Developing Heat Flux and Advanced Material Solutions for Next-Step Fusion Devices

and DIII-D BPMIC Team

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1. Executive Summary

The path towards next-step fusion development requires increased emphasis on the boundary/plasma-material interface. One of the major issues facing the design and operation of next-step high-power steady-state fusion devices is the control of heat fluxes to the plasma-facing components (PFCs), including divertor and main chamber walls. Since the maximum steady-state power load is limited to 10 MW/m² for solid materials (including both graphite and tungsten), it is essential to find plasma solutions that control heat fluxes to keep them within the heat exhaust limitations of the PFCs. This poses an even greater challenge for the Fusion Nuclear Science Facility (FNSF) with the envisioned Advanced Tokamak (AT) or Spherical Tokamak (ST) scenarios because of relatively low density ($n/n_G \approx 0.5$ for both cases) with respect to ITER ($n/n_G \approx 1$) and stringent needs to simultaneously control divertor and core parameters.

In addition, viable PFCs must be developed for the next-step device and are a universal challenge to fusion energy, regardless of confinement concept. Candidate materials should eliminate or reduce core contamination and/or dust production due to erosion, have reasonable lifetimes under high heat and particle and neutron bombardment, and minimize tritium retention. A national initiative to develop these Advanced Materials (AM) and components is needed, especially now as we enter the ITER era of fusion energy development and prepare for the design of FNSF.

It is therefore urgent to find viable boundary/plasma-material interactions (PMI) solutions with adequate core plasma performance to enhance the appeal of fusion power plants. To respond to this challenge, we have established a new Boundary/PMI Center at the DIII-D National Fusion Research Facility with the initiative to address critical PMI issues for fusion in the ITER era, i.e., the next ten-years and beyond. Towards this end, the Center will focus on the following transformational approaches on a schedule that is in-step with, and relevant to, the design of FNSF:

- Finding divertor plasma solutions compatible with the core plasma high performance operational scenarios in FNSF – This entails validation of predictive models for heat flux dissipation, development and test of advanced divertor configurations on DIII-D to reduce the density threshold for detachment.

- Validating AMs for reactor PFCs at reactor-relevant temperatures in DIII-D high-performance plasmas, in collaboration with the broad material research/development community.

- Integrating boundary-materials interface with high-performance AT plasmas to provide viable boundary/PMI solutions for next-step fusion devices.
This boundary/PMI initiative leverages unique DIII-D capabilities and broad collaborative efforts within the DIII-D and DiMES (Divertor Material Evaluation System) programs. It will promote synergistic programs within the broad PMI community, including linear material research facilities, and enable the US to become a forefront international leader in this critical area. This initiative integrates well into the OFES vision for fusion research in the ITER era. Specifically, this initiative fits directly into three of the four newly developed program elements of OFES, i.e., Burning Plasma Science: Foundations, High power, and Discovery Plasma Science. It will also enable us to build a compelling bridge for the US research on Long Pulse.

2. Development of Innovative Heat Flux Solutions

Heat flux mitigation will be a critical constraint of FNSF design and solutions are urgently needed to meet this challenge. As a metric for power exhaust, the ratio of the power flow across the separatrix to the major radius is \( P/R \) \([\text{MW}/m]\) ~ 20 for ITER, but exceeding 30 for FNSF. In addition, the plasma duration expected for FNSF is much longer than ITER, by 2 to 3 orders of magnitude. Furthermore, an efficient, non-inductive current drive and high performance plasma operation favors lower density with the Greenwald density fraction \( n/n_G \sim 0.5 \), as listed in Table 1. Therefore, FNSF will face the following increased PMI challenges:

- **Power exhaust**: The maximum steady-state power handling capability is 10 MW/m² for both graphite and tungsten.
- **PFC erosion**: Requires very low divertor temperature to suppress sputtering, i.e., below 5 eV to avoid physical sputtering.
- **Compatibility with adequate core performance expected for FNSF and DEMO**: With higher \( \beta_N \) and lower \( n/n_G \), compared to ITER.

This demands radiative dissipation of heat flux through divertor detachment without compromising core performance.

