

High Heat Flux Testing Facilities and an Electrothermal-Arc Plasma Source for Plasma-Material Interaction Studies and Diagnostic Development

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ABSTRACT

Tokamak plasma-facing components experience significant stresses from plasma-material interactions (PMI) due to cyclic high thermal loads, plasma exposure, and neutron irradiation. As the field progresses to reactor-level power fluxes, the harsh fusion environment demands much of plasma-facing materials.

Chapter 1 introduces plasma-material interactions and the condition plasma-facing components are expected to endure. While plasma exposure and neutron radiation damage are introduced, the synergetic effects of cyclic high thermal loads under plasma exposure are focused on. This motivates the need for plasma-facing materials studies.

Chapter 2 gives a brief overview of many high heat flux test facilities worldwide. Three prominent high heat flux devices are discussed, including (1) the linear plasma device Magnum-PSI, (2) the electron beam facility JUDITH-II, and (3) the electrothermal arc plasma source SIRENS. This motivates the high heat flux testing of materials in a highly controlled, repeatable, transient-level plasma exposure and plasma heat flux conditions. This is the motivation for the project undertaken at Oak Ridge National Laboratory.

Chapter 3 is the project undertaken and is the main focus of this report. At Oak Ridge National Laboratory, an electrothermal arc plasma source (ET-Arc) was modified and rebuilt in the laser diagnostics lab. The motivation for this project is twofold: (1) to develop two laser diagnostics, a “Portable Diagnostic Package” (PDP) which characterizes the plasma spectroscopically and Digital Holography (DH) which measures the *in situ* erosion of the target; and (2) to use the ET-Arc source and DH to conduct erosion studies under transient-level ($\sim \text{GW m}^{-2}$) heat fluxes. The modifications, upgrades, and diagnostic capabilities are discussed as well as future work.

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CHAPTER ONE

INTRODUCTION AND GENERAL INFORMATION

1.1 FUSION ENERGY

With global increases in population and standard of living, the global power consumption is rising drastically. With energy consumption on the rise, sustainable power generation becomes increasingly important (Fig 1.1). Sustainable baseline power is important due to the intermittency of other renewable energy sources. Magnetically-confined fusion could provide sustainable worldwide power.

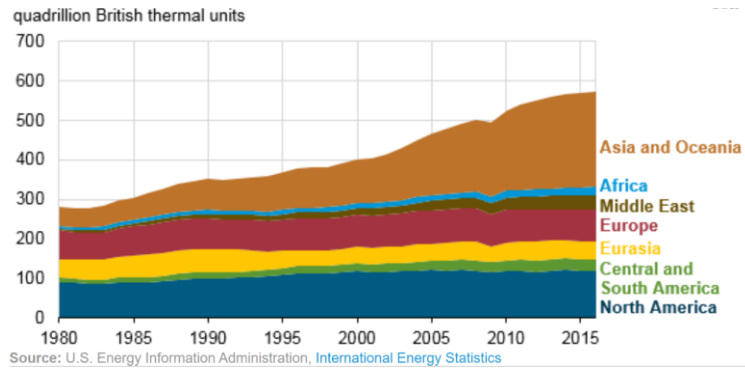


Figure 1.1: Energy use by world region from 1980-2016 is increasing rapidly, particularly in Asia. (EIA)

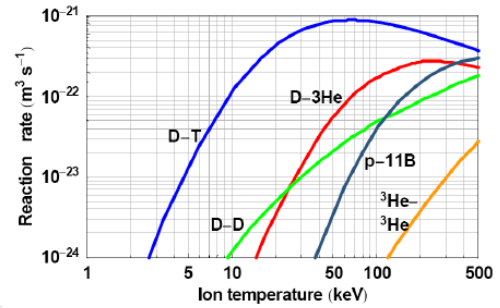


Figure 2.1: Fusion reaction rates (Gabrielli 2014)

Based on the nuclear reaction cross-sections, hydrogenic fusion is the most feasible fusion reaction (Fig 1.2, Table 1.1). A deuteron (D) fuses with a triton (T) to form an alpha particle and a 14.1 MeV free neutron (Eqn 1.1).

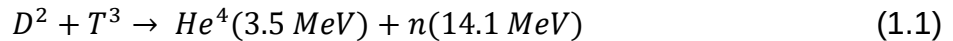


Table 1.1: Fusion reactions (Dolan 2013)

Reaction	Energy Released (MeV)	Threshold Energy (keV)
$D + T \rightarrow He^4 + n$ (14.1 MeV)	17.6	4
$D + D \rightarrow \begin{cases} T + p \\ He^3 + n \end{cases}$ (2.45 MeV)	4.0 3.25	35
$D + He^3 \rightarrow He^4 + p$	18.2	30

For the reaction to be viable, the plasma parameters need to meet the Lawson triple product (Eqn 1.2, Fig 1.3).

$$nT\tau_E > \frac{12kE_f}{\langle\sigma v\rangle_f/T^2} \cong 3 \times 10^{21} \frac{\text{keV}}{\text{s m}^3} \quad (1.2)$$

where n is the particle density, T is the average temperature of particles, and τ_E is the time duration that particles remain in the plasma. Optimization of the triple product increases the probability of fusion reactions occurring. For D-T fusion, $n \sim 10^{24} \text{ m}^{-3}$, $T > 5 \text{ keV}$, and $\tau_E > 0.6 \text{ ms}$. Current tokamaks, such as JET and JT-60U, have been able to reach break-even energy gains, although with D-D plasmas instead of D-T. ITER is designed to reach $Q \geq 10$.

To reach these conditions, plasmas are created and controlled by high-field magnetic devices called advanced tokamaks (Fig 1.4). Tokamaks were invented in the USSR in 1958 and have continued to develop since then. While several other magnetic confinement concepts exist, the tokamak has been the most extensively investigated worldwide due to its efficiency. (Stacey 2005)

The plasma is created and then heated to fusion-relevant conditions. The central solenoid is used to create the plasma by generating a time-varying magnetic field which drives a current through the plasma. Driving a current through a resistive medium, such as plasma, will generate heat ohmically. Then a variety of auxiliary heating mechanisms are used to reach fusion-relevant temperatures.

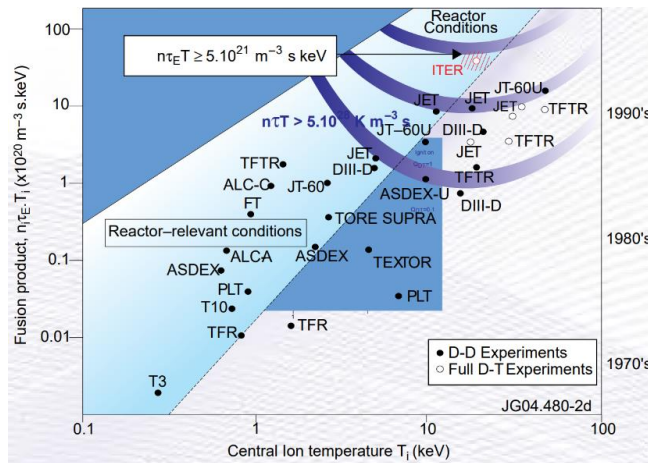


Figure 1.3: Fusion history showing progress in triple product, with dark blue bands (lowest to highest) meaning break even $Q=1$, ITER conditions $Q=10$, and reactor relevant conditions $Q=50$. (Mlynar 2007)

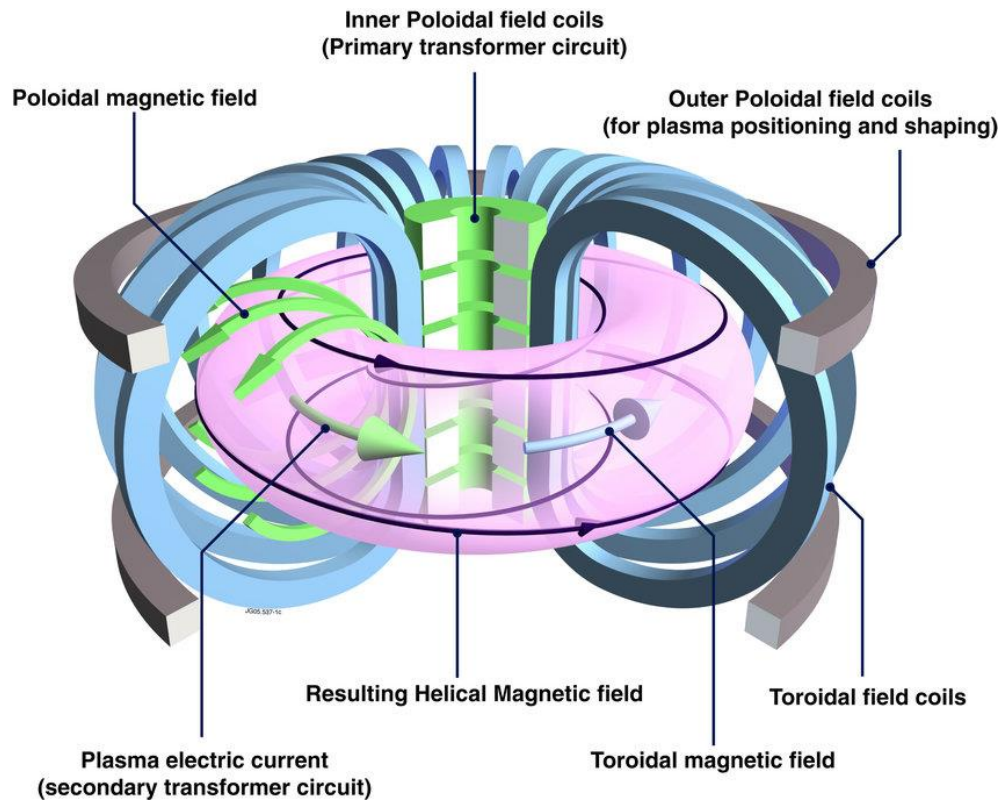


Figure 1.4: A diagram of the tokamak magnetic field coils and confined plasma (pink). The toroidal field coils (blue) confine the plasma inward in a donut; the poloidal field coils (grey) center the plasma, and the central solenoid (green) generates the plasma. The helical arrows (dark blue) show the magnetic field lines the plasma particles follow. (Haupt 2018)

The plasma is confined to a torus shape with the poloidal and toroidal field coils. The poloidal field coils exert vertical magnetic pressure inwards, while the toroidal field coils exert horizontal magnetic pressure. This keeps plasma primarily in the center of the tokamak. However, at an increased plasma energy density, confinement becomes difficult. Over time, the particles leave the plasma through various drift mechanisms or as exhaust. If the particles drift perpendicularly to the magnetic field, they will eventually impact the first wall. Otherwise, they will follow the magnetic field lines along the connection length and then be exhausted by the divertor (Fig 1.5).

The plasma-facing section of the tokamak is typically split into two sections: the first wall and the divertor regions. These two regions have vastly different material qualifications due to their different roles. The first wall constitutes the majority of the surface area in the tokamak. These materials need to have (1) benign plasma interaction, (2) chemical compatibility with the plasma, (3) and irradiation damage resistance. The first wall receives 90% of the exhaust power and particles over a large area. For proposed demonstration fusion reactors, a typical first wall heat flux is 2.5 MW m^{-2} . These thermal loads are tolerable, but particle bombardment can cause sputtering, the process by which plasma ions impinging on the first wall dislodge near-surface lattice atoms where they become plasma impurities. Sputtering will be discussed further in Section 1.2.1. Impurities radiate energy away from the core and radiative losses scale with Z superlinearly $\propto Z^2$, $\propto Z^4$, or $\propto Z^6$ depending on the radiation mechanism. (Stangeby 2000) Hence, high- Z ions are incredibly detrimental to the core plasma temperature. However, the sputtering mechanism is energy- and mass-dependent, so low- Z materials are much more susceptible to erosion than high- Z . Mid- Z is actually worse than either low- or high- Z because mid- Z has low binding energies and has many electron shells, so it sputters easily and has high radiative losses. While most tokamaks have had low- Z first walls, such as graphite or beryllium, high- Z coatings, like tungsten, are also common in tokamaks.

The second part of the plasma facing components is the divertor. The divertor is an upgrade to the initial tokamak concept. A magnetic coil is added underneath the tokamak, creating “figure eight” magnetic field lines. In the cross section, the “figure eight” is not seen, merely

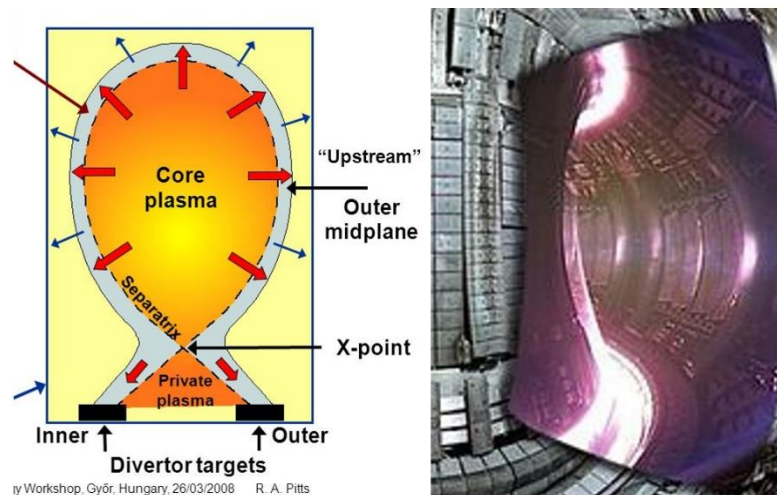


Figure 1.5: (a) Poloidal cross section of a tokamak, showing the core plasma (orange) and the scrape-off layer (grey) which touches the two divertor targets (Pitts 2008); (b) a camera image of the JET tokamak showing high heat flux on the divertor (EUROfusion)

hidden outside the device, but forms an “X-point” where the magnetic field lines cross. Tokamaks can have single- or double-null points, with a divertor below and/or above the device. The X-point separates the plasma into the core plasma on closed magnetic field lines and the scrape-off-layer (SOL) plasma on open field lines (Fig 1.5). The open field lines contact the divertor floor which exhausts impurities and heat, preventing impurity build up and plasma collapse. The divertor improves confinement time, plasma purity, and decreases heat flux at the first wall. (Dolan 2013) However, with the open field lines directly connected to the material surface, the divertor is subject to incredibly high heat and particle fluxes. It receives the remaining 10% of the heat flux incident on a narrow loop around the torus, resulting in 10 MW m^{-2} heat flux and particle fluxes of $\sim 10^{24} \text{ m}^{-3}$. These heat fluxes are at the material and heat handling limits.

The fusion reactor environment is incredibly challenging for materials. Plasma subjects materials to high heat flux, high particle flux, thermomechanical stress, neutron irradiation, and helium implantation. Developing plasma-facing materials resistant to plasma-induced damage is critical to reactor success, both economically and energetically.

1.2 PLASMA-MATERIAL INTERACTIONS

There are many single-variable plasma-material interaction (PMI) mechanisms that synergize to make a truly unique materials environment. The main ways plasma damages materials are: high heat flux (HHF), high particle flux (HPF), neutron irradiation, and particle implantation. Plasma-facing components (PFCs) need high thermal conductivity, sputtering resistance, low radiative losses, irradiation resistance, and favorable high-temperature properties while remaining economical and machinable on a large scale. These factors determine PFC material selection.

Of the plasma-material interactions, Section 1.2.1 will focus on high heat flux (HHF) consequences including heat flux reduction and transient events. Section 1.2.2 will focus on high particle flux and how it affects the plasma and the material surface. Section 1.3 will briefly discuss candidate fusion materials. Finally, Section 1.4 will summarize chapter one and motivate the project.

1.2.1 High Heat Flux (HHF)

For viable DT fusion reaction rates, plasma temperatures are typically ~5-10 keV, >58 million Kelvin (Fig 1.2). However, as particles travel along the closed core field lines, they experience several drift forces such as $\mathbf{E} \times \mathbf{B}$, ∇B , curvature, gravitational, and polarization drifts. These forces pull the particles perpendicular rather than parallel to the magnetic field lines. The boundary of the last closed flux surface begins the scrape-off layer edge plasma (SOL) (Fig 1.5). All magnetic field lines after the last closed flux surface, e.g. SOL plasma, have open field lines that terminate on the divertor. The SOL plasma-wall boundary is created by a limiting surface on the first wall, which can also be an engineered limiter. A limiter constantly scrapes off plasma particles that drift too far off the magnetic field lines and impact the limiter. The edge plasma is much cooler than the core and is typically collisional. SOL widths are typically a few millimeters thick.

