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Plasma-Materials Interactions (PMI) and High-Heat-Flux (HHF) Component Research and Development in the U.S. Fusion Program*

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1. Introduction

Plasma particle and high heat fluxes to in-vessel components such as divertors, limiters, RF launchers, halo plasma scrapers, direct converters, and wall armor, and to the vacuum chamber itself, represent central technical issues for fusion experiments and reactors. This is well recognized and accepted. It is also well recognized that the conditions at the plasma boundary can directly influence core plasma confinement. This has been seen most dramatically, on the positive side, in the discovery of the H-mode using divertors in tokamaks. It is also reflected in the attention devoted worldwide to the problems of impurity control. Nowadays, impurities are controlled by wall conditioning, special discharge cleaning techniques, special coatings such as carbonization, the use of low-Z materials for limiters and armor, a careful tailoring of heat loads, and in some machines, through the use of divertors. All programs, all experiments, and all designers are now keenly aware that PMI and HHF issues are key to the successful performance of their machines. In this brief report we present general issues in Section 2, critical issues in Section 3, existing U.S. PMI/HHF experiments and facilities in Section 4, U.S. International Cooperative PMI/HHF activities in Section 5, and conclude with a discussion on major tasks in PMI/HHF in Section 6.

2. General Issues

The generic areas where plasma-materials interactions and high-heat-flux are important include:

1. Impurity Control
2. Plasma Exhaust and Recycling Control
3. High Heat Flux Removal and Thermomechanical Response
4. Tritium Permeation and Retention
5. Wall Material Conditioning
6. Irradiation Effects

In tokamaks, an additional key area relates to plasma disruptions. Many of these areas have two aspects to them, one relating to effects on plasma performance and the other dealing with engineering design and component performance. For example, one aspect of impurity control relates to the problems of impurity production, impurity transport, radiation properties of atoms, etc.; the second aspect is related to material choices, design, and thermomechanical behavior. Data on sputtering yields, the problem of erosion and redeposition of material, or the loss of material during disruptions, all relate to impurity production. The development of low Z coatings, low Z materials, tile attachment mechanisms, and the performance of specially designed PMI/HHF components are examples of problems relating to design and thermomechanical behavior.

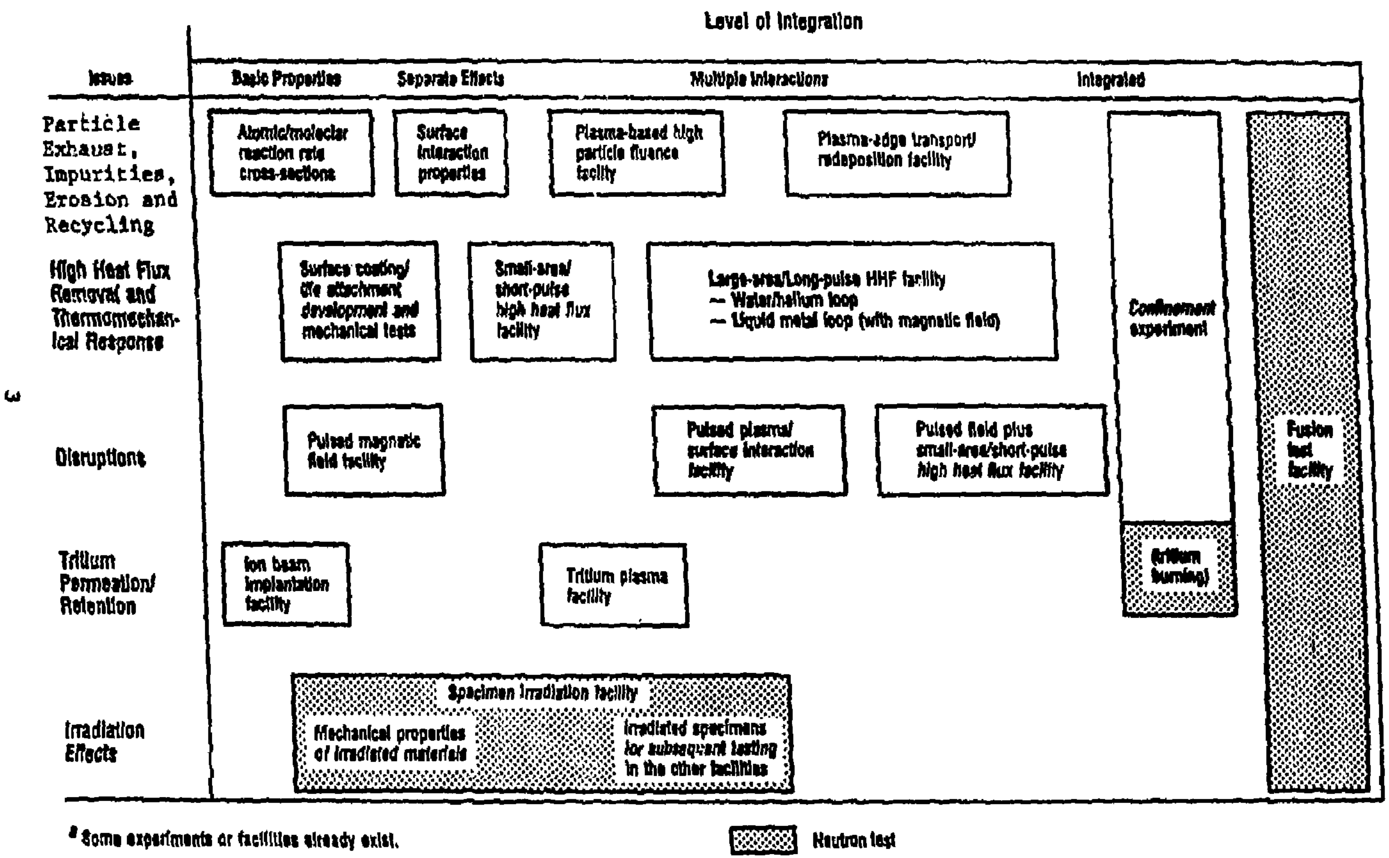
To illustrate the general types of experiments and facilities required in the PMI/HHF area, we have listed on Fig. 1 five key issues:

1. Particle exhaust, impurities, erosion, and recycling
2. High heat flux removal and thermomechanical response
3. Disruptions
4. Tritium permeation and retention
5. Irradiation effects

Shown also on Fig. 1 are experiments and facilities for PMI/HHF. These include facilities to provide basic property information, facilities to carry out specific individual tests, facilities capable of investigating multiple effects simultaneously, and integrated facilities capable of examining PMI/HHF problems in all aspects.

One important point regarding PMI/HHF is that basic problems can be addressed and materials and components can be developed before a fusion neutron test facility is operating. Thus, PMI/HHF components can be developed and tested in non-neutron facilities, including both laboratory and confinement devices, in order to provide the components for a first neutron-producing fusion engineering or test reactor. On the other hand, the separate effects and partially integrated plasma facilities for PMI/HHF should lead to a steady-state or high duty-cycle hydrogen confinement experiment to insure success in this field.

Figure 1. Types of Experiments and Facilities for Plasma Materials Interaction and High Heat Flux[a]



3. Critical Issues: Plasma Materials Interactions and High Heat Flux Components

Issues important to plasma materials interactions and high heat flux problems involve a close interaction of physics and technology. Summarized here are the key technical issues that are important in a 5-15 year timeframe, using as a reference the needs of both a short-pulse ignition experiment and a steady-state, or very long-pulse facility, similar to NET and FER. Specifics on these critical issues are given in a supplement which will be sent separately. Note that while the issues listed here are the critical ones, the specific details and significance of the issues can be quite different for short-pulse versus steady-state machines.

1. Particle exhaust

The design of particle exhaust systems is specific to the type of fusion device. Because closed systems were the first to encounter the need for active impurity control, most work has been done for tokamaks. A key issue for a long pulse high power density device is the choice between a divertor and a pump limiter. Particle exhaust may be required even in shorter pulse devices if pellet fueling is used and density control is required.

2. Recycling

Recycling occurs at the surface of a limiter or wall, or in the throat of a divertor or pump limiter. It is the process by which plasma ions leave the discharge, impinge on a material surface, become neutralized, and re-enter the plasma. Recycling influences density control, fueling, and plasma edge properties. Important questions are:

a. What are the recycling coefficients of materials of interest?
b. How does the recycling coefficient scale with surface conditions, plasma edge conditions, and integrated particle flux?
c. How is the recycling coefficient changed by radiation damage?

3. Surface Conditioning and Cleaning

Surface conditioning is necessary to minimize impurity influx from thermal and particle-induced outgassing and to stabilize the hydrogen recycling properties of the surface. It involves pretreatment (cleaning and vacuum baking) and in-situ conditioning (interaction with hydrogen plasma). The effectiveness of conditioning and cleaning for a given material can be a key factor in its utility. This consideration is a factor, for example, in determining whether beryllium or carbon should be the limiter material for a burning core experiment of for a NET/FER.

4. Erosion and Redeposition

Energetic plasma edge particles or charge exchange neutrals will strike the surfaces of plasma-interactive components (PICs) and the vessel wall, causing erosion by physical and chemical sputtering. These processes are a major source of impurities in the plasma and are also lifetime limiting for components. The problem is addressed by passive (limiter materials selection) and active (gettering, use of magnetic divertor) impurity control methods.

Sputtered particles can undergo recycling and be redeposited on material surfaces. The redeposited material may have properties different from those of the base material. For short pulse, low duty cycle machines, these processes are not important factors; they can be crucial, however, for long pulse and steady state devices.

5. Thermal Hydraulics

Typical heat fluxes at limiters and divertors are 5 MW/m^2, and direct convertor or halo scraper designs for mirrors are expected to have similar parameters. Reversed field pinches and other alternate concepts may operate as high-power-density devices and require removal of high heat fluxes over large fractions of the first wall.

Space concerns in the area of thermal hydraulics are heat transfer limits (normal and augmented), flow distribution and stability, channel erosion, and heat source profile. While considerable experience exists in these areas from other technologies, the data usually apply to parameter ranges and geometries that are not directly applicable to fusion designs. Analysis and testing continue to be necessary to qualify in-vessel components.