Developing advanced divertor solutions necessitates an innovative approach on both the physics and engineering fronts. This can be seen in the following simple formula, which indicates some requisite elements for divertor-based heat-flux mitigation:

\[
q_{\text{target}} = \frac{(1-f_{\text{rad}})P_{\text{SOL}} \sin(\theta_{\text{div}})}{4\pi R_{\text{target}} \lambda_q f_{\text{exp}}},
\]

where \( f_{\text{rad}} \) is the radiative power fraction in the SOL and divertor; \( P_{\text{SOL}} = P_{\text{CD}} + 0.2 \times P_{\alpha} - P_{\text{rad-core}} \) is the power flow into the SOL including current drive and heating power \( P_{\text{CD}} \) and \( \alpha \) particle heating power, \( P_{\alpha} \), and allowing for radiative loss from the confined plasma, \( P_{\text{rad-core}} \), \( \theta_{\text{div}} \) is the angle between the poloidal flux surface and target plate; \( \lambda_q \) is the radial decay length of the heat flux, which is largely set by the
plasma current with $\lambda_q \sim 1/I_p$, and $f_{\text{exp}}$ is the poloidal flux expansion factor. Clearly, addressing heat flux issues requires:

- Optimizing divertor geometry, e.g., $R_{\text{target}}$ and $\theta_{\text{div}}$, to enhance divertor screening for neutrals, improving particle exhaust, reducing impurity contamination and promoting divertor detachment by localized hydrogenic and impurity radiation.
- Optimizing magnetic configuration to maximize poloidal and toroidal flux expansion ($f_{\text{exp}}$, $R_{\text{target}}$) and increase field-line length, thus effectively increasing the divertor volume and heat flux footprint.
- Active radiation and particle control by injecting highly radiative impurities to enhance radiation inside the divertor ($f_{\text{rad}}$), as well as in the core plasma to reduce the power flow into the divertor ($P_{\text{loss}}$), without compromising the core plasma performance.

This initiative leverages unique DIII-D capabilities to address boundary/PMI issues. DIII-D provides a world-class fusion environment with FNSF-relevant plasma scenarios, i.e., $\beta_N$ up to ~5 and $P/R \sim 22$ for DIII-D upgrade, and with broad domestic/international collaborations. In addition, DIII-D accommodates a wide range of magnetic configurations with easily modified divertor hardware and a robust plasma control system. Furthermore, DIII-D has world-leading divertor diagnostics (Fig. 1). This, coupled with the excellent existing platform for boundary/PMI studies based on the DiMES materials validation system and close collaboration with SciDAC PMI modeling effort, provides a unique opportunity to advance code validation.

We will take an integrated approach towards developing innovative boundary/PMI solutions for FNSF. This embraces the three
closely coupled aspects shown in Fig. 2. Model validation is key to advance scientific understanding and to develop predictive capability for boundary/PMI in current and future devices. It requires exercising codes to compare with experiment for validation. In particular, it is essential to establish the capability to accurately predict divertor detachment scenarios and provide a physics basis for robust heat flux and erosion control for next-step devices. The extensive divertor diagnostic set on DIII-D provides a compelling basis for model validation. For example, the Divertor Thomson Scattering (DTS) gives detailed two-dimensional (2D) distributions of divertor plasma temperature and density by sweeping the outer strike point across the array. This technique provides essential data for the understanding of detachment physics (Fig. 3). With this initiative, we will further augment diagnostic capabilities to enhance the model validation efforts, e.g., enhancing the DTS system to be truly 2D without need for strike-point sweeping.

The second key element of this integrated approach is to develop and test advanced divertor solutions in high-performance plasmas relevant for FNSF so the benefit of the geometry in decoupling core/divertor performance can be quantified. DIII-D has flexible divertor hardware and magnetic configurations, as well as a robust control system, allowing us to effectively and quickly explore a wide range of divertor configurations. As an example, Fig. 4 shows a series of divertor configurations with increasing divertor closure for neutrals and impurities and potentially more advanced divertor configurations. The snowflake divertor configuration has been successfully demonstrated in DIII-D, leading to reduction in peak heat flux and broadening of heat deposition on the divertor target plates (V.A. Soukhanovskii et al., “Radiative Snowflake Divertor Configuration Studies in DIII-D”, presented at 21st International Conference on Plasma Surface Interactions, Kanazawa Ishikawa, Japan, May 26 – 30, 2014).

The third element of this approach is to integrate boundary/core to provide robust, validated integrated core/boundary solutions for future fusion devices, taking advantage of advanced tokamak scenarios developed in DIII-D. We will first assess validated boundary solutions in plasma scenarios expected for FNSF, i.e., high power with $P/R \sim 22$, high performance with $\beta_N \sim 2 – 4$, and relatively low density with $n/n_G \sim 0.5$, as well as core magnetic shaping relevant to FNSF.

This integrated core/boundary approach takes advantage of unique DIII-D capabilities and broad domestic and international collaborations to address the challenge facing power and particle handling.
under high performance and advanced tokamak conditions. This is not only of great value to ITER, but will also explore physics and technology for FNSF and next-step devices. *This approach is cost effective and ready to address the critical issues now!*

**Figure 4.** Flexibility in hardware and a robust control system allow DIII-D to explore a wide range of divertor configurations from ITER-like divertor structure to more advanced snowflake and isolated divertor configurations.