Once in the SOL plasma, the particles diffuse either perpendicular or parallel to the magnetic field lines, resulting in perpendicular $q_{\perp}$ and parallel $q_{\parallel}$ heat fluxes.

Perpendicular diffusion results in heat flux on the first wall

$$q_{\perp} = \frac{f_{\perp} P_{SOL}}{A_{FW}} \quad (1.3)$$

where $f_{\perp}$ is total power fraction that diffuses across the field lines, P_{SOL} is the power in the scrape off layer, and A_{FW} is the area of the first wall. A sample calculation is

$$q_{\perp} = \frac{(0.9)(80 \text{ MW } m^{-2})}{600 \text{ m}^{-2}} = 0.1 \text{ MW } m^{-2} \quad (1.4)$$

Otherwise $q_{\parallel} = q_{div}$ where the particles follow the open field lines and impact the divertor. The heat flux $q_{\parallel} = q_{div}$ is

$$q_{\parallel} = \frac{f_{\parallel} P_{SOL}}{2\pi R_t \lambda_q f_{mp}} \quad (1.5)$$

where P_{SOL} the power in the scrape off layer, f_{div} the fraction of the total power on one divertor target, R_t the major radius of the target, λ_q (or λ_{SOL}) the width of the power flux of the SOL at the midplane, and f_{amp} the ratio from the target-to-midplane heat flux width. (Loarte 2007, Dolan 2013, Fedirici 2003) Current tokamaks, including ITER

predictions, produce steady state heat fluxes of $\sim 10 \text{ MW m}^{-2}$. This is at the limit of current heat removal technology. However, reactor-relevant power generation will be much higher, resulting in higher heat fluxes. Predictions for DEMO include $\sim 10^3 \text{ MW m}^{-2}$ unmitigated heat flux (Table 1.2). Hence, heat flux handling is a serious engineering and materials challenge especially for commercial feasibility.

Table 1.2. Comparison of heat flux capacity of current tokamaks to next steps, ITER and DEMO. (Petrie 2009, Soukhanovskii 2017, Asakura 2013, Zheng 2013)

	Current Tokamaks	ITER	DEMO
P_{heat}/R (MW m ⁻¹)	≤ 27	≤ 20	80 – 100
q_{peak} (MW m ⁻²)	$\leq 5 - 15$	≤ 50	100 – 300
$q_{\parallel}$ (MW m ⁻²)	100 – 2,000	5,000	30,000 unmitigated

1.2.1.1 Heat Flux Reduction

There are five main techniques to reduce heat flux on a surface: (1) magnetic flux expansion, (2) radiative power loss, (3) angle of incidence, (4) collisional losses with neutrals, and (5) strike point sweeping (Fig 1.6). Magnetic flux expansion is when the magnetic field close to the incident surface is weaker, allowing the magnetic field lines to separate which increases the area the power is spread over. Radiative power loss is detrimental to the core plasma, but extremely useful at the first wall. Hot particles can radiate away some of the energy via photons, decreasing their energy before impacting the wall. Minimizing the angle of incidence increases the wetted area by $q \sin(\theta)$ which drastically reduces the power density on the wall. Collisional losses are from hot particles transferring some of their energy to cold neutrals in front of the wall before impact.

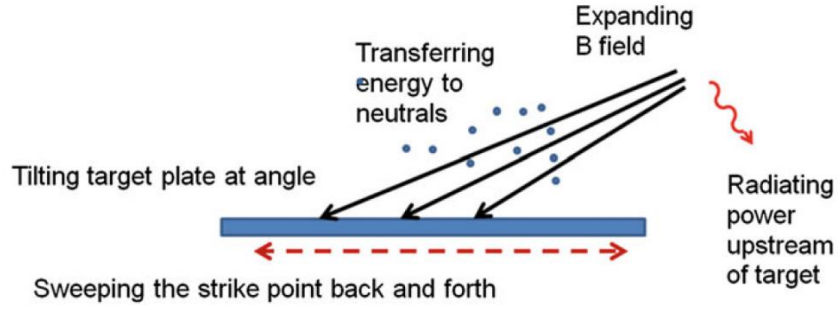


Figure 1.6: The five methods for heat flux reduction. (Dolan 2013)

The strike point is the area where the magnetic field lines touch the divertor. It is a λ_{SOL} -wide ring around the tokamak floor. The strike point can be swept laterally across the divertor to spread the heat load out. For an ITER-like approximation, consider 60% of the power flows in a 5 mm thick shell towards the divertor, with flux expansion 3-5, a SOL width $\lambda_{\text{SOL}} \sim 25$ mm, and an incident angle of $\sim 3^\circ$. Hence, a plasma “shell” containing ~ 60 MW is incident on ~ 50 mm wide area resulting in 20 MW m^{-2} power, not accounting for radiative power loss. (Dolan 2013)

The heat flux values discussed thus far is for steady state (SS) operation. Realistically, off-normal HHF events occur throughout SS pulses that subject PFCs to even higher heat fluxes. Heat flux is not constant since H-mode has frequent edge-localized modes (ELMs) and other transients that can deposit $0.1 - 2 \text{ GW m}^{-2}$ which is incredibly challenging for current materials. These heat fluxes are difficult to produce in current tokamaks, so other plasma or high heat flux sources are used for materials testing. Some types of disruptions are: edge-localized modes, vertical displacement events, and the Greenwald density limit.

1.2.1.2 Transient Events

Vertical displacement events (VDE) occur when the magnetic pressure and particle pressure maintaining the plasma’s vertical position become unbalanced. This can be prevented by proper poloidal field control. The plasma poloidal cross section is modeled as an ellipse of horizontal radius a and vertical radius b . To prevent a VDE, the plasma surface conducting shell needs to have an elongation of $\frac{b}{a} \leq 2$. (Stangeby 2000) If not

corrected, a VDE will allow the plasma migrate to the edge of the tokamak, potentially terminating directly on the wall. (Dolan 2013)

The Greenwald density n_G limit is a relationship between plasma density and power input. With increasing power, escaped plasma particles impact the first wall with higher energy, freeing more impurities. With more impurities in the core, the fuel fraction decreases and radiative power losses increase. Plasma cooling increases plasma resistivity, resulting in an unstable current profile. While the impurity fraction and power input mechanisms do contribute to the change in resistivity, the density limit does not depend on either parameter. When the H-mode density is raised, the plasma will disrupt at the Greenwald limit and return to L-mode. Hence, the mechanism is still not well understood. (Stacey 2005)

Edge-localized modes (ELMs) occur in H-mode at high poloidal mode numbers $m \sim 10$. H-mode is a high confinement mode of tokamak operation that is distinguished from L-mode by a step in the density profile called the pedestal (Fig 1.7). (Chen 2015) The density step creates a high

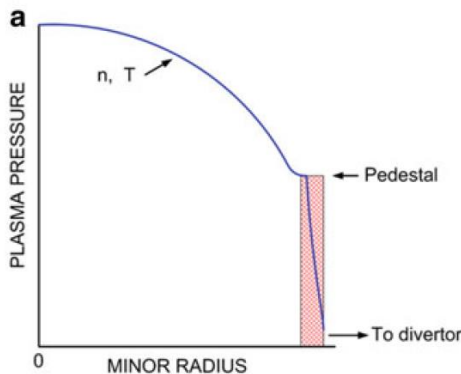


Figure 1.7: The characteristic density and temperature traces for tokamak H-mode (blue) operations. The pedestal (red) can break down during Type-I ELMs. (Chen 2015)

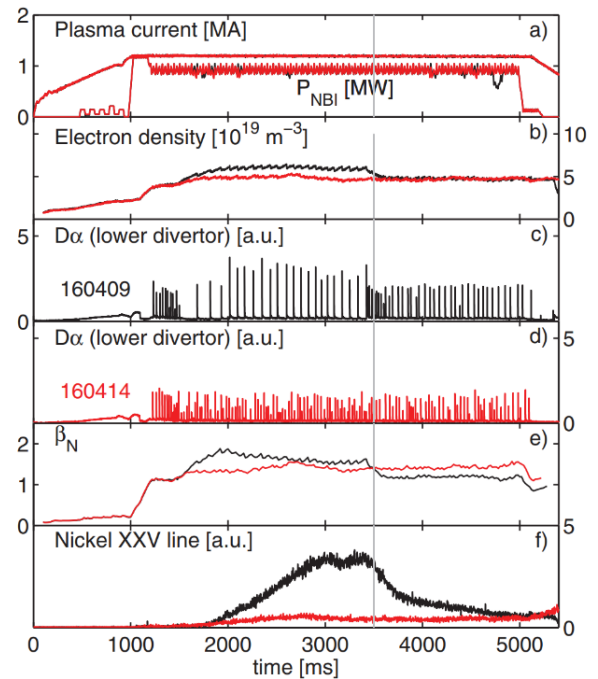


Figure 1.8: Evolution of D_α intensity as a proxy for ELM detection on the divertor. Spontaneous ELM activity (black) has higher intensity and lower frequency than LGI ELM pacing (red). Confinement decreased at $t = 3.5$ s due to $n = 2$ modes. Data from DIII-D. (Bortolon 2017)

shear, azimuthal flow which slows the outward diffusion of the plasma. The pedestal doubles the core density which increases core confinement and power production. However, this transport barrier to the SOL causes impurity build up in the core. Eventually, the impurities destabilize the core, causing a breakdown of the pedestal to exhaust the core impurities onto the divertor through the SOL. ELMs are not well understood but are related to high ∇p and/or ∇J at the plasma edge. ELMs are a peeling-ballooning magnetohydrodynamic instability at the plasma edge, thought to be caused by the sharp ∇p of the pedestal. Peeling is a type of kink mode, where kinks are a bend in the plasma causing a difference in B-field on the inside versus the outside of the bend. Ballooning is where the plasma expands in local areas of weaker magnetic field.

There are three categories of edge-localized modes. About 50-80% of the ELM energy goes to the divertor, with the remaining 50-30% going to the first wall. In divertor configurations with a single X-point, ELMs preferentially deposit more energy at the inner strike point than the outer. (Dolan 2013) Type-I ELMs are Giant Elms and are characterized by high-intensity D_α spectroscopy peaks. The pedestal density increases from $0.2n_G$ to $0.8n_G$ which switches the energy flow from a conductive to a convective regime. (Loarte 2007). During a Type-I ELM, the pedestal plasma just inside the separatrix becomes “connected” to the divertor target. There is a sudden burst of electron energy approximately equal to the pedestal T_e conducted to the target and formation of a high-energy sheath. This is followed later by convective ion fluxes at lower velocities. Giant ELMs can eject up to 10% of the confined plasma energy. The ejected power $\frac{\Delta E}{E} \leq 10\%$ is fairly constant, where the energy loss per ELM will increase as the plasma power increases. Fortunately, the ELM-I frequency is low, ~ 1 -100 Hz. However, since the time scale is very short, it is difficult to mitigate ELM-I heat flux. However, ITER-like conditions measured in JET indicate that 15-20% of the pedestal energy goes to the divertor during an ELM. For reactor-relevant conditions, ELM mitigation will be necessary to preserve the PFCs. Type-II ELMs, or Grassy ELMs, have the lowest heat flux $\frac{\Delta E}{E} \ll 1\%$ and are the rarest type of ELM. Finally, Type-III ELMs

are common, or intermediate, ELMs. They typically expel $\frac{\Delta E}{E} \leq 1\%$ of the plasma energy at high frequencies of 0.1-1 kHz (Fig 1.8). (Stacey 2005, Chen 2015, Dolan 2013, Bortolan 2017)

These are only a few types of off-normal transients that create extremely HHF on PFCs. Hence, heat handling is a critical issue in PFCs. However, the heat flux is delivered by energetic particles, so the interaction is not just thermal. Particle-particle interactions are also critical for PFC lifetime and plasma efficiency.

1.2.2 High Particle Flux (HPF)

Plasma-material interactions can affect both sides of the plasma-surface interface. Plasma-facing materials (PFM) affect the plasma through sputtering, deposition, and/or prompt redeposition. The plasma affects PFMs by 14.1 MeV neutron irradiation, particle implantation, and hydrogenic and helium retention.

1.2.2.1 Plasma Effects

At the plasma-material interface, plasma particles can interact with the surface in several ways: (1) deposition, (2) prompt redeposition, and/or (3) sputtering (Fig 1.9). Depending on which mechanism is dominant, these processes either create a net deposition or a net erosion layer. If deposition dominates, then the PFC tile thickens, but the deposited layer no longer has the thermomechanical properties of the original material. Regardless of whether deposition positively protects the PFCs, or deleteriously changes the PFC properties, measuring and monitoring change is imperative. Otherwise, if the dominant mechanism is net surface erosion, then the plasma slowly eats away at the PFC surface. Some reactor designs, including ITER, have actively-cooled divertors where cooling channels are a few millimeters below the surface. (Campbell 2019) In ITER, once erosion of 8 mm is reached, the fusion reactor would have a “loss of coolant” accident which would flood the vacuum chamber with water vapor. Hence, *in situ* diagnostic capabilities are necessary for the safe operation of fusion reactors.

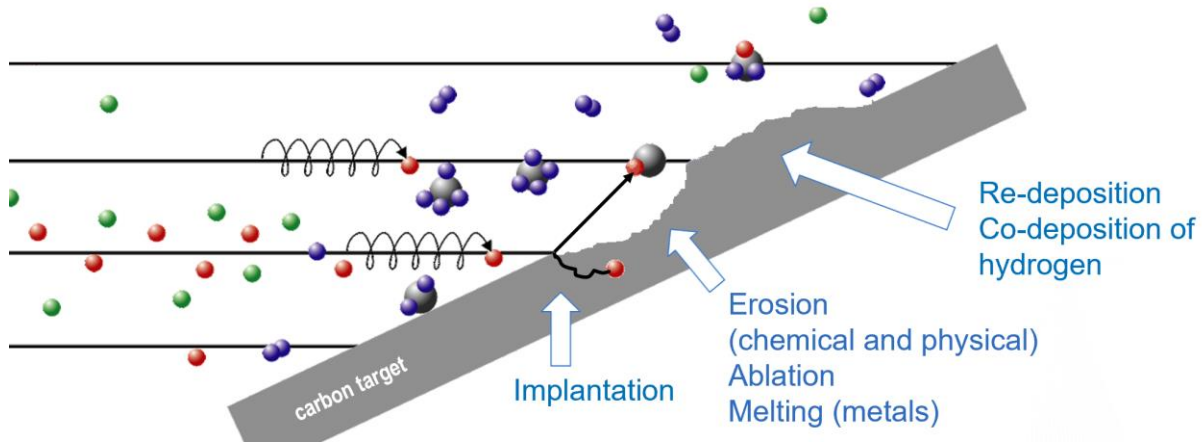


Figure 1.9: The three dominant plasma-material interaction mechanisms: implantation, erosion, and deposition. (Maingi 2015)

The particle flux incident on the wall is comprised of ions, electrons, and neutral atoms and neutral molecules. Electrically charged particles include plasma ions (D, T), impurity ions (He^3 , W, C, Be, N, O, etc), and electrons.

In plasma devices, objects, such as walls, develop electrostatic sheaths due to the increased mobility of the electrons compared to the ions. Electrons, due to their lower mass, will experience less collisions thereby reaching the wall quickly. This leads to the area in front of the wall carrying a slight negative potential. The plasma ions feel this potential drop and accelerate towards the wall, resulting in an increased energy of $\sim 3k_B T_e$. The ions will then Coulombically scatter or be implanted in the wall. (Stangeby 2000)

Charged particles will either recombine to become neutral or Coulombically accelerate through the sheath to the wall. Since neutral particles do not follow the magnetic field lines, they continue their trajectory from the point of recombination straight to the wall. Ions can also become neutral through charge exchange, where a cold neutral near the wall can give up its electrons to a plasma ion, resulting in a new cold ion and a new hot neutral. If the ion does not become neutral, it will accelerate through the sheath to impact the wall.

1.2.2.2 Sputtering

When an ion or energetic neutral impacts the wall, it ballistically transfers energy to the lattice atoms. In some cases, the energy transferred to a surface atom will exceed the threshold energy to liberate it from the matrix. This process is called physical sputtering. Chemical sputtering can also occur, where the energetic particle chemically reacts with the surface, breaking a chemical bond to free an atom or molecule from the surface.

