with a multi-device approach

Adjourn **6:30**
Working dinner: Crosscutting group and sub-panel leads discussion (B318) **7:00**

IX. 4. Appendices – PMI Workshop agenda (p.2)

Epitome - Tuesday 5/5/2015

<u>Joint Parallel Sessions:</u> (Chaired by crosscutting group) **8:30**
Thrust 9&12: talks and structured discussion (A104 – Visualization Wall) **
Thrusts 10&11: talks and structured discussion (Auditorium) *

Coffee break

<u>Joint Parallel Sessions:</u> (Chaired by crosscutting group) **11:00**
Thrusts 9&10: talks and structured discussion (Auditorium) **
Thrusts 11&12: talks and structured discussion (A104 – Visualization Wall) *

Working Lunch served in Auditorium **12:00**

<u>Parallel Sessions:</u> (Led by thrust leaders and deputies) **1:30**
Thrust 9: structured discussion (126 – Engineering Conference Room) **
Thrust 10: structured discussion (Auditorium)
Thrust 11: structured discussion (B252) *
Thrust 12: structured discussion (A104– Visualization Wall)

Coffee break

<u>Parallel Sessions:</u> (Led by thrust leaders and deputies) **3:00**
Thrust 9: structured discussion (126 – Engineering Conference Room) **
Thrust 10: structured discussion (Auditorium)
Thrust 11: structured discussion (B252) *
Thrust 12: structured discussion (A104 – Visualization Wall)

<u>Plenary Session</u> – MBG (Chair: Maingi) * **6:00**
Thrust 9, 10, 11, 12 PRD updates

Adjourn

Group No-Host Dinner, Salt Creek Grill, Forrestal Village **7:00**

IX. 4. Appendices – PMI Workshop agenda (p.3)

Epitome - Wednesday 5/6/2015

<u>Parallel Sessions:</u> (Led by thrust leaders and deputies) **8:30**
Thrust 9: structured discussion (126 – Engineering Conference Room) **
Thrust 10: structured discussion (Auditorium)
Thrust 11: structured discussion (B252) *
Thrust 12: structured discussion (T169 – Theory Conference Room)

Coffee break

<u>Parallel Sessions:</u> (Led by thrust leaders and deputies)
Option: this session may be modified for additional cross-thrust discussions **11:00**
Thrust 9: structured discussion (126 – Engineering Conference Room) **
Thrust 10: structured discussion (Auditorium)
Thrust 11: structured discussion (B252) *
Thrust 12: structured discussion (T169 – Theory Conference Room)

Working lunch served in Auditorium **12:00**

<u>Plenary Session: Status of each sub-panel:</u> (Chair: Zinkle, Auditorium) *
H. Guo/B. LaBombard – SOL and divertor physics – ReNeW Thrust 9 **1:30**
J.P. Allain/R. Doerner – Plasma-materials interactions – ReNeW Thrust 10 **2:00**
C. Kessel/D. Youchison – Engineering Innovations – ReNeW Thrust 11 **2:30**
A. Hubbard/T. Leonard – Core/edge integration issues – ReNeW Thrust 12 **3:00**

Break (no coffee)

<u>Option for Parallel or Joint Sessions:</u> (Led by thrust leaders and deputies) **4:00**
Thrust 9: post-plenary discussion (B252) **
Thrust 10: post-plenary discussion (T169)
Thrust 11: TBD (B205 if needed) *
Thrust 12: TBD

IX. 4. Appendices – PMI Workshop agenda (p.4)

Epitome - Thursday 5/7/2015 (*sub-panel members*)

<u>Plenary Session: Crosscutting discussion</u> (Chair: Hill/Neilson, B318) * **8:30**

Coffee break

<u>Parallel Sessions:</u> (Led by thrust leaders and deputies) **11:00**

Thrust 9: (B205) **
Thrust 10: (B318)
Thrust 11: (B252) *
Thrust 12: (B331 – Director's Conference Room)