### 3. Advanced Materials (AM) Validation

The 10-year goal of this PMI/PFC research initiative for DIII-D is to implement reactor-relevant PFC materials at the temperatures necessary to begin component testing for FNSF. PMI model validation and PFC development is the grand challenge for successful fusion energy. It is a universal challenge since it is critical for any fusion reactor device, and, therefore, solutions developed by this initiative will have a broad impact within the global fusion energy effort. To address this critical issue for next-step fusion development, focused research combining toroidal confinement experiments and linear materials testing facilities are required for integrated systems testing of materials and components, including exposure to off-normal plasma events and a broad spectrum of plasma energy and particle-fluxes. Short-pulse, toroidal devices (~ 10sec), in particular, add further benefit by providing an experimental environment that is relatively easy to access and comprehensively diagnose, as opposed to long-pulse toroidal devices. This is especially important for initial component development (e.g., at technology readiness levels TRL 1-5) where quick turnaround and detailed measurements are desirable for a number of materials and/or components prior to a long-pulse systems test. The DIII-D National Fusion Facility is preparing to address this challenge as part of a national PMI research initiative by providing a flexible, well-diagnosed environment for materials evaluation and integrated testing. This should assist a goal-driven effort to develop novel, reactor-relevant PFC solutions for use in next-step devices.
Within magnetic fusion energy (MFE) research, a clear development path is available to bring novel materials from concept to near end-use readiness. Presently, it is envisioned that tungsten, W, will be the first wall and potentially the PFC material required for the neutron environment of next-step fusion devices. There are, however, concerns for W as a PFC material, in particular about issues of melting/cracking due to high heat flux damage and/or the production of surface nano-tubes (termed ‘fuzz’) that can cause surface arcing and dust production. These concerns probably apply to any high-Z PFC material, as fusion has had a long history of issues with core plasma performance degradation when high-Z PFCs have been used. It is with these issues in mind that this initiative proposes an AM Program using a toroidal device as part of the development process.

**Figure 5.** Integration of multiple aspects of a focused, coordinated effort in PFC development is required in order to develop options for FNSF.

A number of AM concepts can be considered for taming the plasma-materials interface, for example, thin film coatings of low-Z materials on a high-Z bulk or silicon carbide substrates, cutting-edge advanced manufacturing techniques for thin-layer and/or nanostructure alloys, etc. We believe that a well-focused, comprehensive research program targeting such AM solutions will not only focus current experiments on more reactor-relevant PFC/PMI regimes, but will also lead to unique world-wide leadership for the US fusion effort. These topics and details have been discussed extensively in the 2009 FESAC ReNew report and the 2012 FESAC “Materials Science and Technology Research Opportunities Now and in the ITER Era” Report.

As a part of this initiative, DIII-D is planning to test reactor-relevant AM and solid PFC options within the coming 10-year horizon. We see this as a national focus on material/PFC development beyond W for next-step devices, which is urgently needed for MFE in general, but is especially needed now in preparation for an FNSF design in 10-years time. As outlined in Fig. 5, this path involves the coordinated effort of multiple disciplines and devices – each with unique added benefit to a focused effort on materials development. In particular, materials science effort is needed not only to develop materials to
test, but also for modeling the effects of both incident plasma and neutron fluxes on these materials. This fundamental understanding is crucial to the ultimate development of predictive models for materials.

A second part of this development strategy is linear device testing of targeted materials. These devices include such machines as the UCSD PISCES, the proposed ORNL MPEX, and the Magnum-PSI device in DIFFER-FOM in the Netherlands. These devices have easy access to materials samples for detailed measurement and modeling, and they provide a simulated long-pulse environment during the initial technical readiness development phase, especially at the conceptual development stage. Ultimately, toroidal devices, both long- and short-pulsed systems, play the critical role in this strategy. Long-pulse devices will provide the most comprehensive test before an FNSF or DEMO of any material and/or PFC. Historically, long-pulse facilities are restricted in operational space and port access, also the US is not planning any long-pulse devices in the foreseeable future (before FNSF), and so all tests and research will be done on international devices. Therefore, appropriate research arrangements will need to be established. This initiative should include working with both Asian (e.g., EAST, KSTAR, JT60-SA) and European (e.g., WEST and W7-X) devices. Short-pulse toroidal devices are the most critical element, however, since they provide a mitigated risk path as materials move through the TRL process, not only regarding cost, but also regarding accessibility and turn-around time. This is especially vital for rapid development of AMs. Short-pulsed devices allow easier and thereby more comprehensive access for diagnostics. Coupling this measurement capability with linear devices and model validation constitutes a virtually ideal component test environment.