Sputtering is temperature and mass dependent:

$$E_T = \frac{(m_1 + m_2)^2}{4m_1m_2} E_b \sin(\theta) = \frac{E_b}{\Lambda} \quad (1.5)$$

where E_T is the threshold energy, m_1 and m_2 are the masses of the incident and lattice atoms, θ is the angle of incidence, and E_b is the binding energy. (Was 2007) The mass ratio and is written succinctly as Λ . Typical values for the displacement energy is 25-50 eV. Typical ions will impact the wall at 1-15 eV. This simple relationship (Eqn 1.5) is one of the reasons high-Z materials are preferred for the divertor and low-Z materials for the first wall. In addition to other favorable properties, high-Z materials have high binding energies, requiring extremely hot particles to liberate them (Fig 1.10, Table 1.3). In tokamaks, plasma particles near the wall should be ~5 eV. As you can see (Fig 1.10), the low-Z materials such as Be and C are susceptible to sputtering at these temperatures. Conversely, high-Z materials such as Mo and W have very high energy thresholds. This is one reason high-Z materials are preferred for divertors.

For example, the binding energy of tungsten is $E_W = 90$ eV while graphite is $E_G = 30$ eV (Was 2007, ASTM E521). However, if high-Z particles

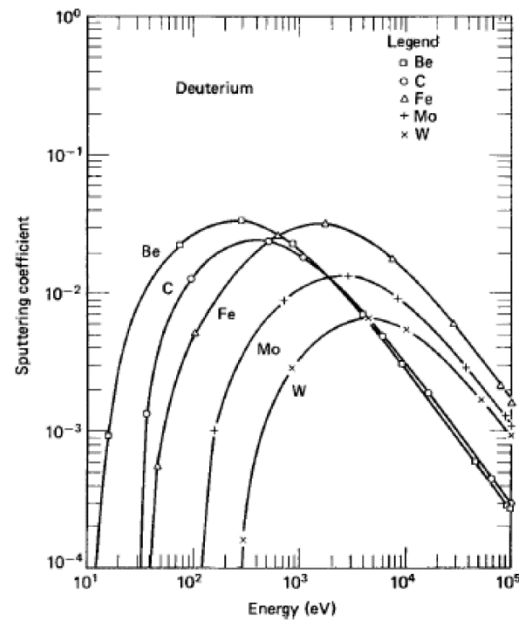


Figure 1.10: Physical sputtering yield (atoms per ion) for deuterium on various materials. (Stacey 2000)

are liberated which can happen by transient heat loads, self-sputtering can severely damage the material.

Table 1.3. Parameters for physical sputtering yield. E_{TF} (eV), Q (atoms per ion) (Was 2007)

	Beryllium			Graphite			Iron			Molybdenum			Tungsten		
	E_{th}	E_{TF}	Q	E_{th}	E_{TF}	Q	E_{th}	E_{TF}	Q	E_{th}	E_{TF}	Q	E_{th}	E_{TF}	Q
H	20	256	0.1	35	415	0.035	64	2 544	0.042	199	4 718	0.007	443	9 870	0.007
D	9	282	0.3	30	447	0.10	44	2 589	0.13	90	4 767	0.023	220	9 923	0.019
T	21	308	0.24	30	479	0.20	40	2 634	0.21	70	4 816	0.045	140	9 977	0.038
⁴ He	30	780	0.59	29	1 087	0.32	33	5 514	0.44	46	9 944	0.12	110	20 373	0.106
C	40	4 152	1.6	42	5 687	1.5	35	20 247	3.2	55	34 183	0.93	80	66 507	0.93
O	70	6 970	1.3		9 298			29 839			48 322		40	91 979	2.2
Self	25	2 208	1.4	42	5 687	1.5	40	174 096	13	64	533 048	18	65	1 998 599	20

Self-sputtering occurs when an ion is freed from the lattice. Because it is an ion, it will immediately feel the magnetic field as it leaves the surface. Charged particles follow field lines via gyromotion. The Larmor radius is mass-dependent (Eqn 1.6),

$$r_L \equiv \frac{v_{\perp}}{\omega_c} = \frac{mv_{\perp}}{|q|B} \quad (1.6)$$

where $v_{\perp}$ is a positive constant denoting the speed in the plane perpendicular to B , m is the mass of the ion, q is the elementary unit of charge, and B is the magnetic field. Therefore, the newly freed ion will follow the magnetic field lines in cyclotron motion. If the Larmor radius is less than the distance away from the surface, the ion will travel back to the surface. This is highly detrimental in the high-Z divertors because W-W collisions will release many more W atoms than HDT-W collisions. Not only does self-sputtering erode high-Z surfaces quicker, it is also extremely deleterious to the plasma.

1.2.2.3 Impurity Radiative Losses

Once impurities enter the plasma, they “steal” heat from the plasma core and dilute the fuel ratio. Worse, they radiate heat away from the core plasma. Radiative cooling scales superlinearly with Z (Eqns 1.7.1-1.7.3) via bremsstrahlung, line, and recombination radiation. (Stacey 2005)

$$P_{brems} \left[\frac{MW}{m^3} \right] \cong 4.8 \times 10^{-43} z^2 n_z [m^{-3}] n_e [m^{-3}] \sqrt{T_e [keV]} \quad (1.7.1)$$

$$P_{line} \left[\frac{MW}{m^3} \right] \cong 1.8 \times 10^{-44} z^4 n_z [m^{-3}] n_e [m^{-3}]^{-1/2} \sqrt{\sqrt{T_e [keV]}} \quad (1.7.2)$$

$$P_{re} \left[\frac{MW}{m^3} \right] \cong 4.1 \times 10^{-46} z^6 n_z [m^{-3}] n_e [m^{-3}]^{-3/2} \sqrt{\sqrt{T_e [keV]}} \quad (1.7.3)$$

The first wall is a lot of surface area, typically hundreds of square meters in advanced tokamaks. Transient heat loading will erode the first wall, so it is imperative to reduce the effect of first wall impurities on the core plasma. If you combine the radiative power loss equations, there is complex behavior due to different temperature and density dependence. However, at fusion-relevant temperatures it is apparent that low-Z elements radiate orders of magnitude less power than high-Z materials (Fig 1.11). If you calculate the radiative power as the triple product increases, there are limits where fusion ceases to generate net power. Hence, the upper limit of impurities in the core can be calculated before the core loses too much power and/or the fuel ratio is too low. Therefore, the plasma can tolerate a lot of low-Z impurities, but very few high-Z impurities (Fig 1.12). This is the main reason why low-Z materials are chosen for the first wall.

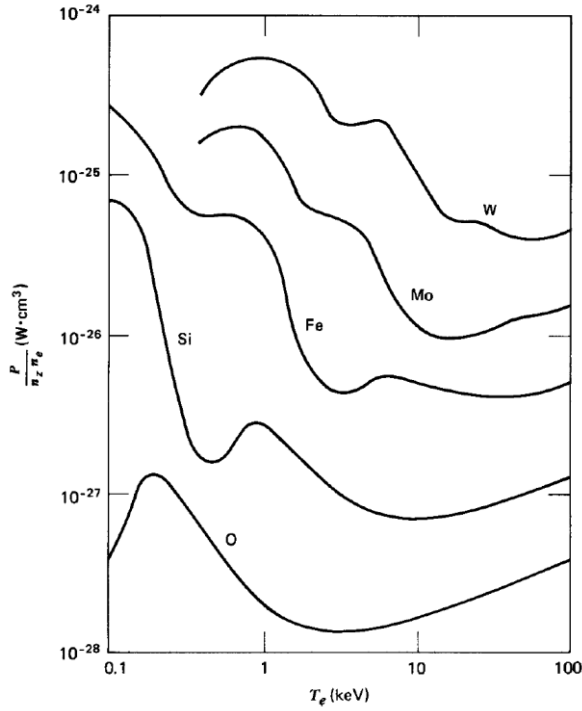


Figure 1.11: Radiative power loss equations 7.1-7.3 are combined to see the radiative power loss as a plasma increases in temperature. (Stacey 2005)

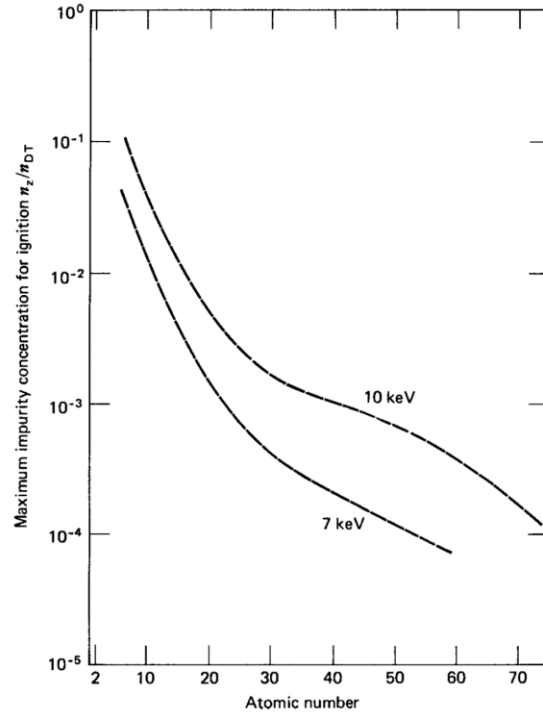


Figure 1.12: The maximum impurity concentration for which ignition can be achieved at two temperatures. (Stacey 2005)

Should sputtering processes be the dominant PMI mechanism, it results in net erosion of the PFC. However, should the incident particles not have sufficient energy to sputter the PFCs, sometimes they are removed from the vacuum vessel via pumping, or sometimes they deposit onto the surface. While deposition is beneficial to the lifetime of the PFC, it is not ideal for the plasma. The deposition layer will have different thermomechanical properties than the matrix. This makes modeling of PFC lifetime difficult and more susceptible to sputtering, tritium retention, and reduces heat handling.

1.2.2.4 Material Effects and Irradiation Damage

The plasma affects PFMs by 14.1 MeV neutron irradiation, particle implantation, and hydrogenic and helium retention. Neutron irradiation creates point defects that, at specific temperature and damage ranges, causes five main degradation mechanisms. Neutron irradiation causes radiation embrittlement and hardening, radiation-induced-segregation and -precipitates, void swelling, and helium embrittlement. Neutron damage is particularly deleterious in fusion materials due to synergistic effects. Plasma implants hydrogen isotopes and fusion ash into the materials. Additionally, neutron capture reactions can produce additional hydrogen and helium in the matrix material. Cavities are sinks for helium and subsequently hydrogen. This damages the PFMs through helium embrittlement, void swelling, and fuzz growth. Additionally, it increases the radioactivity of the first wall by trapping tritium.

The DT fusion reaction (Eqn 1.1) produces an alpha particle and a neutron. By conservation of energy, the neutron receives 75% of the energy. Hence, fusion reactors have a fast, mono-energetic neutron spectrum peaked at 14.1 MeV.

Neutron-material interactions are approximately elastic due to the particle's neutrality, so they will not interact Coulombically with the lattice or the lattice atoms. The neutron absorption and transmutation cross sections are also very low compared to elastic scattering cross sections. Hence, neutrons only “see” nuclei to interact with. When a neutron enters a material, it is a foreign particle in the ordered lattice structure. Typical interactions include: kinetic energy transfer to the lattice atom as a primary knock-on atom (PKA); displace an atom from its lattice site; or transmute an atom via a neutron capture reaction.

High-energy neutrons undergo nuclear reactions, such as neutron capture or neutron multiplication. Nuclear reactions include (n, γ) , (n, α) , (n, p) , and $(n, 2n)$ (Table 1.4).

Table 1.4. Given a neutron fluence of 1 MW y m^{-2} , the material will produce the given dpa, appm He, and appm H calculated from the fusion neutron cross sections. (was 2007)

Material	Displacements per atom (dpa)	Helium (appm)	Hydrogen (appm)
316 stainless steel	10	200	540
Nb	7	24	79
Mo	8	47	95
V	12	57	100
C	10	2700	Small
Be		2800	130

Neutrons transfer energy to lattice atoms through elastic collisions. (Was) The amount of transferable energy is

$$E_t = \frac{1}{2} \Lambda E_i (1 - \cos(\theta)) \quad (1.8)$$

where Λ is the average mass fraction in (Eqn 5), E_i is the energy of the bombarding particle, and θ is the scattering angle of incidence. Hence, the maximum energy transferred would be a direct collision between two atoms of the same mass, resulting in half the kinetic energy being transferred.

1.2.2.5 Neutron Damage

Neutron damage begins when a neutron transfers enough energy to displace a PKA, which then knocks into a secondary atom, which knocks into a third and fourth atoms, etc, producing a chain reaction. This chain reaction is called a collision cascade (Fig 1.13). The collision cascade produces interstitial and vacancy point defects, where interstitials are atoms forced in between lattice sites, and vacancies are empty lattice sites. During the cascade, many point defects are produced which then interact with each other to either group into larger defects or to annihilate each other.

The most common measurement unit for radiation damage is displacements per atom (dpa), or how many times each atom has been displaced from its lattice site. At different temperature ranges and damage doses, simple point defects interact in many different ways to create five main types of radiation damage. They are: (1) radiation hardening and embrittlement, (2) radiation-induced precipitation and radiation-induced segregation causes phase instabilities, (3) irradiation creep and growth, (4) volumetric swelling from void formation, and (5) high temperature Helium embrittlement (Table 1.5). Radiation damage scales with dpa due to defect production, and scales with temperature as a fraction of the melting temperature T_M since it relates to when defects become mobile in the lattice.

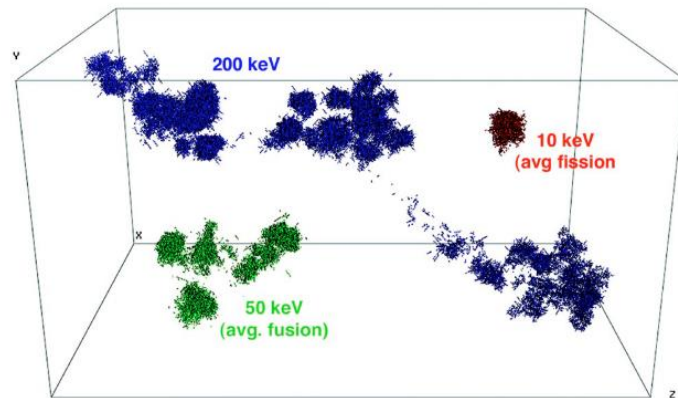


Figure 1.13: A comparison of MD-simulated displacement cascades in Fe for different PKA energies. 10 keV (red) is the average fission cascade and 200 keV (blue) is the average fusion cascade. (Zinkle 2005)

Table 1.5. The five main irradiation damage mechanisms as a function of melting temperature (T_M) and displacement damage (dpa). (Zinkle)

Effect	(T_M)	dpa	Explanation
Rad. Hardening	<0.4	>0.1	Point defects start to combine into dislocations, hardening and embrittling the material; can cause drastic failure
RIP, RIS	0.3-0.6	>10	Defect binding energies cause preferential growth of precipitates
Creep	<0.45	>10	Thermal and strain stresses cause material to swell and localized fracture
Swelling	0.3-0.6	>10	Point defects aggregate, voids and stacking faults cause swelling
He-embrittlement	>0.5	>10	He concentration becomes high enough to diffuse and create bubbles that diffuse to the grain boundaries; blocking the slip planes causing drastic failure

Radiation hardening and embrittlement occurs at $\text{dpa} > 0.1$ and at temperatures below $0.4 T_M$. This makes the material susceptible to extremely brittle fracture. The hardening mechanism is when dislocation lines are not able to travel and get obstructed on defects or precipitates.

Radiation-induced-precipitation (RIP) and radiation-induced-segregation (RIS) occur at $\text{dpa} > 10$ and temperatures in $0.3\text{-}0.6 T_M$ range. The segregation and precipitation of solutes/solvents can change the bulk properties of the material. RIP and RIS occurs when radiation damage produces sinks in the material. There can be a preferential bias to that sink for solutes, which would result in RIP or RIS at the sink. There can also be a preferential bias for the defect types, producing clusters of vacancies or interstitials at the sink which can form defect clusters. If the concentration is high enough, precipitates can form or the chemical composition of existing precipitates can change. RIS is most pronounced at intermediate temperatures since higher temperatures will have an annealing effect. Ultimately, it will decrease to a saturated value when defects are mobile enough where production and recombination reaches steady state.