Adjourn

IX. 4. PMI Workshop Overview and Room Locations

Day/time	Thrust 9 Div/SOL	Thrust 10 PMI	Thrust 11 Engineering	Thrust 12 integration
Mon 9:00	Introduction session MBG Auditorium			
Mon 10:30	Plenary session MBG			
Mon 11:30	Lunch			
Mon 1:00	B331	B318	B252	A104
Mon 5:30	Joint Panel Session MBG			
Mon 7:00	Executive Committee working dinner B318			
Tues 8:30	Joint #9,#12 A104	Joint #10, #11 MBG	Joint #10, #11 MBG	Joint #9,#12 A104
Tues 11:00	Joint #9, #10 MBG	Joint #9, #10 MBG	Joint #11, #12 A104	Joint #11, #12 A104
Tues 12:00	Lunch			
Tues 1:30	EngConfRm	MBG	B252	A104
Tues 6:00	Plenary MBG			
Wed 8:30	EngConfRm	MBG	B252	T169
Wed 12:00	Working lunch in MBG			
Wed 1:30	Subpanel reports MBG			
Thurs 8:30	SP members-only B318			
Thurs 11:00	SP members only B205	SP members only B318	SP members only B252	SP members only B331
Thurs 12:30	Adjourn			

IX. 4. PMI Workshop Thrust 9 Parallel Sessions - Agenda

Monday (5/4/15)

Guidance: Nominal 20 minute time slots – 12 minutes for talk + 8 for questions

Session I (B331 – Director's Conference Room)

1:00	Guo, LaBombard	Organization of subpanel 9 sessions
1:20	C.S. Chang	Importance of SOL plasma kinetic information for more reliable PMI data
1:40	J.M. Canik	Model validation needs in boundary physics
2:00	J.R. Myra	Understanding the SOL: Fundamental Physics Challenges
2:20	A. Anders	Unipolar arcs on the first wall: gaining deeper understanding of arc ignition conditions and development of arc-prevention strategies
2:40	V.A. Soukhanovskii	Snowflake divertor
3:00	*Coffee Break*	

Session II (B331 – Director's Conference Room)

3:30	S.I. Krasheninnikov	Detachment 101
4:10	Structured Discussion	
5:30	Plenary Session	

Tuesday (5/5/15) - Subpanel 9 Parallel Sessions

Session III (126 – Engineering Conference Room)

1:30 Structured Discussion
2:30 *Coffee Break*

Session IV (126 – Engineering Conference Room)

3:00 Structured Discussion

Plenary Session (Auditorium)

6:00 **Thrust 9, 10, 11, 12 PRD updates**
6:45 Adjourn
7:00 *Group No-Host Dinner, Salt Creek Grill, Forrestal Village*

Wednesday (5/6/15) - Subpanel 9 Parallel Sessions
Session V (126 – Engineering Conference Room)

8:30 Structured Discussion
10:30 *Coffee Break*

Session VI (126) -- if not replaced by crosscutting discussion

11:00 Structured Discussion
12:00 *Working Lunch* (Auditorium)

Plenary Session (Auditorium)

1:30 **Thrust 9, 10, 11, 12 PRD updates**
3:30 Adjourn

IX. 4. PMI Workshop Thrust 10 Parallel Sessions –Agenda for Talks

Guidance: Nominal 15 minute time slots – 10 minutes for talk + 5 for questions

Session I - Facilities Monday 5/4 1:00 PM

S.H. Glenzer	Stanford	Opportunities for fusion material science studies at LCLS
M.J. Pellin	ANL	Extreme Materials (XMAT) Beam Line for In Situ Examination of Radiation Damage
R. Majeski	PPPL	Test stands for liquid metal PFC development
R.H. Goulding	ORNL	A multiply-heated RF plasma source for a novel linear divertor simulator
Y. Katoh	ORNL	Impact of Neutron Irradiation on Plasma-Materials Interactions

Session II - Diagnostics Monday 5/4 3:30 PM

T.M. Biewer	ORNL	PMI Diagnostic Development Needs
Z.S.Hartwig (Wright)	MIT	The necessity to advance diagnostics for plasma facing component surfaces
C.M. Parish	ORNL	Qualifying materials' response to plasma-materials interaction
E. Scime	WVU	Two Photon Absorption Laser Induced Fluorescence Measurements of Neutral Hydrogen in the Tokamak Edge