6. Thermomechanical Effects

Large thermal stresses result in PICs from high surface heat fluxes and large temperature gradients. Cycling leads to thermal fatigue, causing damage in two states: crack initiation and crack growth.

Armor tiles are bonded to heat sink structures that are either inertially or actively cooled. The long-term structural integrity of the bond is a critical issue since debonding will cause rapid overheating of the armor material and eventual failure of the system.

7. Disruptions

Disruptions, which are observed in all current tokamaks, are characterized by a rapid reduction in the plasma current accompanied by the localized deposition of much of the plasma energy on an interior surface. There are uncertainties in the magnitude of the induced forces (which determine to a large extent the required structural support) and of the heat fluxes (which can cause surface vaporization and melt layer formation). Ultimately, methods for controlling and eliminating disruptions will need to be developed.

8. Tritium Inventory and Permeation

The extent of tritium permeation through and retention in PICs and the wall is a significant uncertainty in assessing the safety and tritium handling requirements of D-T burning machines. Although a considerable data base has been generated for hydrogen isotope interaction with structural first wall materials, little is known about the interaction of low-Z limiter materials, such as graphite and beryllium, with tritium. These data are critically required for a burning core experiment and beyond.

4. Existing U.S. PMI/HHF Experiments and Facilities

Experiments on plasma-materials interactions, the plasma boundary layer, high heat flux, and surface materials development and performance are carried out in a variety of facilities. In addition, there are important programs carried out cooperatively between the U.S., Europe, and Japan on facilities outside the U.S. Clearly, these programs are important to the overall U.S. effort in PMI/HHF. A summary of existing U.S. facilities and cooperative U.S. programs is listed in Table 1. The efforts are divided into two categories: experiments or facilities that do not involve or require a confinement machine (such as a tokamak), and those which are carried out in a confinement device. In the past, the U.S. has had one tokamak, the Impurity Studies Experiment (ISX-A and B), which focussed on PMI issues. Today, PMI experiments are carried out in existing U.S. devices, but are not the primary activity of those machines.

Table 1

Dedicated U.S. PMI/HHF Experimental Facilities

I. Plasma Materials Test Facility (PMTF) - Sandia - Albuquerque

Features	Characteristics
A. Electron Beam Test System	30 keV, 30 kW
Target Area	1-100 cm^2
Pulse Duration	0.05 sec - continuous
Cooling	Closed Loop Water
Max. Heat Flux	Up to 30 kW/cm^2
B. Multiple Beam Test System	
Steady-State Heat Source	40 keV, 1.6MW H-Beam
Target Area	800 cm^2
Pulse Duration	Continuous
Cooling	Closed Loop Water
Max. Heat Flux	2 kW/cm^2

Table 1 (Continued)

II. Plasma-Surface Interactions Facility (PISCES) - UCLA

Features	Characteristics
Plasma	$n \sim 5 \times 10^{11} - 2 \times 10^{13}\ cm^{-3}$; $T_e \sim 3 - 30 eV$
Target Area	$100 - 400\ cm^2$
Pulse Duration	Continuous Operation
Ion Energy	50-500 eV (bias)
Cooling	Air or Water
Max. Heat Flux	$200\ W/cm^2$
Plasma Flux	$10^{17} - 2 \times 10^{19}/cm^2\text{-}s$

III. Tritium Plasma Experiment (TPX) - Sandia - Livermore

Features	Characteristics
RF glow Discharge Tritium Plasma (200W RF)	$n \lesssim 10^{11}\ cm^{-3}$, $T_e \sim 7 eV$
Pulse Duration	Continuous
Ion Energy	15-300 eV (bias)
Plasma Flux	$10^{17}/cm^2\text{-}s$

Referring to Table 1, there are three primary facilities whose focus is PMI/HHF problems. These are the Plasma Materials Test Facility (PMTF) at Sandia Laboratory, Albuquerque, the PISCES plasma-surface interactions experimental facility at the University of California at Los Angeles (UCLA), and the Tritium Plasma Experiment (TPX) Facility at Sandia Laboratories, Livermore. The PMTF consists of two facilities, an electron beam test system for materials testing, thermal disruption simulation and steady-state heat removal development, and a multiple beam test system which utilizes ion beams and is a dedicated materials and high heat flux test system.

PISCES is a continuously operating plasma facility using a plasma generator and an axial magnetic field to produce plasma that is characteristic of the edge region in tokamaks and other confinement devices. One device exists, PISCES-A, and a second facility, PISCES-B, is under construction. Both devices are dedicated to surface materials and coating behavior under continuous high flux plasma bombardment. Redeposition phenomena have been observed and are under investigation with a variety of candidate materials. In addition, plasma experimental simulations of pump limiter and divertor behavior is an objective of the program.

TPX is a low density discharge facility dedicated to surface materials investigations with tritium. Experiments relating to tritium recycling, permeability, and tritium inventory are a major focus of work. Both PMTF and TPX have investigated beryllium in a cooperative program with JET. The PMTF, PISCES, and TPX are bring used to test graphites for use as target plates in ASDEX-UG.

The U.S. PMI program also has a significant capability to carry out basic experiments to obtain basic-property data. In particular, tandem accelerator facilities exist at Sandia Labs in both Albuquerque and Livermore, and there exists an extensive capability for surface material diagnostics.