Irradiation creep and growth occurs at $\text{dpa} > 10$ and temperatures $< 0.45 T_M$. The material swells in volume and becomes susceptible to brittle fracture. It is a similar concept to thermal creep, but radiation damage enhances the effect such that thermal creep becomes negligible. (Was) Radiation damage increases the number of interstitials and vacancies. Since there are more defects, more dislocation groups and lines are formed and nucleated through stress. The formation and migration of the dislocations results in the creep growth.

Void swelling occurs at $\text{dpa} > 10$ and temperatures in $0.3\text{--}0.6 T_M$ range (Fig 1.14a). It causes the material volume to increase and susceptibility to brittle fracture. For radiation resistant materials, void swelling is the hardest effect to avoid. The mechanism occurs at temperatures high enough to let vacancies diffuse and combine into voids, but low enough that interstitials are not mobile so there is minimal recombination.

Helium embrittlement occurs at $\text{dpa} > 10$ and high temperatures $> 0.5 T_M$ (Fig 1.14b). Helium is introduced into the material either from neutron capture reactions or is implanted by the plasma. As a noble gas, helium diffuses easily through a matrix material toward sinks. Typically, He nucleates into bubbles at grain boundaries, turning them into “swiss cheese.” This provides a fracture pathway along grain boundaries which embrittles the material. The bubbles also prevent slipping and stress relief at the grain boundaries. Additionally, the He bubbles promote swelling. For fast fusion neutrons, swelling in stainless steel is shown to be maximized at fusion-relevant He/dpa ratios. (Zinkle 2013)

The plasma is incident on the surface at $\sim 3k_B T_e$, and can implant into the material at the keV energy range. This creates an “implantation layer” where the particles come to rest, creating a near-surface layer with high concentration of solutes. Only the near-surface retention is evaluated because He, which is insoluble, will feel the free surface (first wall) and preferentially move towards it instead of deeper into the matrix. This encourages He to over-nucleate, creating very strong sink strength in the implantation layer. He would migrate towards the bulk, but would be trapped by vacancies, preferentially forming He-vacancy clusters, which then act as sinks for the D interstitials. Molecular dynamics simulations identified a binding energy of 2-3 eV which suggests a strong trapping of H in the He-matrix denuded zone. Hence, helium-vacancy complexes at the near surface reduces hydrogenic retention by creating a barrier to bulk diffusion.

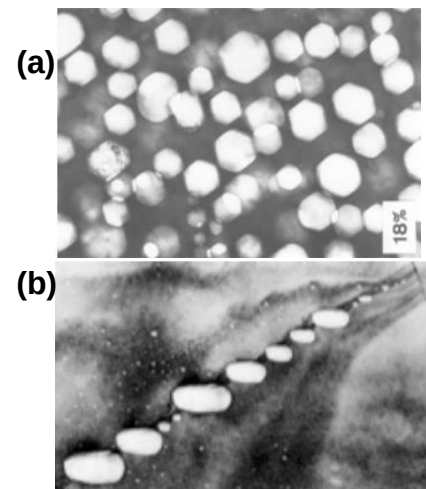


Figure 1.14: (a) void swelling (b) He embrittlement along grain boundaries (Zinkle 2005)

This increases hydrogenic trapping at the cavities, which quickly saturates the implantation layer thereby increasing desorption at the plasma-facing surface.

Due to the HHF and HPF, plasma is harsh and demanding on PFMs. PFMs need thermomechanical resistance, radiation resistance, chemical compatibility, and nonperturbative to the plasma.

1.3 FUSION MATERIALS

The plasma-facing wall is typically split into two sections since they have very different roles. Therefore, fusion materials are typically broken into two categories: first wall and divertor armor. Since the first wall has such a large surface area, it is critical that the impurity generation is low. Hence, low-Z materials, such as Be or C, are used. However, the divertor needs to survive heat fluxes that are at the limit of solid materials while maintaining thermomechanical integrity. In addition, PFMs need to be radiation-resistant and regulation-compliant.

Desirable material properties include: high strength, good ductility, crack resistant, creep resistant, and low hydrogenic solubility. Preferred thermal properties include high thermal conductivity, high melting point, low vapor pressure, thermal shock resistance, and low thermal expansion. These thermomechanical properties are necessary to withstand the HHF. The divertor is typically made of high-Z refractory materials such as tungsten and molybdenum. (Zinkle 2005, Zinkle 2013, Zinkle 2014, Linke 2019, Coenen 2017, Wirtz 2016)

For example, If the heat flux is 10 MW m^{-2} and the constrained wall thickness is 1 mm, the thermal stress in iron is 1,160 MPa but in tungsten is 130 MPa. Hence, divertor materials need to have good thermal shock resistance.

Required radiation properties include sputtering and blistering resistant, swelling resistant, and low transmutation cross sections. These properties are necessary for the tokamak and reactor site to remain radiologically safe. If the materials are selected properly, it is possible for fusion radiological waste to be in Class C waste disposal. This means the radioactivity is so low the site only needs ~100 years to reach tolerable radiation levels. PFMs need to be chemically compatible with the plasma, blanket

design, and coolant, particularly to O, C, H, N and permeable to T. Tritium needs to be recycled. While meeting these requirements, it must be economically advantageous, commercially available, and machinable. (Dolan 2013)

Because of those qualifying properties, a safety analysis of all the elements in the periodic table found a few families of materials that are most suitable for fusion energy systems. (Piet 1991) These families include SiC/SiC ceramics fiber composites, refractory alloys, and reduced activation ferritic/martensitic (RAFM) steels (Table 1.7). Current material candidates are tungsten W, tungsten fiber composites W_f/W, beryllium Be, carbon fiber composites CFC, and silicon carbide ceramic composites SiC/SiC.

(Zinkle 2005) Most fusion materials research is being conducted on these materials. (Zinkle 2013, Zinkle 2014, Linke 2019) Materials research is predominantly limited by availability of testing facilities.

Table 1.7. Fusion material families determined by ARIES program (Dolan 2013, Piet 1991)

Material Family	Elements
Ceramic composites	SiC
Refractory alloys	V, Cr, Ti, Si
RAFM steels	Fe, Cr, W, V, Ta

1.4 CONCLUSION

The fusion reactor environment is incredibly challenging for materials. Plasma subjects materials to high heat flux, high particle flux, thermomechanical stress, neutron irradiation, and helium implantation. The main plasma-material interaction mechanisms are: high heat flux, high particle flux, neutron irradiation, and particle implantation. Plasma-facing components need high thermal conductivity, sputtering resistance, low radiative losses, irradiation resistance, and favorable high-temperature properties while remaining economical and machinable on a large scale.

To develop and qualify advanced fusion materials, HHF and HPF test facilities are important, both for steady-state and transient level experiments. Current research focuses on qualifying materials, mostly with single-variable testing due to lack of fusion-relevant test facilities for multivariable testing. For full fusion materials qualification, it

will need to undergo multivariable testing for radiation damage, HHF, HPF, and cyclic thermomechanical shock.

Hence, this project was motivated by diagnostic development under HHF and HPF expected from steady state and transient plasma events. Current tokamaks and plasma devices are not able to produce ITER- or reactor-relevant HHF and HPF. This is mainly because transient plasma events expel a significant fraction of the plasma power over a small area in a short time. Additionally, most PMI studies rely on *in vacuo*, *post mortem*, or *ex situ* measurements. Hence, fusion testing facilities are needed for *in situ* diagnostic development at reactor-relevant HHF and HPF.

At Oak Ridge National Laboratory (ORNL), an electrothermal arc plasma source is being used to develop an *in situ* laser diagnostic for erosion studies. The plasma pulse is $t = 0.1 - 2,000 \mu\text{s}$ long, with a heat flux $q \sim 0.1\text{-}2 \text{ GW m}^{-2}$ and particle flux $\Gamma \sim 10^{22} \text{ particles m}^{-3}$. Thus, t , q , and Γ are all similar to ELM transient events. (Gebhart 2018) Recently, the vacuum chamber and target plate of the ET-Arc were modified to accommodate more diagnostics for diagnostic development. Modifications include the vacuum chamber, target plate design, diagnostic suite, and triggering system.

CHAPTER TWO

HIGH HEAT FLUX TEST FACILITIES

2.1 OVERVIEW

There are many types of high heat flux (HHF) test facilities (HHFTF). They can simulate steady-state and/or transient heat loads. ITER is expected to produce steady state heat fluxes of 0.1 MW m^{-2} on the first wall and 10 MW m^{-2} at the divertor. Transients increase the heat flux on the first wall to $\sim 2.5 \text{ MW m}^{-2}$ and $\sim 2 \text{ GW m}^{-2}$ on the divertor. The thermal loads will also be cyclic due to the pulsed nature of tokamaks. Hence, high heat flux testing, thermomechanical loading, and thermal fatigue tests are critical for qualifying PFCs.

Some types of HHF test facilities are linear plasma devices, electron beams, x-ray sources, high powered laser irradiation, neutral beam heating, quasi-steady state plasma accelerators, applied-field magnetoplasma dynamic thruster, plasma arc lamps, infrared heaters, and electrothermal arc sources. A few tokamaks also specialize in PMI and PFM testing, including WEST in France and EAST in China. Each type has a unique operating regime and are optimized for different applications. Across the world, there are several of each type of HHF devices, so this chapter is by no means an exhaustive list of existing HHF test facilities. A collection of HHFTF is listed in Table 2.1. The most common HHFTF are linear plasma devices and electron beam facilities. In fusion, there are three synergistic damage mechanisms: thermal loads, plasma exposure, and neutron irradiation.

Linear plasma devices (LPD) provide two of these phenomena with plasma-induced high heat flux, allowing realistic PMI studies. LPDs also have a lot of design flexibility, allowing for steady state and/or transient heat flux and particle flux. Fusion-relevant first wall or divertor conditions can be well-simulated. Additionally, there is a lot of diagnostic access in the source region, plasma-target interface, and the target chamber. This allows for well-characterized plasmas, plasma near the surface, and the surface itself in fusion-relevant scenarios. In Section 2.2, a prominent linear plasma device, Magnum-PSI, will be discussed in detail.

Electron beam (EB) facilities are common HHFTF because they have low capital cost, well-known technology, highly reproducible conditions, high beam control capability, and low beam variation. The sample chambers can be quite large, making EB facilities excellent test beds for large components. EB devices can provide high heat flux with high repetition rates, making them ideal for cyclic thermal loading, thermal-mechanical shock, and thermal fatigue tests. In Section 2.3, a prominent electron beam HHF facility, JUDITH-II, will be discussed in detail.

In Table 2.1, the “other” category showcases the variety in HHF devices. VISION-I (Versatile Instrument for Studies of ION Interactions) is a plasmatron device in Belgium which is a steady state device to study hydrogenic isotopes in mixed materials. (Uytendhouwen 2008, Zayachuk 2011, Zayachuk 2014) GLADIS (Garching Large Divertor Sample test facility) is an ion beam facility at Maxwell-Planck Institute in Germany and produces HHF of $3\text{--}55 \text{ MW m}^{-2}$ for $0.1\text{--}45 \text{ s}$. (Greuner 2007, ipp.mpg.org) A quasi-stationary plasma accelerator (QSPA) was built in the Ukraine with parameters of $n \sim 10^9 \text{ m}^{-3}$, $\Gamma \sim 10^{22} \text{ m}^{-2}\text{s}^{-1}$, $q \sim 1.7 \text{ MWm}^{-2}$ for pulses $t = 250 \text{ }\mu\text{s}$. (Garkusha 2017) Two infrared $0.2\text{--}1.4 \text{ }\mu\text{m}$ PALS (plasma arc lamps) at ORNL test small divertor components with steady state heat fluxes of either 4 or 27 MW m^{-2} with pulse durations of 30 s . (Charry 2015, Sabau 2014) An applied field magnetoplasmadynamic (AF MPD) device in Korea has been developed recently as a divertor simulator, with parameters to-date of $T_e = 4 \text{ eV}$, $n_e = 1.4 \times 10^7 \text{ m}^{-3}$, $q = 2.5\text{--}4 \text{ MW m}^{-2}$, and $\Gamma = 10^{22}\text{--}10^{23} \text{ m}^{-2}\text{s}^{-1}$. (Chai 2020)

2.1.1 Table

Table 2.1		Summary of high heat flux test facilities. Tokamak (green) LPD (blue) EB (purple) Arc (red) Other (orange)				
Device	Location	q (MW m ⁻²)	T _e (eV)	n _e (m ⁻³)	Γ (m ⁻² s ⁻¹)	Unique Features
ITER divertor	France	20	1-10	10 ²⁰ - 10 ²¹	10 ²⁴ -10 ²⁵	For comparison
Magnum-PSI	Netherlands	< 200	0.1- 10	10 ¹⁹ - 10 ²¹	10 ²³ -10 ²⁵	HHF, HPF, IBA
Pilot-PSI	Netherlands	< 600	1-5	10 ²⁰ - 10 ²¹	10 ²⁴	IBA, SS & transient HHF
PSI-2	Germany	0.2	1-40	10 ¹⁷ - 10 ¹⁹	10 ²¹ -10 ²²	Toxic & activated materials
JULE-PSI	Germany	0.1-2	1-20	10 ¹⁷ - 10 ¹⁹	10 ²²	Toxic & activated materials, SS & transient HHF
PISCES-A	USA	0.00125	<10	10 ⁷	10 ²¹ -10 ²³	Ion acceleration, large plasma area
PISCES-B	USA		3-50	10 ¹⁷ - 10 ¹⁹	10 ²¹ -10 ²³	Be compatible
STEP-A	China		< 40	10 ¹⁶ - 10 ¹⁸	10 ²⁰ -10 ²²	
NAGDIS-II	Japan				~10 ²²	Detachment studies
MPEX	USA	10	< 15	10 ¹⁹	> 10 ²⁴	Activated samples, high fluence
JUDITH-II	Germany	2.5-1,510	-	-	-	Activated materials
FE200	France	0.1-100	-	-	-	
EB1200	USA	8.7-120	-	-	-	
TSEFEY	Russia		-	-	-	
IDTF	Russia	5-20	-	-	-	
KoHLT	Korea	5-20	-	-	-	Be compatible
JEBIS	Japan		-	-	-	
HELCZA	Czech Rep.	<20	-	-	-	
SIRENS	USA	<70	1-6	10 ²⁵ - 10 ²⁷		
VISION-I	Belgium	-	-	-	10 ²⁰ -10 ²¹	Retention studies
GLADIS	Germany	3-55	-	-	-	Proton accelerator
QSPA	Ukraine	< 1.7		10 ⁹	10 ²² -10 ²⁶	Short μs pulses
AF-MPD	Korea	4	4	10 ⁷	4x10 ²²	
PALs	USA	4-27	-	-	-	Low cost

2.2 LINEAR PLASMA DEVICES (LPD)

Linear plasma devices (LPD) are typically used to simulate divertor plasma conditions. Both the plasma and the PMI can be studied with these devices. LPD typically operate in steady state regimes. Magnum-PSI (plasma-surface interactions) is a cutting-edge LPD facility located at the Dutch Institute for Energy Research (DIFFER). It provides complementary capabilities with the other two facilities there, Pilot-PSI LPD and the ion beam facility (Fig 2.1).