This initiative leverages the unique DiMES platform for materials evaluation and PMI studies on DIII-D. Coupled with a large diagnostic suite already available, DiMES provides a quantitative and nearly comprehensive capability for research on critical PFC materials issues such as erosion, redeposition, and migration of low-Z and high-Z materials. As an example, DIII-D recently performed a molybdenum (Mo) erosion/migration study using the DiMES system where small samples of Mo were exposed to the divertor plasma for ~30 sec of plasma exposure time, Fig. 6. The net/gross erosion and material migration patterns were measured using both in- and ex-situ measurement methods. Fig. 6 shows the 1 cm diameter Mo sample set into the DiMES graphite head and the surface concentration of deposited Mo on the graphite surface surrounding the Mo sample from ex-situ Rutherford Backscattering (RBS) measurements. These measurements were then
compared to modeling using the OEDGE code package coupled with the HEIGHTS code package, and good agreement was found. DIII-D has a history of being a groundbreaker and test-bed for new and innovative diagnostics such as DiMES, the DTS, Divertor Thomson Scattering system, etc. As some of the critical challenges of PMI/PFC research are probably not yet known or lack adequate measurement, a central part of this initiative is a focus on coupling new diagnostic capability to the DIII-D PMI program, e.g., SMART tiles. In addition to the diagnostic capability, which makes such quantitative analysis possible, DIII-D affords a unique test environment in that high-Z materials are truly trace elements, so their migration is easily measured, in contrast to devices with high-Z PFCs. The AIMS in situ surface analysis diagnostic [Whyte et al, USBPO Newsletter, March 31, 2014] for measuring PFC material migration would be particularly effective in the DIII-D environment.

In addition to extensive diagnostic development, this initiative will include heating the walls to temperatures relevant to next-step devices ($T_{wall} \geq 500 \degree C$). A major issue in next-step devices is the retention of tritium. Retention as well as permeation into the bulk will occur at radically different rates at $>500 \degree C$, compared with present tokamak operating temperatures. Hot wall operation is likely to entirely alter D (T) recycling at the solid surfaces compared with present-day devices, potentially having a strong effect on the behavior of the confined plasma, including fusion performance.

The critically important and major next step will be to couple a high-performance tokamak core plasma and hot-wall operation. In a sequenced plan, DIII-D will first heat the DiMES system to 700$\degree C$. Next, we are planning to install a heated exposure and test facility that could change tiles in DIII-D using a remote arm. Finally, extensive areas of PFCs (e.g., the entire divertor and main chamber) will be designed and heated to high temperatures.

The near-term research focus will be the quantitative study of migration of high-Z materials as called out by recent advisory panels. Addressing the basic physics issues of high-Z material erosion and migration is facilitated by DIII-D’s carbon PFCs since high-Z materials are truly trace elements. Through close collaboration, studies of erosion/redeposition and migration in DIII-D will assess and extend findings of detailed measurements of plasma-material interactions made in linear device facilities. Medium-term research will include integrating the acquired experience of diagnostic techniques and PMI understanding for application to international long-pulsed devices. Materials and component solutions developed by the fusion community will be implemented and evaluated in DIII-D, taking advantage of its unique capability to not only test the proposed PFC solutions for relevant tokamak conditions, but also evaluate their compatibility with high performance AT operational regimes.

Such a comprehensive, nationally focused effort on PFC development is well matched to the 10-year vision of the DOE Office of Fusion Energy Sciences (OFES). In particular, this initiative targets a critical frontier in applied science generally – the development of materials that can withstand uniquely harsh environments. The materials challenge is arguably the largest one facing fusion energy development. This initiative will provide a platform for developing predictive models and new diagnostic methods for “taming the PMI” in fusion. By creating a national program in this area and engaging national laboratories and universities, this initiative will foster the development of new scientists within this field. This national focus will result in the formation of centers of expertise in the critical and potentially transformative field, combining material science, plasma physics, and advanced engineering science.
4. Summary and Schedule

This initiative will leverage existing DIII-D capabilities and broad domestic and international collaborations, establish a basis for robust heat flux and PMI control in realistic fusion plasma environment for FNSF, and promote critical research aimed at identifying boundary/PMI solutions for next-step devices generally. The timeline of the DIII-D Boundary/PMI Initiative is shown in Fig. 7. In order to provide timely information for the design of FNSF, we are planning to develop and assess high performance core scenarios with the existing carbon divertor through 2020. In the meantime, we will build a solid physics basis for the development of new divertor configurations in DIII-D and test AMs provided by the materials research and development community in collaboration with linear devices, such as PISCES, proposed MPEX and Magnum-PSI, and international long-pulse facilities, such as EAST, KSTAR and JT-60SA. We will start to test the new divertor with hot walls beginning in 2021, followed by integrating core/divertor scenarios with reactor-relevant AM PFCs.

Figure 7. Timeline of DIII–D boundary/PMI initiative to address this critical issue for FNSF.