Beyond these capabilities, there exist several facilities which could be used for PMI/HHF research, development, and testing but which are now dedicated to other areas of physics or technology development. Three of these are the RF Test Facility (RFTF) at Oak Ridge National Laboratory, the

Continuous Current Tokamak (CCT) at UCLA; and the FELIX facility at Argonne National Laboratory (ANL). Characteristics for each facility are listed on Table 2.

The RFTF is constructed and is to begin operation in the immediate future. It will produce a mirror confined plasma using 300kW of ICRF power. Also available is 200kW of ECH at 28GHz. Long range plans called for an additional 1.5MW of ICRF. The RFTF primary mission is the development of RF launching and transmission systems, but it could be used for PMI/HHF testing of large scale components.

The CCT is a new tokamak operating at UCLA with the physical size of PLT, but operating at low field (1-7kG). The primary objective of CCT is ICRF heating and current drive with the intermediate goal of achieving steady-state operation at 10kA (pulsed up to 100kA for 0.3 s and return to 10kA). The plasma edge density in such a device is expected to be around $5 \times 10^{11} cm^{-3}$ with $T_i = T_e = 10-30eV$ in the edge. Plans call for investigation of phenomena such as erosion and redeposition to compare with results from continuously operating simulation experiments such as PISCES.

Finally, the FELIX facility at the Argonne National Laboratory is designed to carry out electromagnetic tests on components. It can be used in conjunction with liquid-metal blanket MHD experiments and to simulate the electromagnetic characteristics of current disruption in plasma-interactive components such as limiter or divertor plates.

Table 2

U.S. Facilities with PMI/HHF Capabilities Now Dedicated to Other Missions

I. RF Test Facility (RFTF)-ORNL

Features	Capabilities
Plasma	$n \leq 5 \times 10^{12}\ cm^{-3}$; T_e, $T_i \simeq 30 - 100eV$ (Mirror Confined)
Target	$> 1000\ cm^2$
Pulse Duration	Steady-State with 300 kW ICRF
Cooling	Closed Loop Water
Ave. Heat Fluxes	$100-300\ W/cm^2$
Primary Mission	RF Component Development and Testing

II. Continuous Current Tokamak (CCT)-UCLA

Features	Characteristics
Plasma	$n \sim 5 \times 10^{12}\ cm^{-3}$; $T_i = T_e \simeq 10-100eV$ (Tokamak Confined)
Target Area	Test Components $> 1000\ cm^2$
Pulse Duration	.3 sec. @ I_p=100kA Continuous @ I_p=10kA
Cooling	Water

Table 2 (Continued)

Ave. Heat Flux	10–100 W/cm^2
Primary Mission	ICRF/Fast Wave Current Drive, Coupling and Heating in Tokamaks

III. FELIX - ANL

Features	Capabilities
Poloidal, Toroidal	Test induced currents and forces created in In-Vessel Components by transient magnetic fields
Field Simulation Facility	

Facility Specifications

Solenoid Field Strength	1.0 T (4.0 T)[a]
Dipole Field Strength	0.5 T (1.0 T)[a]
Dipole Ramp Time	5 ms (3 ms)
Test Volume	0.75 m^3

Diagnostics
- Hall Probes
- Force Sensors
- Displacement/Rotation Sensors
- Rogowski Coils
- IR Thermography

Data Acquisition
- 30 channels, 2048 pts./channel
- Data can be transferred to MFE NET.

[a] Indicates upgrade field levels.

5. Existing U.S.-International Cooperative PMI/HHF Activities

Several programs important to U.S. PMI/HHF and plasma boundary physics activities are carried out as part of international collaborations. The largest of these are summarized in Table 3. Some of these have been completed recently. With Europe, the U.S. is carrying out the ALT-I pump limiter program involving UCLA and Sandia at the TEXTOR tokamak, IPP/KFA-Juelich. A program on pump limiters and fueling for TORE-SUPRA in France is led by ORNL and Sandia. Experiments are to begin in late 1987 or 1988. Beryllium limiter tests have been completed for the JET project at the ISX-B tokamak at Oak Ridge and in PMTF at Sandia. Testing of divertor target plate materials is on-going at PMTF and at PISCES, UCLA for the ASDEX-UG at MPI-Garching. Recently, the ALT-II pump limiter project on TEXTOR between the KFA-Juelich, the U.S. (UCLA, Sandia, and ORNL), and Japan (IPP-Nagoya) has been approved by all parties. A toroidal belt pump limiter is under construction for installation on TEXTOR in early 1987. Japan-U.S. and European-U.S. workshops have been held on edge physics, pump limiters, and divertors, and future workshops are planned. International cooperation in plasma edge physics, PMI and HHF, is a central ingredient of U.S. efforts in these areas.