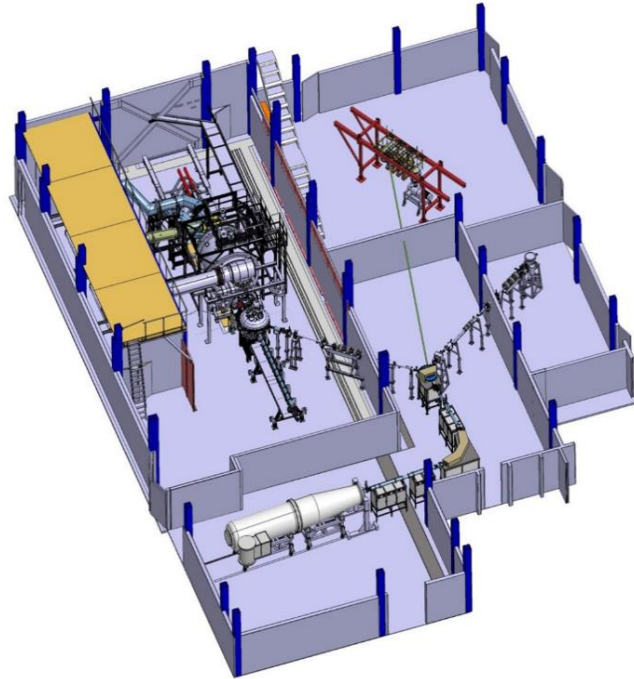


Figure 2.1: Overview of the DIFFER facility with Magnum-PSI at the top left, Upgraded Pilot-PSI at the top right, the ion beam analysis station at the lower right, and the Ion Beam Facility at the lower left. (differ.nl)

2.2.1 Magnum-PSI

The Dutch Institute for Fusion Energy Research (DIFFER) has three devices: Magnum-PSI, Pilot-PSI, and an ion beam facility (IBF) (Fig 2.1). (differ.nl) Both Magnum and Pilot are integrated with the IBF, allowing for increased *in vacuo* capabilities. Both Magnum and Pilot are LPDs with wall-stabilized DC cascaded arc plasma sources. (van Eck 2018) Both target chambers interface with the ion beam facility for additional surface analysis). Magnum-PSI is designed to replicate steady-state divertor-like HHF and HPF with long exposure times. (Morgan 2017, van Eck 2019, de Temmerman 2013).

The design specs for Magnum were centered around ITER-relevant parameters. These parameters include steady state heat flux $>10 \text{ MW m}^{-2}$, transient heat flux $\sim 1 \text{ GW m}^{-2}$ repeating $\sim 100 \text{ Hz}$, electron density $n_e \sim 10^{19}\text{-}10^{21} \text{ m}^{-3}$, electron temperature $T_e \sim 0.1\text{-}10 \text{ eV}$, and particle flux $\sim 10^{23}\text{-}10^{25} \text{ m}^{-2}\text{s}^{-1}$. (van de Pol 2018)

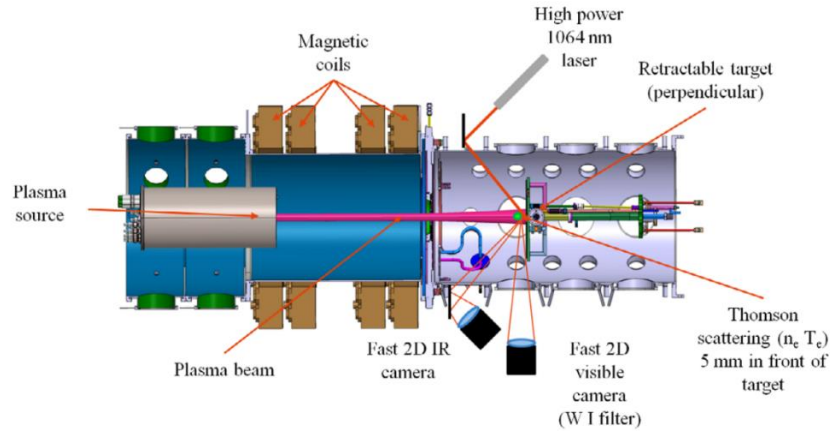


Figure 2.2: A schematic of Magnum-PSI showing the diagnostics. (Morgan 2014)

2.2.1.1 Diagnostics

The diagnostic suite includes optical emission spectroscopy (OES), fast visible cameras (FVC), resistive bolometry, Thomson scattering (TS), and IR camera thermography. (van de Pol 2018) The IBF has been connected to the target exchange and analysis chamber on Magnum, same as Pilot, allowing for an array of ion beam analysis techniques (Table 2.2).

Table 2.2. The ion beam analysis capabilities at DIFFER. (differ.nl)

Techniques	Elements	Sensitivity	Probing Depth	Depth Resolution
Rutherford Backscattering Spectrometry (RBS)	$Z \geq 5$	10 ppm (large Z) a few at.% (C on c-Si substrates)	1-2 μm	5-20 nm
Elastic Recoil Detection (ERD)	H, D	H: 0.1 at.%, D: 100 ppm	500 nm	20 nm
Nuclear Reaction Analysis (NRA)	$Z = 1-15$	Isotope dependent $10^{14}-5 \times 10^{15} \text{ at.cm}^{-2}$ 0.1-2 at.%	Isotope dependent, Several microns	Isotope dependent
Particle Induced X-ray Emission (PIXE)	$Z \geq 13$	Z dependent, Bulk: 0.1-100 ppm, Tin foils: $10^{13}-5 \times 10^{15} \text{ at.cm}^{-2}$	Few tens of microns	Poor
Particle Induced Gamma Emission (PIGE)	$Z = 1-15$	1-10 ppm for F, Li, B, and N	Few tens of microns	Isotope dependent
Elastic Backscattering Spectrometry (EBS)	$Z \geq 2$	10 ppm (large Z) – 1 at.%	10 μm	20-300 nm

2.2.1.2 Heat Flux

Magnum is equipped for both steady-state and transient heat flux capabilities. At DIFFER, heat flux is calculated using the THEODOR code for 2D inverse heat transfer. (Tokunaga 2005) The THEODOR code accounts for a surface layer on top of a substrate with two different thermal conductivities of the layer. (Hermann 2001) This analysis prevents overestimating the surface temperature, and underestimating the incident heat flux. It also accounts for sheath effects, where the heat flux is defined as

$$q = \gamma k_B T_e \Gamma_{se} \approx \left(\frac{2}{3m_i} \right)^{1/2} \gamma n_e (k_B T_e)^{3/2} \quad (2.1)$$

where γ is the sheath transmission coefficient (including ions), Γ_{se} is the ion flux to the surface, and m_i is the ion mass. Heat flux calculations are based on TS T_e and n_e measurements, and γ was experimentally measured to be 7.7 (Morgan 2014). Information on whether the THEODOR code accounts for vapor shielding was not found.

For steady-state operations, typical average heat fluxes are 10 MW m^{-2} at 1.2 T, 4 MW m^{-2} at 0.8 T, and 1 MW m^{-2} for 0.4 T (Fig 2.3). (van de Pol 2018) Peak heat fluxes will be higher than average due to the centrally-peaked n_e and T_e profiles. With the new superconducting magnet, steady-state operations can produce heat fluxes up to 50 MW m^{-2} (Fig 2.3). (van Eck 2018, van Eck 2019) The operational space

with the new magnet has only been characterized up to 1.6 T out of the 2.5 T capability.

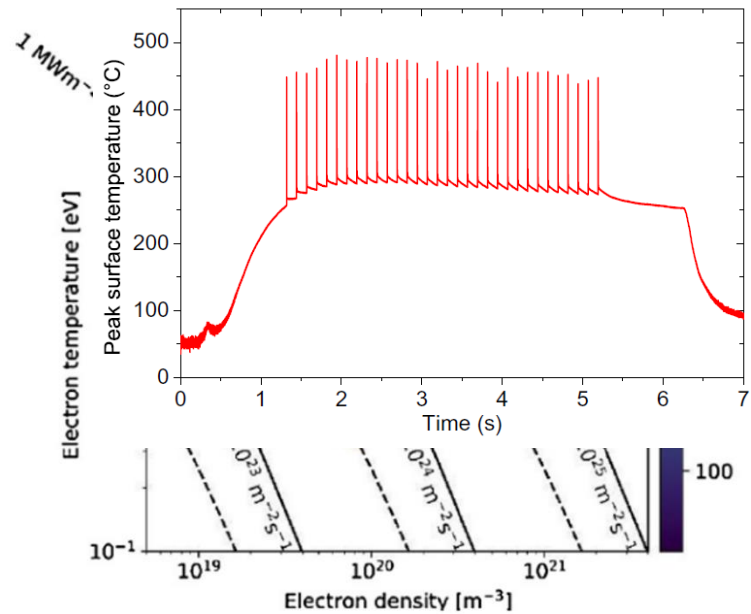


Figure 2.3: Magnum-PSI heat flux as a function of electron temperature, electron density, and source current. Heat fluxes are the curved, dashed lines, and particle fluxes are the solid black lines. (van Eck 2019)

Magnum also has the capability to simultaneously produce ELM-like heat fluxes of $\sim 1 \text{ GW m}^{-2}$ in addition the steady-state operation. This is accomplished by using a high-power welding laser to create 0.1-3 ms pulses. (van Eck 2019) The laser will increase heat fluxes, but not particle fluxes. Transient studies are typically done on Pilot-PSI because it has an eight-capacitor bank that can be used to superimpose transient heat fluxes on the steady-state pulse (Fig 2.4). These pulses are ~ 1 -1.5 ms in duration and can produce heat fluxes 275-625 MW m^{-2} (Morgan 2017) at 10 Hz. These capacitive discharges increase both heat fluxes and particle fluxes. The system increases n_e from 10^{20} m^{-3} to 10^{21} m^{-3} , T_e from 1 to 5 eV, and q from 10 to 130 MW m^{-2} . (Morgan 2014)

Figure 2.4: Temperature evolution of a tungsten target in Pilot-PSI. Transient pulses of 55 J at 8 Hz were superimposed on the steady state cascaded arc discharge. (Morgan 2017)

2.2.1.3 Particle Flux

Particle fluxes are calculated from TS-measured T_e and n_e profiles using the Bohm criterion for the sheath-edge density to be half the upstream density. The plasma column is assumed to be a symmetric, 2D-Gaussian with half the total power within the FWHM. The superconducting magnet was installed recently in 2018, allowing for high fluence exposures with high diagnostic access. The magnet was upgraded to a superconducting magnet which increased the magnetic field capabilities from 1.6 T to 2.5 T, which better confines the plasma for steady state operations. (van Eck 2018) With CuFe, cooled magnets, Magnum reached particle fluxes of $\sim 10^{21} \text{ m}^{-2}\text{s}^{-1}$. With the increased magnetic field from the superconducting magnet, particle fluxes of $\sim 10^{25} \text{ m}^{-2}\text{s}^{-1}$ were achieved. (Morgan 2014) With these HPF, it is possible to reach the ITER 12-month divertor lifetime fluence of $\sim 10^{30} \text{ m}^{-2}$ in ~ 30 hours of exposure time. This is the first device that can perform full lifetime PFC tests on a reasonable timescale. In Morgan 2018, the longest published exposure time is 6.5 hours, while another experiment has been conducted for 19.7 hours (unpublished). (van Eck 2019)

2.2.1.4 Significant Results and Contributions

Significant contributions from DIFFER include PFC studies with Magnum-PSI and Pilot-PSI. Magnum exposes PFC samples to ITER-like conditions for expected lifetime studies. Magnum-PSI is one of the top facilities world-wide for continuously operating lifetime studies of HHF and HPF. The transient simulator Magnum uses is a high-powered Nd-YAG laser, which achieves HHF but not HPF, so the transient simulation is not realistic. The beam size is small and decreases with increased heat flux, so Magnum is not able to conduct HHF testing of large components. Pilot-PSI is able to couple transient level HHF and HPF with steady state operation for PFC studies.

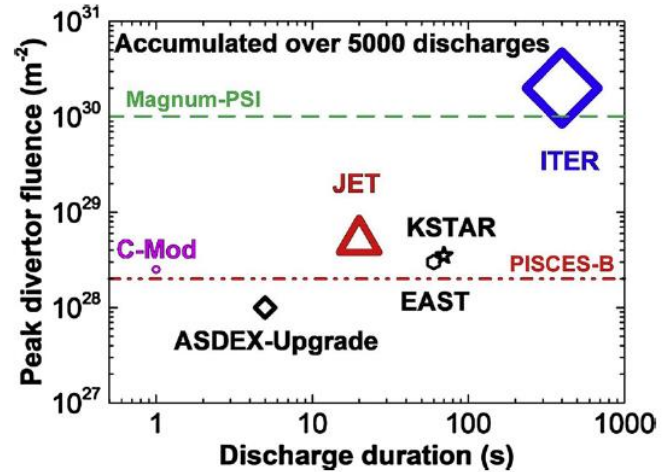


Figure 2.5: Peak ion fluence in the divertor for different devices given 5,000 typical plasma discharges. (van Eck 2019)

2.3 ELECTRON BEAM (EB)

Electron beam (EB) facilities are relatively common HHF facilities due to commercial availability, low capital cost, high repetition rates, large test samples, and control over beam size and power. (Hirai 2005) They have flexible operations for cyclic exposure or long exposures and homogenous heat loading over large areas. Hence, EB are an excellent option for high thermal loads and cyclic power loading, allowing for thermal fatigue and thermal shock testing. (Hirai 2005, Bobin-Vastra 2005)

2.3.1 JUDITH-II

The German facility Forschungszentrum Julich has four HHF devices: JUDITH-1, JUDITH-2, PSI-2, and JULE-PSI (Fig 2.6). The JUDITH EB facility is the JUelicher Divertor Test facility in Hot cell. PSI-2 and JULE-PSI are linear plasma devices, where JULE-PSI is designed based on PSI-2 and under construction. JULE-PSI will also be able to handle hot and toxic samples. JUDITH-1 and JUDITH-2 (Fig 2.7) are electron

beam facilities, where JUDITH-2 is an upgraded model of JUDITH-1 that is located in a hot cell, allowing for toxic and activated sample testing. The vacuum vessel can hold sample sizes of $0.5 \times 1 \text{ m}^2$.

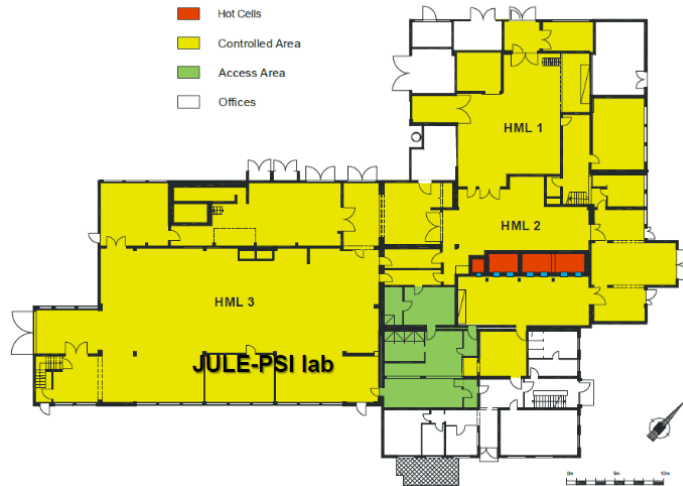


Figure 2.6: HML-1: JUDITH 2, HML-2 JUDITH-1 Upgrade, HML-3 JULE-PSI and Be analysis station.

The high temperature materials laboratory at Forschungszentrum Julich. JUDITH-1 and -2 are electron beam facilities, where JUDITH-2 is equipped for activated and toxic materials. PSI-2 and JULE-PSI are linear plasma devices, where JULE-PSI is the same design as PSI-2 but equipped for activated and toxic materials. (Kreter 2013)

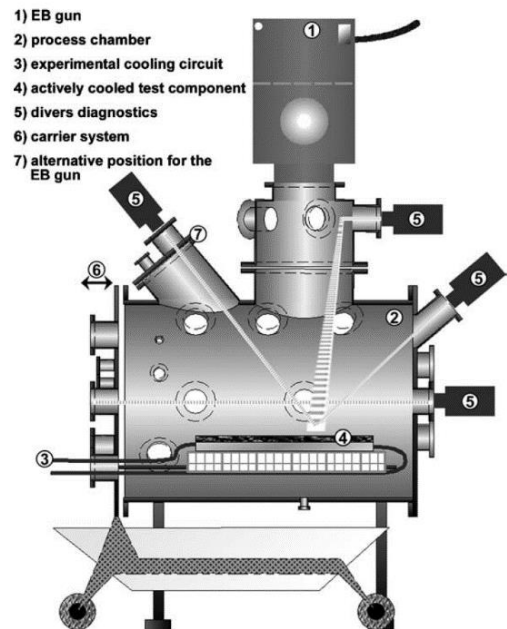


Figure 2.7: JUDITH-2 schematic (Majerus 2005)

2.3.1.1 Diagnostics

The JUDITH diagnostic suite includes IR camera thermography, three pyrometers ($200\text{-}1100 \text{ }^\circ\text{C}$; $550\text{-}1600 \text{ }^\circ\text{C}$; $1000\text{-}3500 \text{ }^\circ\text{C}$), thermocouples for surface temperature measurements, and photodiodes for velocity of released particles. (Majerus 2005)

2.3.1.2 Heat Flux

The JUDITH-II device is located in a hot cell. It is used for both cyclic and quasi-stationary thermal loads at $\sim 2.5 \text{ MW m}^{-2}$ for expected first wall heat flux exposures. (Weber 2015) Using a beam-scanning mode, it is also capable of ELM-like thermal loads with power densities in the GW m^{-2} range and pulse durations of a few milliseconds. (Unterberg 2011, Majerus 2005)

2.3.1.3 Particle Flux

EB facilities obviously produce electron beams which do not provide significant or relevant particle fluxes. Although plasma environments have electron flux which pass energy to the wall, PMI is mainly triggered by neutron, ion, or neutral particle fluxes. They also have very low densities and number of particles, so the fluence generated is very small. Hence, EB facilities cannot be used for particle flux exposures.