Session III - Modeling Tuesday 5/5 1:30 PM

C.H. Skinner	PPPL	Coordinated experimental-modeling approach to low-risk PFCs for FNSF/DEMO
B.D. Wirth	UT-K	Status of Modeling Plasma - Materials Interactions: Unresolved Issues & Future Opportunities
D. Curreli	UIUC	Challenges and strategies to experimental validation of multi-scale nuclear fusion PMI computational modeling
P. Krstic	SUNY	Integrated, Multi-Scale Plasma-Material Interface Simulation

IX. 4. PMI Workshop Thrust 11 Parallel Session –Agenda for Talks

Guidance: Nominal 20 minute time slots – 12 minutes for talk + 8 for questions

<u>Session I</u> Monday 5/4 1:00 PM

A. Lumsdaine	Engineering Enhanced Heat Transfer Materials
L. Garrison	Development of advanced tungsten and alternative materials through advanced manufacturing
D.G. Whyte	Plasma facing engineering solutions enabled by modularity & demountable coils
F. Volpe	Feedback Stabilization of Flowing, Electromagnetically Restrained Liquid Metal Walls
D. Andruczyk	Liquid Metal's Role to Improve Power Handling through Engineering Innovation

<u>Session II</u> Monday 5/4 3:30 PM – structured discussion

<u>Session III</u> Tuesday 5/5 1:30 PM

J. Singh	Fabrication of Net-shaped Functional Graded Nano-dispersion Strengthened Tungsten Alloys for Structural Applications in Fusion Energy

All subsequent parallel sessions – structured discussion

IX. 4. PMI Workshop Thrust 12 Parallel Sessions – Agenda for Talks

Guidance: Nominal 20 minute time slots – 12 minutes for talk + 8 for questions

Parallel Session 1 Monday 5/4 1:00 PM**:**

1:00	Hubbard + Leonard	Scope of panel, intended output (if needed after am session)
1:20	Jon Menard	Potential challenges, research needs, and solutions for core-edge integration
1:40	Jim Terry	Challenges for integrating power-handling constraints and those of a high-performance core
2:00	Dick Majeski	Low recycling walls and confinement
2:20	C. S. Chang	Importance of kinetic physics in core-edge integration
3:00	Coffee Break	

3:30 Parallel Session II: Structured Discussion

Possible topics:

Priority Research Topics. Do we have the right set? What is missing?

Metrics: Can/should we quantify some of these issues, to serve as template for assessing initiatives?

Initiatives: Considering whole set of white papers (including those not in talks), do we have all bases covered? If not (and perhaps ahead of meeting), assign people to summarize options in other realms on Tuesday.

All subsequent parallel sessions – structured discussion

IX. 4. PMI Workshop Joint Parallel Sessions on Tuesday 5/5 –Agenda for Talks

<u>Thrusts 9 & 12: Joint Parallel Session</u> Tuesday 5/5 8:30 AM
Guidance: Nominal 20 minute time slots – 12 minutes for talk + 8 for questions

B. LaBombard	ADX: a high field, high power density, advanced divertor and RF tokamak
X.Q. Xu	Develop a Validated Predictive Modeling Capability for Localized Transient Events under Detached Divertor Operations
R. Nygren	Smart Tiles and MEMS-based sensors - new age of wall/edge diagnostic
M. Kotschenreuther	Cumulative sensitivity of high Q operation on ITER and burning plasmas to issues of integrated operation

<u>Thrusts 10 & 11: Joint Parallel Session</u> Tuesday 5/5 8:30 AM
Guidance: Nominal 15 minute time slots – 10 minutes for talk + 5 for questions

R. Nygren	Advanced Manufacturing and Engineered Materials – A New Vision for Materials and PFC Development
M. Kotschenreuther	Implications of Recent SOL Projections, and Tungsten Sputtering, on Tolerable ELM size: SOL physics, and plate design
Y. Wang	Controlled He Release Through Nanocomposite Materials Design
G. Wright	Operation of a Tokamak with a Hot Wall
M. Shimada	Tritium and Nuclear Sciences Initiative for Burning Plasma Long Pulse PMI
R. Goldston	An Example Opportunity for Divertor Innovations: The Lithium Vapor-Box Divertor
B. Koel	Liquid Metals as Plasma facing Materials for Fusion Energy Systems
J. Caughman	Reliable Long Pulse Plasma Heating and Current Drive using ICRF