6. Major Tasks in PMI/HHF

A list of major tasks is given in Table 4 for the periods 1985-90, 1990-95, 1995-2000, and 2000-05 in four general categories: particle control; impurity control; high heat flux; and transient or off-normal events. This chart is based on the assumption that in the next 5-15 years, there will be a short-pulse compact ignition experiment and a long pulse or steady-state facility similar to NET or FER. A test plan and sequence of evaluation and testing milestones are given in Fig. 4. Specific evaluation/design points, designated E, and testing periods, designated M, are shown. Confinement experiments will either utilize the results of tests or be the test vehicle themselves. That is, confinement experiments are key test facilities in the PMI/HHF area, along with the simulation and test facilities existing or planned. Assumed in this plan is the existence and use of upgrades or planned confinement machines, including TEXTOR, TORE SUPRA, ASDEX Upgrade, and ATF Upgrade, and the completion or upgrading of PMTF and PISCES facilities.

Table 3

U.S.-International Cooperative PMI/HHF Activities

Program	Participants: U.S.	Participants: Europe	Participants: Japan	Features/Objectives
ALT-I Pump Limiter in TEXTOR	Sandia & UCLA	IPP/KFA-Juelich		Module Pump Limiter with Several Head Designs and Several Materials (graphites, TiC, Inconel) Status: Final Year
Beryllium Limiter Experiment in ISX-B	ORNL & Sandia	JET		Test use of Beryllium Limiters in a Tokamak and in PMTF Status: Completed
Pump Limiters/Fueling in TORE SUPRA	ORNL & Sandia	F. aux Roses and Cadarache; C.E.A.		One Modular Pump Limiter with active cooling. Pellet Fueling. Status: Active; Design Phase
ALT-II Toroidal Belt Pump Limiter in TEXTOR	UCLA Sandia ORNL	KFA-Juelich	IPP-Nagoya	Complete Toroidal Belt Pump Limiter: Graphite Tiles and Radiative Cooling Status: Active; Construction
Divertor Materials Tests for ASDEX-UG	Sandia & UCLA	MPI-Garching		Test Materials (graphites) for ASDEX-UG Divertor Status: Active: Experiments in PMTF/PISCES/Diagnostic Facilities

Experiments in PMTF, PISCES, and TPX can be supplemented through the conversion of other facilities to address PMI/HHF problems. This was referred to in the section on existing U.S. facilities. Each of these facilities can carry out separate or multiple effects experiments. Ultimately, the results should be normalized to performance data from experiments in actual confinement devices. In this regard, a long pulse, high duty cycle or steady-state confinement device is needed.

For the tokamak program, a toroidal confinement facility is needed. Over the next 5 years, cooperative research on TEXTOR and the TORE SUPRA will provide data on machines with 3 to 30 second pulse lengths. However, TEXTOR and TORE SUPRA have low duty cycles. The Continuous Current Tokamak, CCT, will achieve steady-state with RF current drive and can provide important benchmark data for off-line facilities like PISCES and PMTF. The U.S. program does plan to utilize international cooperation to achieve the progress and results that will be needed in the next 5-15 years. Discussions are already underway with both Europe and Japan. Cooperation with Japan in the steady-state PMI/HHF area is being pursued.

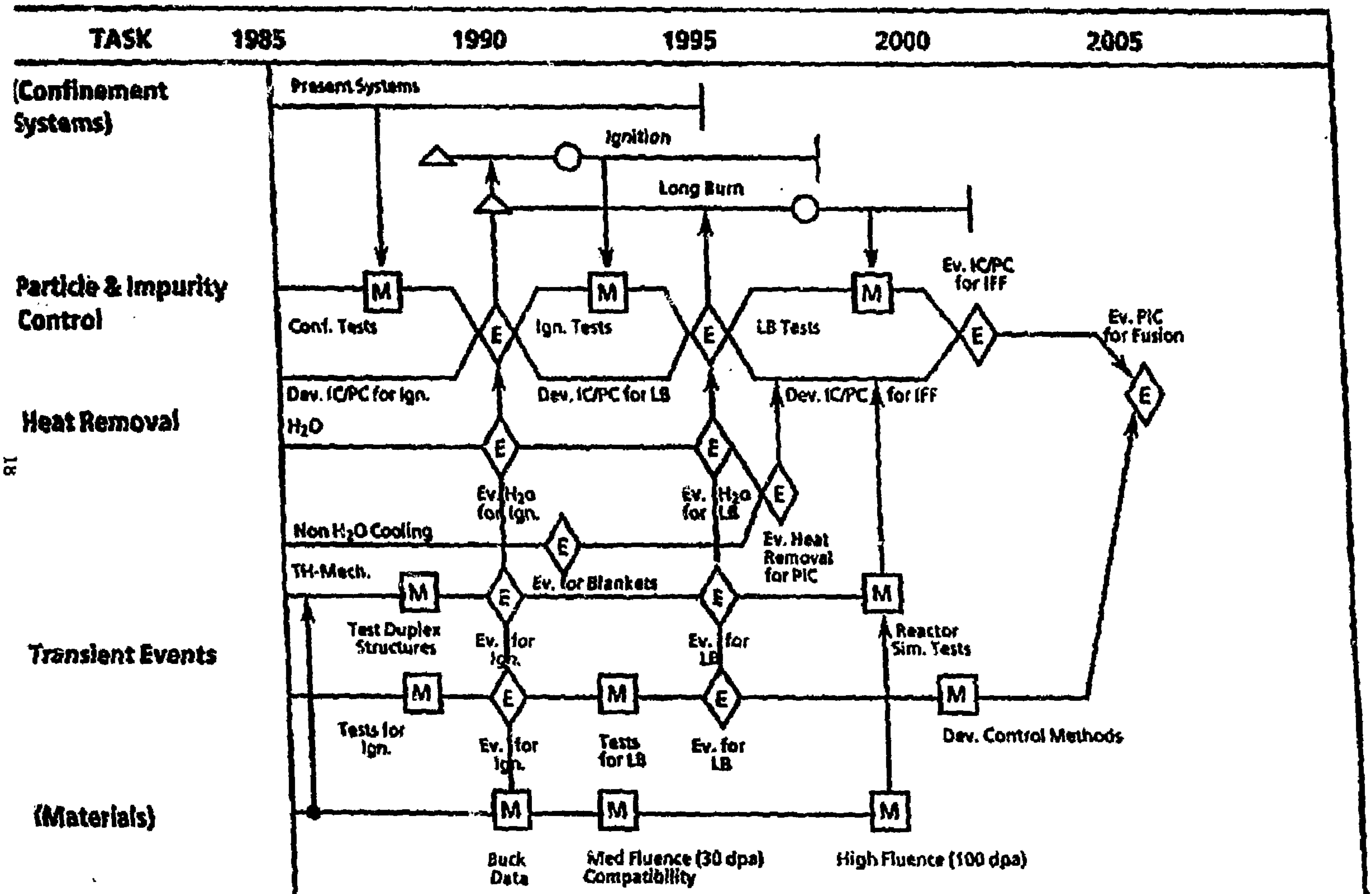
In the longer run, 5-15 years from now, a continuous, high power confinement experiment is needed. This could be achieved through the upgrade of ATF or TORE SUPRA. These toroidal facilities will continue to be supplemented by special high heat flux facilities, such as an upgrade of PMTF to its original design capability, and improved plasma facilities such as a PISCES-UG, a Mirror Device, or an RF plasma machine.