2.3.1.4 Significant Results and Contributions

Most ITER components, including monoblock sets and divertor cartridges, are tested in EB facilities due to availability and flexibility. EB vacuum chambers can host larger samples than typical laboratory scale plasma devices. EB repetition rates are beneficial for cyclic thermal fatigue testing which is essential in qualifying new reactor components.

2.4 ELECTROTHERMAL ARC PLASMA SOURCE (ET-ARC)

Electrothermal arc plasma sources (ET-Arc) are typically used for HHF testing (Gilligan 1993, Almousa 2016), mass accelerators (Edwards 1995), and aerospace plasma thrusters (Edamitsu 2006). Compared to LPDs and EBs, this type of ET-Arc sources are not very common in fusion applications. Other types of arc sources, like the Magnum cascaded arc source for steady state plasmas, or the capacitive discharge on Pilot for transient plasmas, are more common in fusion applications.

The ET-Arc produces plasma by sending a discharge through a small capillary liner. The plasma is produced when large power is discharged into the capillary liner which ablates and is Joule heated into plasma. As the cylindrical capillary ablates, a strong pressure gradient is created along the capillary, resulting in convective plasma flow out of the capillary towards the material sample. At North Carolina State University (NCSU), researchers have studied ET-Arc sources for a variety of purposes, including HHF testing for PMI erosion studies and tokamak fuel-pellet launchers. In Chapter Three, modifications to the ET-Arc system at Oak Ridge National Laboratory (ORNL) are discussed. The system this project used is based on the SIRENS device developed at NCSU in the 1980s. (Bourham 1989, Gilligan 1993, Hankins 1993)

2.4.1 SIRENS

SIRENS (Surface Interaction Research Experiment at North Carolina State) is an ET plasma device designed and built at North Carolina State University in 1987. (Bourham 1989, Auciello 1987) The goal of SIRENS is to study HHF effects on materials.

SIRENS consists of a 300 μF capacitor with stored voltage of 10 kV. It discharges through a spark-gap switch to the cathode capillary in the ET source. The capillary is 9 cm long with an inner diameter of 4 mm and is made of Lexan. A trigger to the spark-gap switch causes breakdown in the spark gap, which produces a plasma arc.

This development of SIRENS was a US Department of Defense project so most of the detailed information is published in inaccessible DOD reports. (US Strategic Defense Command Contract DASG60-90-C-0028) More recent publications on this project focus on the theory and modelling of ET capillaries and mass accelerators. (Almoussa 2016) From this project, NCSU developed two codes: ETFLOW and (Zhagloul 2004) ZEUS (Gilligan 1993). ETFLOW is one-dimensional and time-dependent while ZEUS is zero-dimensional and time-dependent. (Winfrey 2012, Sharpe 2001)

2.4.1.1 Diagnostics

The comprehensive diagnostic suite is not published but the existing publications refer to high voltage measurements and optical emission spectroscopy. The capacitor discharge is measured with high voltage probes and with a Rogowski current monitoring coil. The plasma arc is shorter than the capacitor discharge, so the power absorbed in the plasma is simply calculated by measuring the energy in the capacitor before and after the shot. (Bourham 1989) OES using a multichannel analyzing system was developed to measure plasma temperature $T_e \sim 1\text{-}2\text{ eV}$ and composition.

2.4.1.2 Heat Flux

SIRENS can produce HHF $< 70\text{ GW m}^{-2}$ with a pulse duration of from the microsecond to millisecond range, depending on the pulse-forming network. (Sharpe 2000). The maximum pulse duration is 0.2 ms. SIRENS is a low temperature

$T_e = 1\text{-}6\text{ eV}$ and high density $n_e = 10^{25}\text{-}10^{27}\text{ m}^{-3}$ device. (Gilligan 1993, Hankins 1993, Almousa 2016) This is similar to the source at ORNL – see Section 3.1.1.

2.5 CONCLUSIONS

There are many different types of HHF devices. The main concepts are based on linear plasma devices, electron beam devices, and electrothermal arc sources. As ITER nears completion, components will need to be tested under fusion-relevant conditions. High heat flux testing, thermomechanical loading, and thermal fatigue tests are critical for qualifying PFCs. Each device type has a specific operating space and is optimized for different applications.

Hence, this project was motivated to develop improved diagnostics under HHF and HPF conditions expected from transient plasma events. Current tokamaks and plasma devices are not able to produce ITER- or reactor-relevant HHF and HPF. This is mainly because transient plasma events expel a significant fraction of the plasma power over a small area in a short time. Additionally, most PMI studies rely on *in vacuo*, *post mortem*, or *ex situ* measurements. Hence, fusion testing facilities are needed for *in situ* diagnostic development at reactor-relevant HHF and HPF.

At Oak Ridge National Laboratory (ORNL), an electrothermal arc plasma source is being used to develop an *in situ* laser diagnostic for erosion studies. The ET-arc source produces plasma heat flux of $\sim 0.1\text{-}2\text{ GW m}^{-2}$ and particles flux of $\sim 10^{22}\text{ particles m}^{-3}$ which is similar to ELM transient events. (Gebhart 2018) Recently, the vacuum chamber and target plate of the ET-Arc were modified to accommodate more diagnostics for diagnostic development. Modifications include the vacuum chamber, target plate design, diagnostic suite, and triggering system.

CHAPTER THREE

EXPERIMENTAL

3.1 OVERVIEW

An electrothermal-arc plasma source (ET-Arc) has been developed at Oak Ridge National Lab (ORNL). It produces heat and particles fluxes similar to ELMs on divertor plasma facing components in tokamaks. Recently, the vacuum chamber and target plate of the ET-Arc were modified to accommodate the development of two laser-based diagnostics. Modifications were made to the vacuum chamber, target plate design, and triggering system. These modifications were made to accommodate two new laser diagnostics for *in-situ* diagnostic development and testing.

The ET-Arc source plasma pulses are created by a ~6 kV capacitive discharge through the source capillary in the source. The source liner is ablated due to Joule heating to form a high-velocity unidirectional plasma jet which exits the capillary to impact the target. The current discharge circuit configuration forms pulse lengths of 0.1-2,000 μs at full-width half-maximum duration and delivers heat fluxes of 0.25-2.1 GW m^{-2} . The plasma was characterized with optical emission spectroscopy and analyzed with the collisional radiative model of He I line ratios. The electron temperature and electron density range from $T_e \sim 1\text{-}5 \text{ eV}$ and $n_e \sim 10^{22}\text{-}10^{28} \text{ m}^{-3}$ respectively. (Coburn 2020)

The diagnostic suite includes two Tektronix high voltage probes measuring the voltage on the capacitor and at the source anode, a Pearson current monitor to measure discharge current, and a FLIR SC4000 infrared camera to measure heat flux on the target. The ET system was modified to include two laser diagnostics in development which will more thoroughly characterize the plasma and the target surface. A portable diagnostic package, including Thomson scattering (TS) and optical emission spectroscopy (OES), will be developed on the ET-Arc source. The spectroscopy suite measures electron temperature, electron density, ion temperature, ion density, and ion velocity of the plasma. (Biewer 2020) Both TS and OES are well-developed measurement techniques for measuring plasma parameters. Another novel laser

diagnostic, digital holography, will be implemented on the ET-Arc source as well. Digital holography uses laser interferometry to reconstruct 3D holographic images of a target. (Thomas 2014, Thomas 2016, Biewer 2018, Smith 2020) By monitoring the surface topology, before-and-after comparisons allows for *in-situ* net erosion measurements. *Ex-situ* digital holography characterization of stainless-steel targets exposed to the ET-Arc source show a surface erosion of ~150 nm per shot. (Smith 2020) The triggering system will be revised and optimized to synchronize with the laser diagnostics. To best accommodate the DH signal, the vacuum chamber was designed so the laser has a perpendicular view of the target. The target was designed to include a “step” that creates a region protected from direct plasma incidence, reducing or elimination erosion. This shaded area is to be used as a fiducial during the early *in-situ* experiments.

A PMI erosion thesis study was conducted by Coburn at NCSU. (Coburn 2020) His experiments were conducted on both DIII-D and the ET-Arc source, but only the ET-Arc results will be summarized here. Net erosion measurements were performed on several candidate PFMs: high purity β -3C CVD silicon carbide, tungsten, and Ti_3SiC_2 and Ti_2AlC MAX phase ceramics. The samples were cylindrical with either 1.6 mm or 6 mm in height, and a 6 mm diameter surface that was exposed to the plasma. The description of the target holder did not include a heat sink, so it is assumed the target temperature was allowed to increase during plasma exposure. A new, multi-technique *ex situ* measurement approach was used, including focused ion beam spectroscopy (FIB), scanning electron microscopy (SEM), and atomic force microscopy (AFM). Trenches with depth markers were created in the FIB and characterized in the SEM before exposure. Then the samples were exposed in the ET-Arc source to heat fluxes of $0.9\text{-}1\text{ GW m}^{-2}$ with 1 ms plasma pulses for either one pulse or multiple (five) pulses. The erosion rates are estimated from *ex situ* comparisons of unexposed surfaces to the single exposure, and again for the unexposed surface to the multiple exposures target. The erosion is calculated as erosion depth (μm) over time (Δt) exposed to the plasma. Percent error and/or standard deviation as a percent vary from 7-34%. The tungsten samples melted, showing a melt-layer depth of $<7\text{ }\mu\text{m}$ and molten pits of $2\text{-}10\text{ }\mu\text{m}$ in

diameter. The W erosion rate for the single exposure was $117 \mu\text{m s}^{-1}$ and $32 \mu\text{m s}^{-1}$ for multiple exposures. The MAX phase ceramics suffered surface fracture and ejected material, with a damage layer $>4 \mu\text{m}$ for Ti_2AlC and $>11 \mu\text{m}$ for Ti_3SiC_2 . The Ti_3SiC_2 erosion rate for single exposure was $80\text{-}775 \mu\text{m s}^{-1}$ and $584 \mu\text{m s}^{-1}$ for multiple exposures. The Ti_2AlC erosion rate for single exposure was $85\text{-}470 \mu\text{m s}^{-1}$ and $186 \mu\text{m s}^{-1}$ for multiple exposures. SiC was the least damaged with no surface fracturing and erosion rate for single exposure of $90\text{-}128 \mu\text{m s}^{-1}$ and $24\text{-}29 \mu\text{m s}^{-1}$ for multiple exposures. (Coburn 2018, Coburn 2020)

Moving forward, proof-of-principle plasma measurements for the portable diagnostic package and *in situ* net erosion experiments for digital holography will be conducted. The ET-Arc source will provide the capability for future plasma-material interactions experiments on advanced fusion materials.

3.2 ET-ARC PLASMA SOURCE

3.2.1 The Plasma Source

The electrothermal (ET) arc is a pulsed plasma source that operates in the ablative arc regime. The main source components are an electrode, an ablating liner, a ground housing, and an insulating liner (Fig 3.1).

The ET-Arc source plasma pulses are created by a capacitive discharge. The $340 \mu\text{F}$ capacitor can be charged up to 10 kV for 17 kJ of stored energy, but is typically operated at $\sim 6 \text{ kV}$. The capacitor is discharged through a spark gap switch, which relies on an arc forming between the sides of the switch. (Fig 3.2) This pulse is on the order of hundred microseconds. After the switch is the pulse-forming network which introduces an inductor and resistor to form an RLC

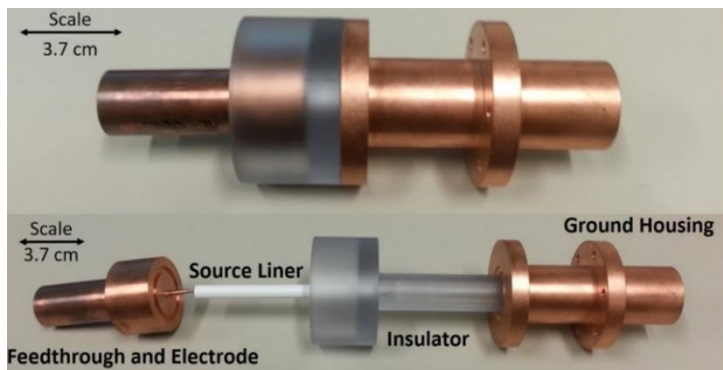


Figure 3.1: Photograph of the source (top) and the components (bottom). (Gebhart 2016)

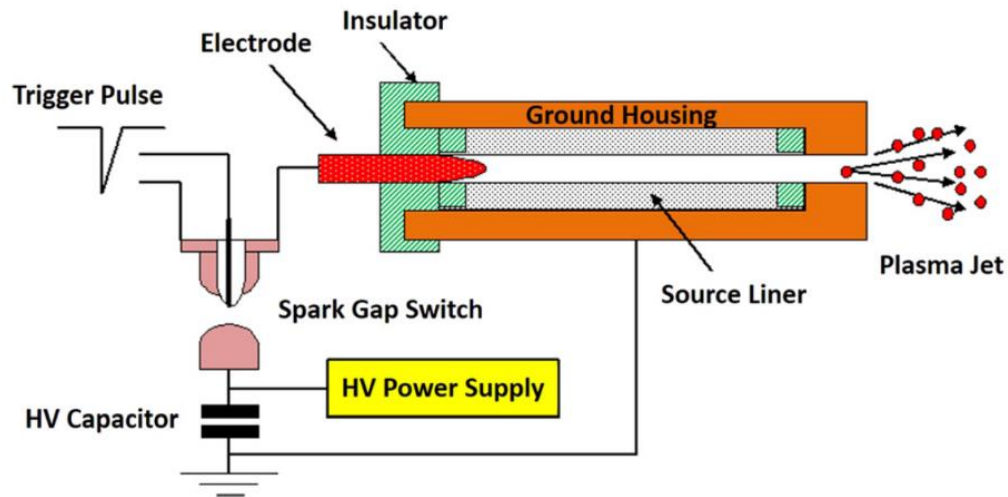


Figure 3.2: A schematic of the electrothermal-arc plasma source. (Gebhart 2016)

circuit. By changing the resistance and inductance, the pulse length can be controlled from 0.1-1,000 μs . Hence, a range of heat fluxes are produced by controlling (1) the discharged energy from the capacitor, and (2) the pulse length. The capacitor discharges through a 4 mm diameter, 100 mm long capillary in the source (Fig 3.1) The capillary diameter is only a few millimeters, so when a huge amount of energy (up to 17 kJ) is in a small area, the source liner ablates. The capillary liner is typically made of polycarbonate materials, like Lexan, but makes very dirty plasmas. In this application, a boron nitride (BN) ceramic is used. BN was chosen because in fusion plasmas B, N, and BN have low collisional- and radiative-losses. The Joule heating from electric current passing through or by the liner rapidly heats the material and causes dissociation. This creates a “dusty” plasma of primarily He, B, and N with a strong pressure gradient that pushes the plasma out of the capillary towards the target. More details on the design and operation of the arc source can be found in (Gebhart 2016, Gebhart 2017, Gebhart 2018). The source delivers heat fluxes of 0.25-2.1 GW m^{-2} .