IX. 4. PMI Workshop Joint Parallel Sessions on Tuesday 5/5 – Agenda for Talks

<u>Thrusts 9 & 10: Joint Parallel Session</u> Tuesday 5/5 11 AM - Noon
Guidance: Nominal 15 minute time slots – 10 minutes for talk + 5 for questions

G. Tynan	Addressing PMI Challenges with Complementary Linear Device and Confinement Device Studies
D. Buchenauer (R. Kolasinski)	Neutral H sensor for C-X H flux on wall and divertor
I. Hutchinson	Divertor Detachment Basic Physics
T. Schenkel(Anders)	Multi-scale and time-resolved studies of point defect dynamics in materials, to further the understanding of PMI for fusion

<u>Thrusts 11 & 12: Joint Parallel Session</u> Tuesday 5/5 11 AM - Noon
Guidance: Nominal 15 minute time slots – 10 minutes for talk + 5 for questions

S. Wukitch	PMI Challenges and Path towards RF Sustainment of Steady State Fusion Reactor Plasmas
R. Nygren	Understanding Design Integration to Confirm the Credibility of Liquid Surface PFCs
R. Majeski	Lithium walls for fusion
L. Zakharov	Flowing Liquid Lithium (24/7FLiLi): the technology step to burning plasma regimes

IX. 5. Appendices – List of Submitted PMI White papers

Allain, J.P	UIUC	Challenges and strategies to experimental validation of multi-scale nuclear fusion PMI computational modeling
Anders, A.	LBNL	Unipolar arcs on the first wall: gaining deeper understanding of arc ignition conditions and development of arc-prevention strategies
Andrucyzk, D.	UIUC	Liquid Metal's Role to Improve Power Handling through Engineering Innovation
Bertelli, N.	PPPL	Integrating RF power into scrape-off-layer plasma simulation
Biewer, T.M.	ORNL	PMI Diagnostic Development Needs
Briesemeister, A.R.	ORNL	Compatibility of RMP ELM Control and detached divertor conditions
Buchenauer, D.	SNLCA	Neutral H sensor for C-X H flux on wall and divertor
Callis, R.	GA	Center for Applied Fusion Material Research
Canik, J.M.	ORNL	Taking the next step in boundary model validation
Canik, J.M.	ORNL	The importance of the parallel plasma transport close to the material surface for PMI
Caughman, J.B.	ORNL	Reliable Long-Pulse Plasma Heating and Current Drive using ICRF
Chang, C.S.	PPPL	Kinetic Simulation of Scrape-off and Edge-Core Plasmas Using PIC Method for High Fidelity PMI Research
Coburn, J.	NCSU	New Focuses for Future PMI Studies: Testing Innovative Materials, Focusing on Material Temperature Control, and Implementing Cross-disciplinary Research
Curreli, D.	UIUC	Large-Scale Integrated Modeling of Plasma Boundary and Plasma-Material Interactions
D'Ippolito, D.A.	Lodestar	ICRF-Edge and Surface Interactions
Delzanno, G.L	LANL	Dust, a critical player in the complex plasma-material interaction problem for long pulse tokamaks
Donovan, D.C.	UT-K	Surface Heat Flux Characterization on Linear and Toroidal Confinement Devices
Ellis, R.	PPPL	Near-term test facilities for liquid metal plasma facing components
Garrison, L.	ORNL	Development of advanced tungsten and alternative materials through advanced manufacturing