Table 4

Major Tasks in PMI/HHF Area

	1985-90	1990-95	1995-2000	2000-2005
Particle Control	Develop and verify models of plasma edge and plasma materials recycling. Perform tests in existing confinement devices.	Develop predictive capability for particle control in major high power, long pulse hydrogen confinement experiments based upon previous developed models and data from plasma simulation experiments.	Verify particle control predictive capability in long pulse high power tokamak experiments such as ETR.	Demonstrate particle control for long pulse, burning plasma conditions.
Impurity Control	Choose leading concepts and materials for impurity control for both ignition and long pulse devices. Test specific issues of impurity control in existing confinement and off-line devices.	Choose impurity control system for long pulse high power device such as ETR and demonstrate impurity control in ignition device and long pulse high power hydrogen experiments.	Verify impurity control predictive capability in long pulse high power tokamak experiments such as ETR.	Demonstrate IC for long pulse, burning plasma conditions.
High Heat Flux	Select leading structural designs and materials. Perform thermal-mechanical tests on lab. size specimens. Develop predictive capability for thermal-mechanical interactions.	Perform test to determine capability to cool with non-water coolants. Choose materials and structures for long pulse device.	Perform tests to determine HHF limits for reactor components.	Demonstrate HHF capability for leading designs of plasma side components.
Off-Normal Events	Develop predictive capability for disruptions in ignition and long pulse devices. Test materials and structural response to simulation of disruption conditions.	Develop methods for minimizing effects of disruptions and other off-normal events.		Demonstrate capability for resistance to off-normal events.

Figure 4. General test plan and timing sequence of evaluation and testing as a function of five major task areas.



Appendix

Specifics on Critical Issues

A. Impurity Control

Impurity control depends on the conditions of the plasma edge, the transport of impurities into the plasma core, and the transport and exhaust of helium ash. It is well appreciated that conditions at the plasma edge can directly influence core plasma confinement time. These complex problems depend on the technique used for particle and heat removal. For example, will limiters or divertors be the approach in tokamaks? How will direct convertors and halo scrappers be designed in mirrors? Tasks which must be accomplished are:

(1) Determine edge plasma conditions, the degree to which they can be controlled, and the impact on core confinement.
(2) Determine the required heat-bearing geometry and heat load distribution.
(3) Determine impurity transport through, i.e., into and out of the plasma edge.

Impurity introduction of low Z and high Z atoms adversely effects plasma performance and erosion can limit the lifetime of plasma interactive components. Low Z impurities, such as oxygen and carbon, are released from surfaces in today's fusion devices by sputtering of the adsorbed species (ion-induced desorption) or from electronic excitation by electron and photon bombardment. However, desorption may become less important in high duty cycle devices where the desorbed product will be pumped away and not replaced.

High Z impurities are released from plasma interactive components by physical sputtering, chemical erosion, arcing, blistering, evaporation, and disruption. Physical sputtering is considered the dominant source of high Z impurities, both in present machines as well as in future reactors. The transport of impurities in the plasma and around the machine will directly determine plasma component response.