The plasma spot size varies but is typically $\sim 0.2 \text{ cm}^2$ (Fig 3.3). The plasma beam is Gaussian and is modelled as such. The plasma plume is highly turbulent when it impacts the target (Fig 3.3). (Coburn 2020, Gebhart 2016)

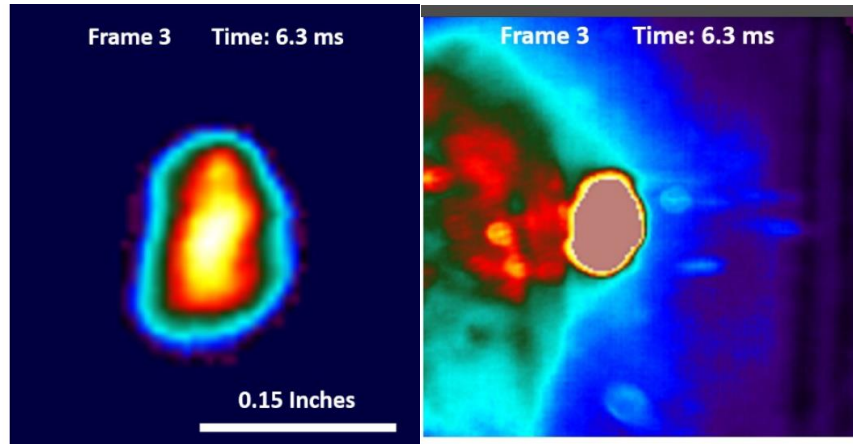


Figure 3.3: (left) An IR camera image of the plasma spot on target, temperature range 700-1500 °C and the exposed area is 0.278 cm²; (right) An IR camera image of turbulence in the plasma plume as it impacts the target, temperature range 150-400 °C (Coburn 2018)

3.2.2 Original Set Up

The ET-Arc was developed as a means to simulate transient pulses on the linear plasma device at ORNL, Proto-MPEX. (Rapp 2017, Rapp 2020) Currently, this is not viable because Proto-MPEX is slated to be shut down during the construction of MPEX. The original setup was at ORNL in building 7625. As a stand-alone device, the source can be used as a test bed for *in situ* laser diagnostic development. The entire setup consists of a vacuum vessel, high voltage power supply and a cage. The cage is for safety reasons and is electrically grounded. If the cage is opened, it immediately grounds the high voltage components.

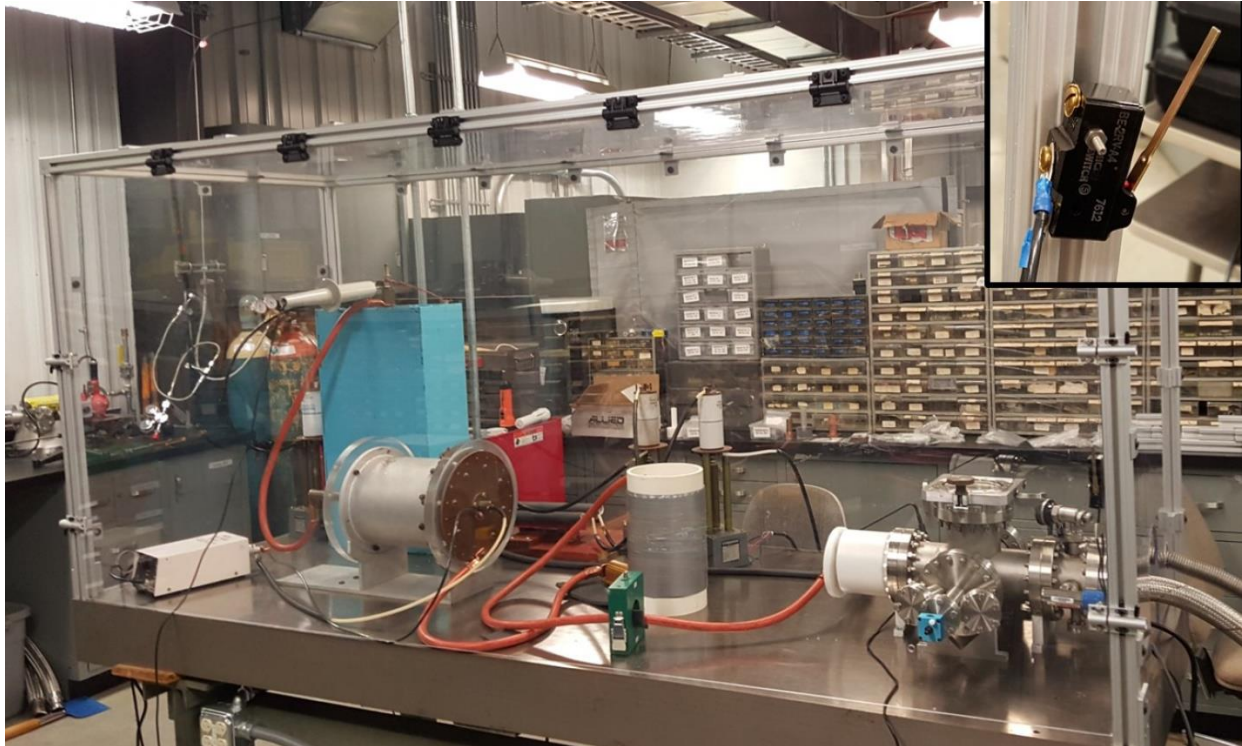


Figure 3.4: Photograph of the ET-Arc at Oak Ridge Natl. Lab building 7625. Top left is the safety switch which makes sure the cage cannot be opened with high voltage components on. (photo courtesy of T.E. Gebhart 2016)

3.2.3 Diagnostic Suite

The reconfiguration and upgrades of the ET-Arc source are mostly for increasing diagnostic access for diagnostic development and PMI studies. The source had an Ocean Optics LIBS 2500+ spectrometer for optical emission spectroscopy to obtain plasma density and temperature measurements, a FLIR SC4000 infrared camera for heat flux measurements, two Tektronix P6015A high voltage probes to measure the capacitor voltage and the discharged voltage at the anode, and a Pearson Model 5664 current monitor to measure how much current went into the plasma. (Gebhart 2018)

3.3 UPGRADES AND MODIFICATIONS

The aspects of the ET-Arc that were modified include the: (1) vacuum chamber, (2) diagnostic suite, (3) spark gap switch, (4) target design, and (5) the triggering

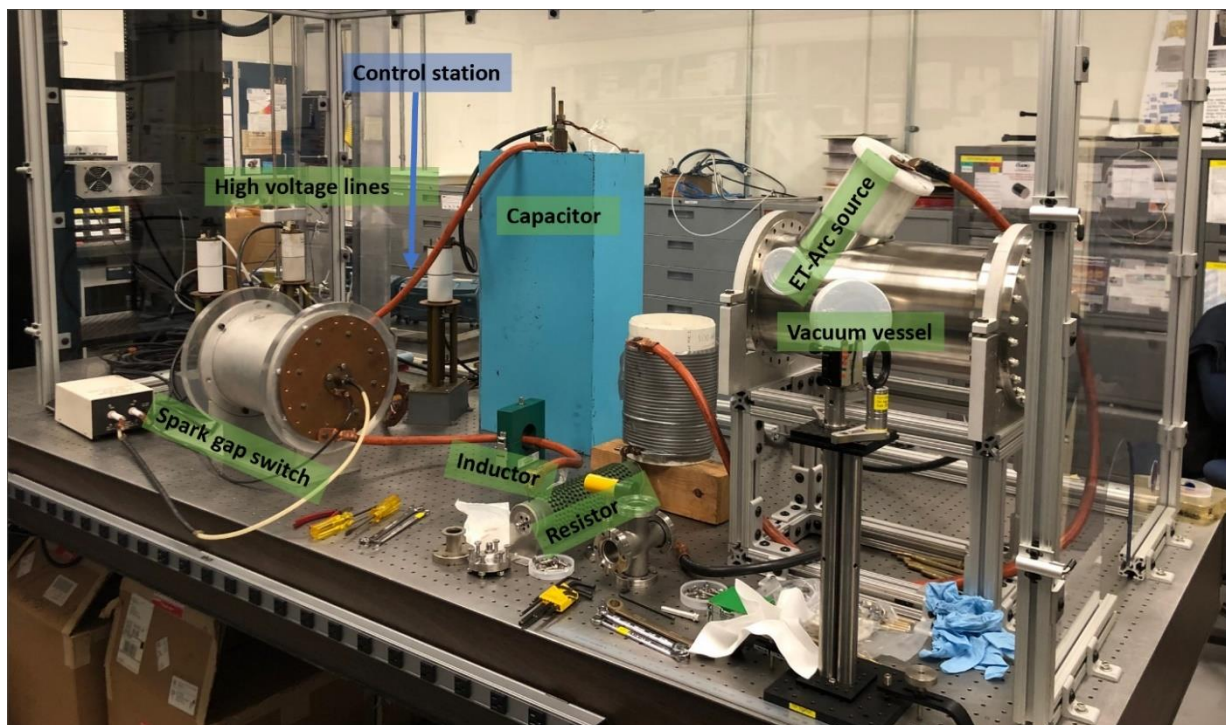


Figure 3.5: A photograph of the current ET-Arc set up at Oak Ridge Natl. Lab building 5800. (Photo courtesy of T.M. Biewer)

system. The vacuum chamber was modified to accommodate two laser diagnostics: a Thomson scattering spectroscopy system and a digital holography surface imaging technique. This change was a design constraint to provide enough room inside the vacuum vessel for the DH laser to impact the target at 90°.

3.3.1 Vacuum Vessel

The vacuum chamber was modified to accommodate two laser diagnostics: Thomson scattering (TS) with optical emission spectroscopy

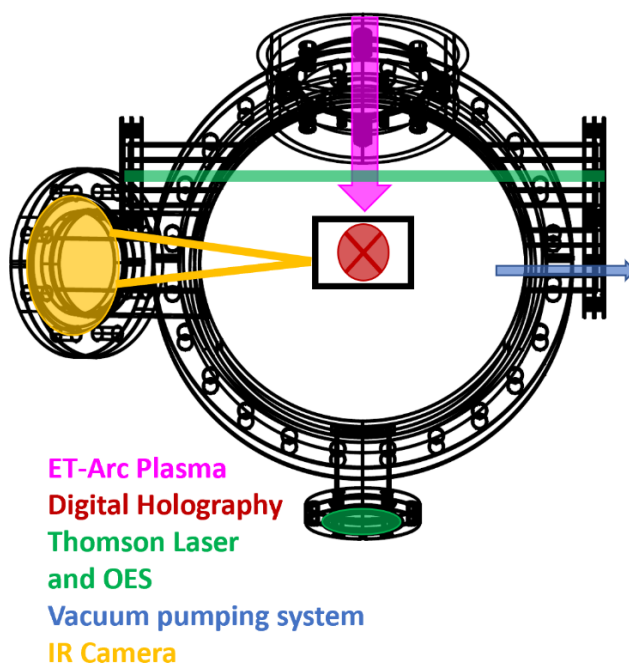


Figure 3.6: The new ET-Arc vacuum chamber with lines-of-sight for the diagnostics. See the legend for color code.

(OES), and Digital Holography (DH). The ET-Arc source is serving as the test bed for these diagnostics.

The cross section of the vacuum chamber (Fig 3.6) (not to scale) shows the layout of the diagnostic suite. The plasma enters from the top (magenta) and impacts the target (black rectangle) at a 35° angle. The DH (red) views the target at a 90° angle, with the incident beam reflecting backwards. The TS beam (green) passes above and in front of the target a few centimeters away from the target with the fiberoptics viewing the Thomson laser at a 90° angle from the bottom of the vacuum vessel. TS will be able to characterize the plasma, but not directly in front of the target. The last port is for the IR camera which views the target at a 45° angle.

3.3.2 Diagnostic Suite

The diagnostic suite contains the four existing diagnostics (Section 3.2.3), and the entire system has been modified to accommodate testing of two new laser diagnostic systems. A digital holography (DH) laser diagnostic will be implemented to measure target erosion *in situ*. (Biewer 2018, Smith 2020) A portable diagnostic package (PDP) containing both OES and Thomson Scattering will also be developed on the ET source. (Biewer 2020)

3.3.2.1 Digital Holography

Digital holography is a laser interferometry system comprised of (1) a coherent light source, (2) the detector which forms the interferogram, and (3) an imaging object. The IR CCD camera is the detector, which records the phase fringes as bright and dark bands. DH records a 3D image of the surface and will be implemented for *in situ* erosion measurements. The change in a surface is taken by subtracting frames to get the phase difference, which can be converted to height. Hence, DH can measure a change in depth for an exposure time, thereby determining

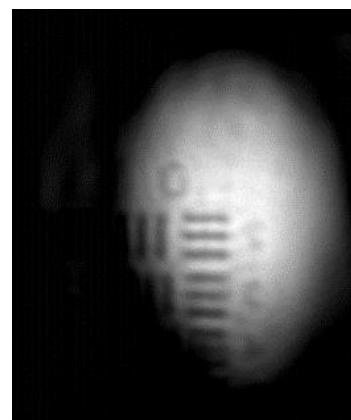


Figure 3.7: The digital holography beam spot on the ET-Arc table. Ellipse is $8.4 \times 14.2 \text{ mm}^2$ and the resolution is $89 \text{ }\mu\text{m/pixel}$. Image courtesy of C.D. Smith and C.E. Thomas.

the erosion rates. Results from *ex situ* DH characterization of stainless-steel targets exposed to the ET-Arc source indicated that surface erosion of ~ 150 nm per shot occurred and an *in-situ* DH characterization of similar targets has been planned. (Smith 2020)

The setup is changing from a stationary target on the DH table to a target inside the ET vacuum vessel. The laser beam travels ~ 1 m and views an 8.4 by 14.2 mm² ellipse at 89 μ m per pixel resolution (Fig 3.7). The new setup of the DH and ET tables is shown in (Fig 3.8). More details on the DH-ET system and measurements will be published later. (Smith 2020)

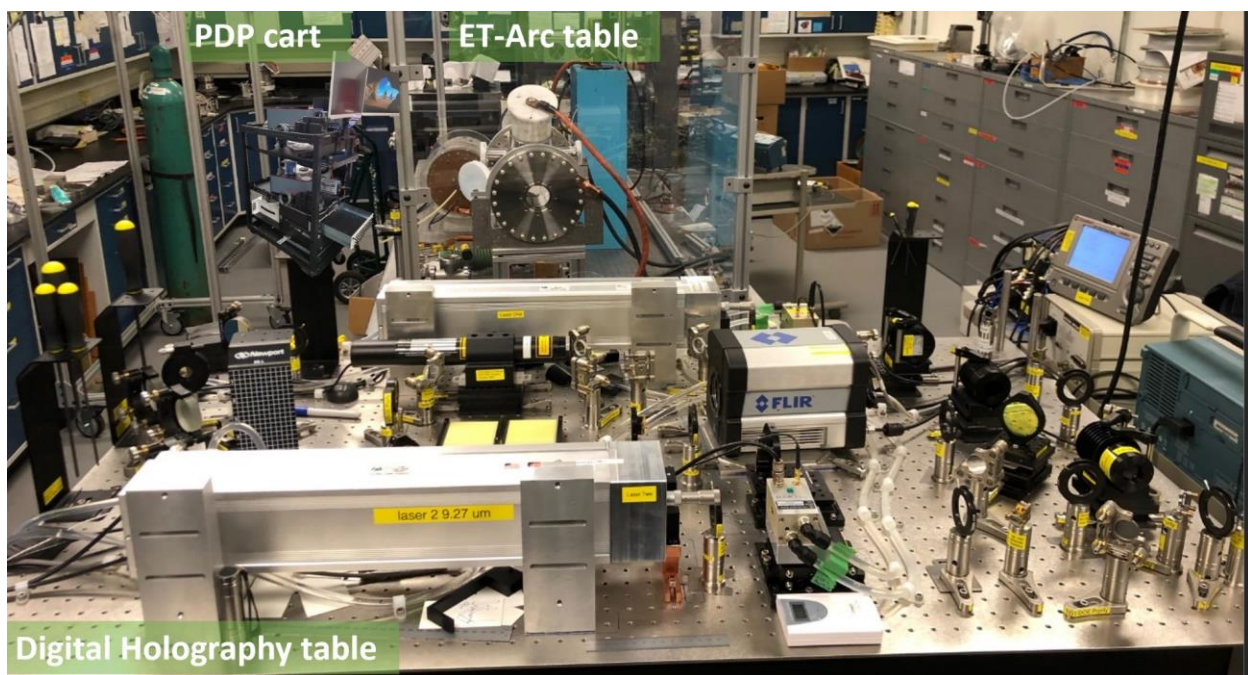


Figure 3.8: The diagnostic lab D115 in building 5800 at Oak Ridge Natl. Lab, showing the digital holography table and the PDP spectroscopy cart in relation to the ET-Arc plasma source. (photo courtesy of T.M. Biewer)

3.3.2.2 ARPA-E Portable Diagnostic Package (PDP)

The portable diagnostic package (PDP) is designed to be a spectroscopy suite that can be implemented easily on any fusion device (Fig 3.9). (Biewer 2020) The PDP project is funded by the Advanced Research Projects Agency – Energy (ARPA-E). that will conduct measurements at various laboratories across the U.S. Proof-of-principle measurements will be conducted at ORNL before the PDP is deployed. It includes

Thomson scattering and optical emission spectroscopy. To date, TS is the “gold standard” for plasma T_e and n_e measurements. TS is an active measurement where the laser photons scatter off the plasma electrons and produce a shift in wavelength. From this shift, T_e and n_e can be determined. The OES system measures the characteristic emission spectra of plasma ions to infer T_i , n_i , and v_i due to Doppler broadening of the light. Hence, both electron and ion parameters can be well characterized (Table 3.1). Assembly will be conducted at ORNL and proof-of-principle measurements will be on the ET source.