Glenzer, S.H.	Stanford	Opportunities for fusion material science studies at LCLS
Goldston, R.J.	PPPL	An Example Opportunity for DIvertor Innovation: The Lithium Vapor-Box Divertor
Goulding, R.H.	ORNL	Use of Multiple RF Heating Sources in a Linear Divertor Simulator
Guo, H.Y.	GA	Developing and Validating Heat Flux Solutions for Next-Step Fusion Devices
Hartwig, Z.S.	MIT	The necessity to advance diagnostics for plasma facing component surfaces
Hutchinson, I.H.	MIT	Divertor Detachment Basic Analysis
Joseph, I.	LLNL	Theory and Simulation of Resonant Magnetic Perturbations
Katoh, Y.	ORNL	Impact of Neutron Irradiation on Plasma-Materials Interactions
Koel, B.E.	PU	Liquid Metals As Plasma facing Materials For Fusion Energy Systems
Kolemen, E.	PU	Advanced Magnetic Divertor Control
Kotschenreuther, M.	UT-A	Implications of Small SOL widths on Tolerable ELM Size and ELM Tungsten Sputtering
Kotschenreuther, M.	UT-A	Cumulative Integrated Performance on ITER that allows Q=10
Krstic, P.	SUNY	Integrated, Multi-Scale Plasma-Material Interface Simulation
LaBombard, B.	MIT	ADX: a high field, high power density, advanced divertor and RF tokamak
Leonard, A.W.	GA	A Pedestal Transport Initiative to Resolve the Compatibility of Core Plasma Scenarios with Boundary Plasma Solutions in Burning Plasma Tokamaks
Lore, J.D.	ORNL	Addressing the need for fluid plasma boundary modeling
Lumsdaine, A.	ORNL	Engineering Enhanced Heat Transfer Materials
Majeski, R.	PPPL	An approach to a tokamak reactor with fast flowing liquid lithium PFCs
Majeski, R.	PPPL	Test Stands for Liquid Metal PFC Development
Majeski, R.	PPPL	Low recycling walls and confinement
Majeski, R.	MIT	Achieving and exploring reactor-level PMI simulation in small-scale devices
Menard, J.E.	PPPL	Potential challenges, research needs, and solutions for core-edge integration

Mirhoseini, S.M.H.	Columbia	Feedback Stabilization of Flowing, Electromagnetically Restrained Liquid Metal Walls
Moeller, C.P.	GA	A Traveling Wave Helicon Launcher for > 1 GHz
Myra, J.R.	Lodestar	Understanding the SOL: Fundamental Physics Challenges
Nygren, R.E.	SNLA	A New Vision for Materials, In-vessel Components and Diagnostics for the Plasma Edge
Nygren, R.E.	SNLA	Advance Manufacturing and Engineered Materials - A new vision for materials and PFC development
Nygren, R.E.	SNLA	Smart Tiles and MEMS-based sensors - new age of wall/edge diagnostic
Nygren, R.E.	SNLA	Understanding Design Integration to Confirm the Credibility of Liquid Surface PFCs
Parish, C.M.	ORNL	Qualifying materials' response to plasma-materials interaction
Pellin, M.J.	ANL	Extreme Materials (XMAT) Beam Line for In Situ Examination of Radiation Damage at the Advanced Photon Source
Rapp, J.	ORNL	Integrated PMI R&D with a multi-device approach
Reusch, L.M.	UW-M	Integrated Data Analysis of Measurements in the Edge of Fusion Devices
Schenkel, T.	LBNL	Multi-scale and time-resolved studies of point defect dynamics in materials, to further the understanding of PMI for fusion
Shimada, M.	INL	Tritium and Nuclear Sciences Initiative for Burning Plasma Long Pulse PMI
Simonen, T.C.	UC-B	Three Game Changing Advances: A Simpler Fusion Concept
Singh, J.	PSU	Fabrication of Net-shaped Functional Graded Nano-dispersion Strengthened Tungsten Alloys for Structural Applications in Fusion Energy
Sizyuk, T.	Purdue	The Effect of Mixed and Impurity Materials on the Performance and Reliability of Plasma facing Components
Skinner, C.H.	PPPL	Coordinated experimental-modeling approach to low-risk PFCs for FNSF/DEMO
Soukhanovskii, V.A.	LLNL	Taming the plasma-material interface with the snowflake divertor
Stangeby, P.C.	Univ.	Flow-through solid PFCs using carbon or