B. Erosion and Redeposition

Energetic plasma-edge particles or charge-exchange neutrals will strike the surface of plasma-interactive components (PICs) and the wall thereby causing erosion. Eroded particles enter the scrapeoff plasma, undergo reionization, and can be redeposited on material surfaces. The redeposited material may have properties different from those of the base material. Because of the short pulses and low duty factor of current and near-term machines, erosion and redeposition is not typical of what is to be expected in future long pulse or steady-state devices. It has already been learned in recent laboratory experiments that surfaces undergoing redeposition are different in appearance and surface structure from unbombarded material. Little information exists about the properties of these surfaces. Key issues are the amount of redeposition, the net erosion rate, the sputtering yield of redeposited surfaces, the secondary electron yields (which influences sheaths and therefore the bombarding energy of ions), the binding properties of redeposited material, and alterations in thermal properties. The present data base for predicting behavior in future machines is poor.

C. Plasma Recycling

Critical issues for fueling large, dense, steady state plasmas are:

(1) Assess the importance of fueling location on particle confinement.
(2) Develop techniques for fueling the plasma core.

Hydrogen recycling effects are defined to include fuel recycling between the plasma and PIC's, as well as tritium inventory and permeation issues in D-T burning reactors. In present day devices, the lack of data and theory for the reflection coefficient below 100eV is the most critical recycling problem. In a similar vein, hydrogen desorption measurements from surfaces by ion-impact, photons, and electrons, are only in their infancy. What is most needed is a general model applicable to a machine environment.

In future D-T burning machines, trituim transport in the first wall and limiter/divertor will be a serious environmental, safety, and economic concern. Tritium inventory will be a critical problem area for D-T ignition machines, while tritium permeation will become an important concern for high duty cycle machines such as NET/FER. While a reasonable data base and theoretical understanding exists for most metals, very little is known for hydrogen interaction with low Z materials such as graphite, carbides, etc., at relevant operating temperatures.

Related questions are:

(1) What are the recycling coefficients of the materials of interest?
(2) How does the recycling coefficient scale with surface conditions, plasma edge conditions, and integrated particle flux?
(3) How is the recycling coefficient changed by radiation damage?

D. Plasma Exhaust

Plasma exhaust will be needed in reactors to remove helium and, in near term, long-pulse experiments, to help control the plasma density. Exhaust in toroidal devices can be achieved with pump limiters or divertors, and has been demonstrated with both approaches. Key issues relate to the flow of helium relative to H or D/T, techniques for improving duct conductances to reduce port size and neutron leakage, and the efficiency of particle exhaust techniques when used in conjunction with edge plasma modifications, such as with RF or an ergodic layer. The performance of materials and heat removal problems are discussed elsewhere.

E. Wall Conditioning

The question is whether long-pulse, high-power fusion machines will require new wall preparation procedures. Discharge cleaning techniques may not be adequate in hot wall, quasi-steady-state reactors. Laboratory studies of wall conditions under simulated reactor thermal and particle loads and experiments in a long-pulse, steady-state confinement device are needed to settle this issue.

Very little experience on the conditioning of high flux limiter surfaces (e.g. graphite) exists and will require future laboratory research. The need for in-situ surface conditioning may tend to diminish in the future because the high heat loads and much higher duty cycle will accelerate the conditioning process. However, erosion and redeposition will govern the nature and composition of surfaces, and this is an unknown.

F. High Heat Flux and Thermomechanical Behavior

The in-vessel components (IVC's) (or plasma-interactive components) will be subject to high heat fluxes, ranging up to 5-10 MW/m^2 during normal operation to much higher loads for short times if disruptions occur. The behavior of a component will, of course, depend on its design. For example, parameters of importance for a limiter designed with ceramic tiles attached to a high-sink subplate will include the heat flux distribution, the pulse length, the tile material and thickness, the gap conductance and the thermophysical properties of the materials.

Off-normal operation results in a very high heat flux which causes high stresses in materials. The high heat fluxes can also produce erosion due to melting and/or evaporation or sublimation. The heat fluxes, stresses, and erosion due to off-normal operation dominate the requirements for material mechanical and thermal properties. These requirements are very stringent and are strong drivers in any design. Off-normal operation is being extensively studied on present machines. TFTR, JET, and JT-60 will provide very useful data to reduce the uncertainty of the requirements.

1. Heat Transfer and Thermomechanical Response

Thermal response involves thermohydraulics, structural design, and materials.

a) Energy Removal and Recovery

In-vessel components provide the particle, energy and information interface between the plasma and surrounding reactor. In a reactor, one should recover this energy at high temperature. Thus, while energy removal is a near-term requirement, energy recovery is an additional long-term goal. The issues and potential solutions are the same for both the near and long term, except that energy recovery requires high-temperature coolant and leaves less margin for heat transfer uncertainties.

In addition to water, long-term options include coolants such as helium, liquid metals, organics or molten salts. The choice is closely related to the primary blanket coolant choice.

Adequate characterization of the heat source profile is a general issue since there is usually little tolerance for unexpected local hot spots, but such characterization is specific to the fusion device.

The primary uncertainties in heat transfer are in the critical heat flux (CHF) (water, organics, molten salts), pressure drop (liquid metals) and flow stability (all). For up to about 5 MW/m^2, there is reasonable confidence in design predictions, although one-sided CHF measurements for liquid coolants (water) at long L/D are desirable. Liquid metals will require substantial effort to reach these heat flux levels because of uncertainties in MHD effects in realistic geometries. For heat fluxes above 5 MW/m^2, it is necessary to know the relevant CHF limits. If heat transfer augmentation is used, it will be necessary to determine the corresponding heat transfer and pressure drop correlations.