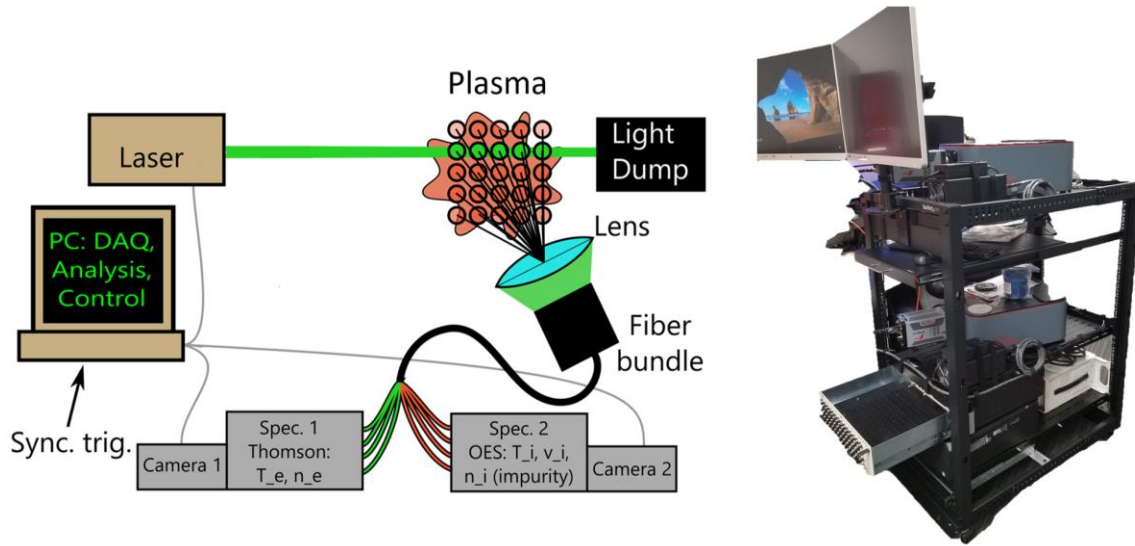


Figure 3.9: (left) A diagram of the portable diagnostic package with the Nd:YAG Thomson scattering laser (green) interacting with the electrons in the plasma (red), the fiberoptics (blue) signal is used for both Thomson and optical emission spectroscopy.(right) depiction of the PDP cart (Biewer 2020)

Table 3.1. The Thomson scattering and optical emission spectroscopy measurement ranges and resolutions for the portable diagnostic package. (Biewer 2020)

Parameter	Diagnostic	Range and Resolution	Spatial / Temporal Resolutions	Spatial Points
T_e	TS	5-1000 eV	~10 ns	10
n_e	TS	$>10^{19} \text{ m}^{-3}$	~10 ns	10
T_i	OES	5-1000 eV	> 2 ns	10
n_i	OES	Impurity	> 2 ns	10
v_i	OES	$>10 \text{ km s}^{-1}$	> 2 ns	10

3.3.3 Target Design

The material targets were redesigned to optimize the digital holography measurements. There are six targets made of 304 stainless steel. Four have a polished mirror finish, and the remaining two are polished plus wet-bead blasted to slightly increase surface roughness. The target is designed with a step to create a fiducial shadowed region in the plasma-exposed target region to aid in *in situ* erosion measurements. This is necessary for DH to have an un-eroded surface for calibrating the 3D depth scale. However, the step is detachable so the plate can be flat for *ex situ* analysis. The screws make a 2" square and use 1/4-20" bolts to be compatible with optical mounts. In Fig 3.11 (left), the ET-Arc source is yellow, the target plate is green, and the target step is red. The rectangular, cyan step is 0.25" thick and, based on a 35° plasma incidence angle, it produces a shadow of ~0.357". Erosion on targets from past experiments shows ~30% of this area is not truly shadowed, probably due to plasma flow turbulence. In Fig 3.11 (right), the vertical line to the left of the step is where the corrected shadow should be, the red ellipse is the 8.4 by 14.2 mm² DH laser beam, and the horizontal line is where the center of the plasma jet should strike.

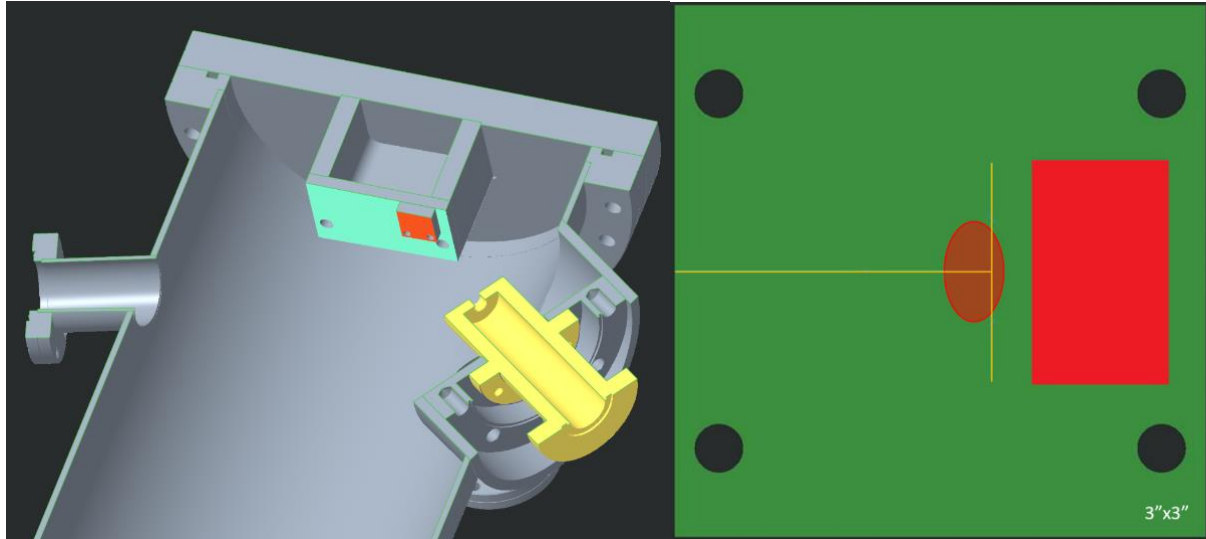


Figure 3.11: (left) Cross section of the vacuum chamber with the ET source (yellow), the target (green), and the target step (red). (right) Diagram of the target plate showing the DH laser spot (red), the shadow from the step (vertical line), and the center of the plasma (horizontal line).

3.3.4 Triggering System

With the new set up, the ET-Arc source, DH laser system, and the PDP spectroscopy system all need to be synced. A high degree of accuracy is necessary due to the short duration of the plasma pulse. A lot of effort was put into avoiding ground loops. All the DH components, e.g. the DH laser, operating computer, etc, are electrically isolated from the ET components. On the ET table, the high voltage components are in direct contact with the table which is grounded to the building ground. Hence, everything on the table share a ground that is connected to the building ground. There is a power strip on the side of the table that shares a ground with the table, so there is a ground loop between the table and the strip. Components that do not have their own grounds will be affecting by the ground loops.

The arc-triggering system was updated to synchronize with the added laser diagnostics (Fig 3.12). The ET-Arc has its own trigger generator that sends a TTL trigger to the two oscilloscopes and the HV trigger for the spark gap. The DH, TS, and ET-Arc systems are linked by a LabJack pulse generator and the DH pulse generator. The DH pulse generator has four settings, one of which is the “master pulse” on channel D. The master pulse triggers the LabJack and to the BNC 500B. The LabJack triggers

the acquisition system on the DH computer and the BNC 500B triggers the ET pulse generator. An additional pre-trigger from the BNC 500B can be sent to the PDP.

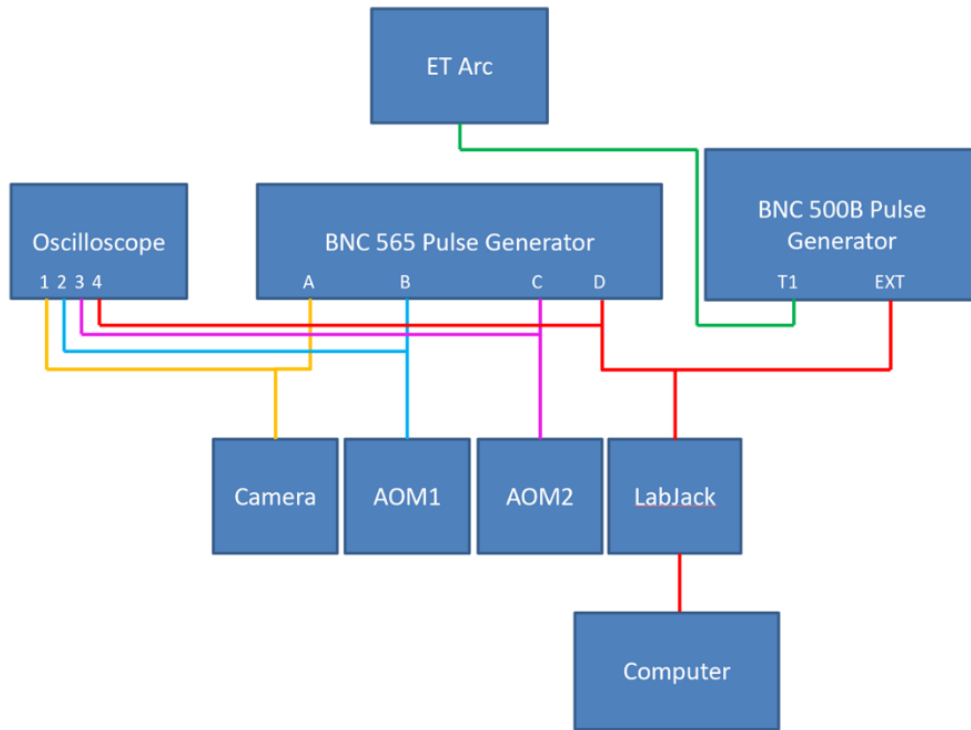


Figure 3.12: A diagram of the triggering system to link the DH and ET systems. (Diagram courtesy of C.D. Smith)

3.4 FUTURE WORK

The original goal was for the project to rebuild and operate the ET plasma source while the DH and PDP teams implemented the diagnostics on the source and performed experiments. Due to the COVID-19 suspend work, the “intended work” plan has become the “future work” plan (Table 3.2).

Most of the arc components are set up. The vacuum chamber, high voltage components, and safety cage are completed. Before testing, it is necessary to install ZnSe vacuum windows, cut laser entry holes into the cage plexiglass, mount the target plate, enclose the Thomson laser path, connect the vacuum pump, and install the high voltage probes. Then, the ET-Arc can be tested and conditioned.

The ET-Arc will be used for *in situ* measurements with the DH and the PDP. Timelines are uncertain due to research delays from the pause in operations due to

COVID-19 personnel access restrictions, but the goal is for digital holography and the PDP spectroscopy suite to be installed on the ET-Arc over the fall. Experiments that will be conducted include: (1) vibration study, (2) single-laser *in situ* DH erosion measurements, (3) initial PDP measurements, and (4) dual-laser *in situ* DH measurements.

The goal of the vibration study is to quantify how vibration and solid body motion will affect DH measurements. Using an accelerometer, measurements can be made before, during, and after a plasma pulse. If the vibration is too much, the stand and components can be modified to reduce vibration. This is critical to maintaining good resolution for deployment on other plasma devices.

The *in situ* erosion measurements are the main deliverable of the DH diagnostic. Before plasma exposure, profilometry and SEM will be performed for pre-exposure comparison. Measurements are made before, during, and after the plasma pulse. Part of the laser spot will measure a shadowed region, allowing for direct depth comparison. The arc source may need several plasma pulses before erosion is sufficiently pronounced to be measurable.

The PDP is a spectroscopy suite with well-understood measurement and analysis techniques. Once implemented, aligned, and calibrated, the main deliverable for the PDP is proof-of-principle measurements of the plasma parameters n_e , T_e , n_i , T_i , and v_i using this portable system.

Table 3.2: Summary of tasks for this project, including what was done and what remains to be done due to COVID.

Task	Status	Team
Move and rebuild the Arc source from 7625 108 to 5800 D115	Done	ET
Modify the vacuum chamber to accommodate more diagnostics	Done	ET
Design and obtain new target plates with fiducial step	Done	ET, DH
Set up system to accommodate the DH & PDP	Done	ET
Sync triggering systems between plasma source, digital holography, and the ARPA-E spectroscopy suite.	In progress	ET, DH
ANS TOFE FST paper	In progress	
Upgrade spark gap switch	In progress	ET
Test and troubleshoot system	To be done	ET
Operate ET source	To be done	ET
PDP alignment and implementation	To be done	PDP
PDP proof-of-principle measurements	To be done	PDP
DH alignment and implementation	To be done	DH
DH first <i>in situ</i> erosion measurements	To be done	DH
Assess vibration and solid body motion on ET Arc	To be done	DH
Assess surface finish for non-magnetic 302 or 304 stainless steel	To be done	DH
Assess surface finish assessment for W and SiC	To be done	DH
Set up dual-laser DH system for <i>in-situ</i> measurements	To be done	DH
Assess thermal growth and contraction in target materials using DH and IR camera	Done	DH
Ex-situ measurements of ET Arc erosion on acceptable targets	Done	DH

CHAPTER FOUR

SUMMARY AND CONCLUSIONS

Fusion reactors produce an incredibly harsh environment where plasma-facing materials need to withstand cyclic high thermal loads, plasma exposure, and neutron irradiation. Some of the most damaging plasma-material interactions are high heat flux, high particle flux, thermal fatigue, neutron irradiation, as well as hydrogen and helium implantation.

Naturally, single- and multi-variable testing facilities are necessary for each of these damage mechanisms. High heat flux (HHF) and high particle flux (HPF) facilities are critical for developing and qualifying advanced fusion materials. Current tokamaks are not able to produce ITER- or reactor-relevant HHF and HPF. Additionally, most PMI studies rely on *in vacuo*, *post mortem*, or *ex situ* analyses. Hence, fusion testing facilities are needed for *in situ* diagnostic development at reactor-relevant HHF and HPF.

While there are many types of HHF facilities, the main ones discussed in this report are linear plasma devices, electron beam devices, and electrothermal arc (ET-Arc) plasma sources. This project, chapter three, specifically concentrated on diagnostic development on the ET-Arc source at ORNL for *in situ* PMI studies from transient plasma events. The ET-Arc source produces plasma heat flux of $\sim 0.1\text{-}2\text{ GW m}^{-2}$ and high particles flux due to densities of $n_e \sim 10^{22}\text{-}10^{28}\text{ particles m}^{-3}$ which is similar to ELM transient events. (Gebhart 2018) Recently, the ET-Arc was modified for diagnostic development. Modifications include the vacuum chamber, target plate design, diagnostic suite, and triggering system.

The goal of the project was to re-assemble and to operate the ET plasma source while the DH and PDP teams implemented the diagnostics on the source and performed experiments. Due to the COVID-19 suspension of work, most of the “intended work” plan has become the “future work” plan.

Once operational, the ET-Arc will be used for *in situ* measurements with the DH and the PDP. DH and PDP will be installed on the ET-Arc in the Fall. Vibration and *in situ* erosion experiments will be conducted. Then, the setup will be complete for future PMI experiments.

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