	Toronto	other low-Z refractory coatings
Sternlieb, A.	Ariel U.	The Liquid Lithium Wall/Divertor Pathway to Fusion Energy
Tang, X.	LANL	Feedback of PMI on SOL Plasmas
Terry, J.L.	MIT	Challenges for integrating power-handling constraints and those of a high-performance core
Thomas, D.M.	GA	Advancing Plasma-Material Interface Solutions for Next-Step Fusion Devices
Tillack, M.S	UCSD	The Materials-Design Interface for Fusion Power Core Components
Unterberg, E.A.	ORNL	The Challenge in Compatibility of Main-chamber Materials with Next-Step Fusion Devices
Wang, Y.Q.	LANL	Controlled He release through nanocomposite materials design
Whyte, D.G.	MIT	Assessment of reactor PMI science in small-scale devices
Wirth, B.D.	UT-K	Status of Modeling Plasma - Materials Interactions: Unresolved Issues & Future Opportunities
Wright, G.	MIT	Operating a tokamak with a high-temperature wall
Wright, G.	MIT	The Need for Fusion-Relevant Irradiation in Understanding the Plasma-Material Interactions and Ultimate Design of Next Generation PFCs
Wukitch, S.	MIT	PMI Challenges and Path towards RF Sustainment of Steady State Fusion Reactor Plasmas
Xu, X.Q.	LLNL	Develop a Validated Predictive Modeling Capability for Localized Transient Events under Detached Divertor Operations
Yoda, M.	GIT	Development of Helium-Cooled Tungsten Divertor Systems
Youchison, D.L.	SNLA	Thin LM/solid hybrid FWs for DEMO Blankets
Zakharov, L.E.	PPPL	Flowing Liquid Lithium (24/7FLiLi): the technology step to burning plasma regimes

IX. 6. Appendices – List of Common Acronyms

ASDEX-U	ASDEX-Upgrade fusion facility, Germany
BES	Basic Energy Sciences, U.S. Department of Energy
CC	Crosscutting
C-MOD	Alcator C-Mod fusion facility, Boston, MA
CX	Charge Exchange
DBTT	Ductile to Brittle Transition Temperature
DFT	Density Functional Theory
DIII-D	DIII-D fusion facility, San Diego, CA
DPA	Displacements Per Atom
DD	deuterium-deuterium
DT	deuterium-tritium
DTT	Divertor Test Tokamak
DEMO	Demonstration fusion power plant
EAST	Experimental Advanced Superconducting Tokamak fusion facility, China
EC	Electron Cyclotron
ECH	Electron Cyclotron Heating
ECRF	Electron Cyclotron Radio Frequency
ELMs	Edge localized modes
E.U.	European Union
eV	Electron Volt
EFDA	European Fusion Development Agreement
FES	Office of Fusion Energy Sciences, U.S. Department of Energy
FESAC	Fusion Energy Sciences Advisory Committee
FNSF	Fusion Nuclear Science Facility
IC	Ion Cyclotron
ICRF	Ion Cyclotron Radio Frequency
ILW	ITER-like Wall (installed in the JET device)
ITER	ITER fusion facility, France
ITPA	International Tokamak Physics Activity
JET	Joint European Torus fusion facility, United Kingdom
JT-60SA	Superconducting fusion facility under construction, Japan
KSTAR	Korean Superconducting Tokamak Advanced Research fusion facility, Korea
LCFS	Last Closed magnetic Flux Surface
LH	Lower Hybrid
LHCD	Lower Hybrid Current Drive
LHRF	Lower Hybrid Radio Frequency
LIBS	Laser-Induced Breakdown Spectroscopy
LM	Liquid Metal
MD	Molecular Dynamics
MeV	Million Electron Volt
MFE	Magnetic Fusion Energy
MST	Madison Symmetric Torus fusion facility, Madison, WI

NSTX	National Spherical Torus Experiment fusion facility, Princeton, NJ
NSTX-U	National Spherical Torus Experiment Upgrade fusion facility, Princeton, NJ
PFC	Plasma Facing Components
PFM	Plasma Facing Materials
PFR	Private Flux Region
PFS	Plasma Facing Surfaces
PMI	Plasma Materials Interactions and/or Plasma Materials Interface
PRD	Priority Research Direction
PSI	Plasma Surface Interactions
RAFM	Reduced Activation Ferritic Martensitic steel
ReNeW	MFE Research Needs Workshop report (2009)
RF	Radio Frequency
RMP	Resonant Magnetic Perturbation
SOL	Scrape-off Layer
TBM	Test Blanket Module
TBR	Tritium Breeding Ratio
TCV	Tokamak fuson facility, Switzerland
UQ	Uncertainty Quantification
W7-X	Wendelstein 7-X stellarator fusion facility, Germany
WEST	Tungsten Environment Steady-state Tokamak fusion facility, France