At higher heat fluxes and associated higher flow velocity, flow distribution and stability are bigger problems. The onset of flow and channel erosion problems is not well-defined. Practical experience with water suggests 20 m/s as an upper limit for straight channels. For comparable inertial force, this suggests limits of 300 m/s for He (5 MPa), 30 m/s for Li and 20 m/s for organic coolants. (Higher values require particular attention to design of headers and structural support, and may not be entirely predictable). Practical helium flow limits of about 1/3 sonic speed also imply about 300 m/s at 5 MPa. Higher pressures appear possible. For organic and molten salt coolants, further uncertainties include the control of fueling under high temperature or irradiation-induced decomposition (e.g., by smooth geometry and coolant chemistry control).

b) Thermomechanical Effects

Thermal fatigue and bond integrity are two problem areas that are of long-range concern. For in-vessel components, the growth of thermal fatigue cracks can cause: 1) debonding of armor tiles, 2) coolant leakage, or 3) catastrophic fracture of pressurized components. Prevention of each of the three failure modes will require extensive testing and improved models for failure analysis.

The important basic materials properties are fatigue crack growth rate and fracture toughness for the appropriate environment, i.e., vacuum, high temperature, water, neutrons, etc. The interaction of fatigue with creep damage will become important when pulse lengths become long. For stainless steel, the materials data base is adequate for design purposes. However, for materials relevant to plasma interactive components, such as beryllium, molybdenum and copper alloys, the data base for thermal fatigue is inadequate. Even if the basic properties are well known, thermal fatigue is such a complex phenomenon that accurate predictions are unlikely. This places a greater reliance on full scale testing to qualify these components. As the availability, number of cycles, heat fluxes, neutron damage, and operating temperatures increase for long-term applications, thermal fatigue will clearly become a critical issue that may play an important role in deciding between pulsed or steady-state fusion devices.

For long-term applications the effects of neutron damage will pose serious problems for bond integrity. Differential swelling between the two bonded materials will generate interface stresses that will grow larger with time. Also, gas production and impurity transport to the interface may contribute to bond failure. Finally, neutron induced embrittlement will reduce the ductility of metallic heat sinks and pose problems for the bond integrity. The response of bonded structures to neutron irradiation is a serious problem that will require additional attention. It may be that stresses at interfaces will be affected by radiation-induced creep.

c) Disruptions - A Tokamak-Specific Issue

The forces produced during the current decay phase of disruptions determine to a large extent the structural support required for the vacuum vessel and for plasma interactive components. The major forces generated during this phase result from coupling of the induced eddy currents in the vessel of other plasma interactive components with the externally applied magnetic fields.

The best hope for eliminating the disruption problem in tokamaks is the detection of disruption precursors followed by the activation of soft landing feedback systems. Obviously, research efforts aimed at the design of durable plasma interactive components must continue. If plasma interactive components capable of surviving multiple disruptions cannot be designed, future tokamaks may require frequent component replacement. This will seriously impact the availability of these devices.

G. Materials Development and Selection

Graphite is the leading candidate, since its refractory nature and the lack of a liquid phase makes it an almost ideal material from the point of view of disruption survivability. Low cost, fabricability, and low atomic number are also positive attributes. However, conditioning techniques for graphite need further development. Also, with typical specific surface areas of $\approx 1\ m^2/g$ for typical nuclear grade graphites, internal porosity can be a major source of oxygen due to water uptake during air exposure.

Chemical erosion is a major disadvantage of graphite, especially in the temperature range of 70-1300K. Erosion yields of unity are observed at 2000K for light ion bombardment. More data are needed on the high temperature erosion of graphite at the particle fluxes expected in some ignition experiments. The bulk properties data base of graphite relating to heat transfer are adequate. However, more work is needed on the ultimate strength at elevated temperatures, and on thermal shock resistance at high power levels. High temperature lattice diffusivity of hydrogen isotopes in graphite is poorly understood.

The use of graphite for large area limiters, as in TFTR and ALT-II in TEXTOR, has consequences for particle handling. Graphite limiters or carbonized walls can store large amounts of hydrogen. It has been observed that the recycling of hydrogen from carbon limiters can exceed unity, requiring some particle exhaust to control density. Furthermore, pellet injection and possibly neutral beam injection will require some particle removal. In such cases, pump limiters or divertors taking high heat loads will be required, even in short pulse ignition experiments.

Beryllium has excellent thermal properties and does not exhibit elevated erosion characteristics seen in graphite at elevated temperatures. Its main drawbacks are the melt layer formation during disruption and safety considerations (at least during fabrication and initial assembly). With active heat removal, beryllium can be a viable candidate in long-pulse experiments. Data needs include more detailed information on disruption response and on hydrogen retention and release characteristics.

New materials are being developed to overcome shortcomings of current technology. These include, for example, carbon composites, graphite doping (C-SiC), Cu-silicon carbide, graded coatings and Cu-Li alloys.