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Technical Assessment of the Critical Issues and Problem Areas in the Plasma Materials Interaction Field

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FOREWORD

Technical Assessment of the Critical Issues and Problem Areas in the Plasma Materials Interaction Field

At the present time, the magnetic fusion energy effort in materials development is incorporated within the Reactor Technologies Branch of the Division of Development and Technology in the Office of Fusion Energy (OFE). Also included within the Reactor Technologies task area is work on High Heat Flux Materials and Component Development (HHFMCD), which is closely linked to the Plasma Materials Interaction (PMI) area.

As a reflection of increased programmatic emphasis on alternate concepts, the PMI Task Group has been expanded to include members of the mirror and compact toroid communities. The chairmanship of this Task Group remains with Dr. Walter Bauer, Sandia National Laboratories (SNL), Livermore, California. In addition, a new task group on HHFMCD has been initiated under the chairmanship of Dr. Mark Davis, Sandia National Laboratories (SNA), Albuquerque, New Mexico, and Dr. Wilhelm Gauster (SNA) functions as coordinator between the activities of the two groups. Two new Technical Assessments and Program Plans have been initiated in the PMI and the HHFMCD areas.

This PMI Technical Assessment is the first update of the Fusion Reactor Materials Program Plan which was completed in 1978 and which consisted of four elements:

- Alloy Development for Irradiation Performance (ADIP)
- Damage Analysis and Fundamental Studies (DAFS)
- Plasma Materials Interaction
- Special Purpose Materials (SPM)

In the intervening six years, significant progress has been made in each of these areas. In particular, the PMI area has evolved from a program focused on the generation of basic scientific data for such processes as sputtering and blistering and ion irradiation performed in the laboratory environment to a program deeply involved with diagnostics, edge modeling, coating development, surface conditioning, and high heat flux and PMI systems design and fabrication.

It is widely recognized that it is of vital importance for these programs to be able to focus part of each of the individual elements of this task area on the design, fabrication, and maintenance of near-term PMI systems which provide the integrating function for all separate elements of the program.

Much of the success of the PMI (and HHFMCD) program(s) is a direct result of this focusing. At present, tasks are being carried out in (and linked to the success of) present and planned magnetic fusion physics facilities, both within the U.S. and abroad. International collaboration and joint design on such components as pump limiters, divertors, halo scrapers, diagnostics, and wall conditioning have been performed both in conjunction with the U.S. plasma physics community and that of other nations. Such Reactor Technologies PMI and HHFMCD tasks are being carried out in and for such devices as TFTR, TEXTOR, JET, and MFTFB.

These Technical Assessments and Program Plans are being prepared by task groups composed of persons from the various laboratories and contractors that contribute to the Magnetic Fusion Energy program. Each task group of six to ten principal investigators and/or consultants work under the guidance of a chairman drawn from a national laboratory and his counterpart, a staff member of the Reactor Technologies Branch. In the case of PMI and HHFMCD, the counterpart is Dr. Marvin M. Cohen. The efforts leading to the present Technical Assessment in the PMI area were chaired by Professor Robert Conn of the University of California, Los Angeles (UCLA). The Program Plan, which will represent the OFE strategy for the implementation of a program designed to address the requirements set out in the Technical Assessment, will be chaired by Dr. Bauer. In the area of HHFMCD, the Technical Assessment will be chaired by Professor Mohamed Abdou, UCLA, and the Program Plan chaired by Dr. W. Gauster and co-chaired by Professor Abdou.

Each chairman operates through a number of ad-hoc sub-task groups which were charged with problem definition and program planning for specific technical areas.

The assumptions inherent in the planning process are (1) the demonstration of the scientific feasibility in the TFTR by 1987, and (2) the operation of a modest long-pulse ignition machine before the end of the century. Beyond these assumptions, the Technical Assessments and Program Plans deal with generic materials and diagnostic and systems integration needs, irrespective of the magnetic confinement system. To the extent that such generic problems apply to hybrid reactors and laser fusion reactors, the plans are applicable to them as well. However, they do not include tasks specific to hybrids (fuels, for example) or laser fusion. (For example, optical materials and ultra high frequency pulsing or ramp rates.)

The emphasis of the present planning process is to examine potential problems, state of technical readiness, and to prioritize materials-related requirements which must be satisfied for the successful development of fusion reactors.

It is important to realize that the assessments and plans describe problem areas, and the approach to solutions as seen today are significantly different from those which were outlined in 1978, and that these will have to be updated periodically. Furthermore, they should be regarded as outlining the major avenues to be explored, rather than as a detailed road map. Although a task structure will be outlined in the Program Plans, the detailed approach to the solution of specific problems will be proposed by individual investigators.

Including memberships on sub-task groups, a total of over 50 individuals will be involved in various stages of the operation of the PMI and HHFMCD

Technical Assessments and Program Plans. The wide representation of national laboratories, universities, and industry was encouraged to remove institutional bias to the greatest extent possible.

In conclusion, I would like to take this opportunity to thank all of the members of the Task Group and the technical community who contributed to this effort and who continue to be the most important element in the success of the reactor technology area.

Gregory M. Haas

Gregory M. Haas, Chief
Reactor Technologies Branch
Division of Development and Technology
Office of Fusion Energy
Office of Energy Research

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Executive Summary

A technical assessment of the critical issues and problem areas in the field of plasma materials interactions (PMI) in magnetic fusion devices shows these problems to be central for near-term experiments, for intermediate-range reactor devices including D-T burning physics experiments, and for long-term reactor machines.

Critical technical issues are ones central to understanding and successful operation of existing and near-term experiments/reactors or devices of great importance for the long run, i.e., ones which will require an extensive, long-term development effort and thus should receive attention now.

Four subgroups were formed to assess the critical PMI issues along four major lines: 1) PMI and plasma confinement physics experiments; 2) plasma-edge modelling and theory; 3) surface physics; and 4) materials technology for in-vessel components and the first wall. The report which follows is divided into four major sections, one for each of these topics.

The main technical issues, questions, and immediate needs for each sub-area are:

A. PMI and Plasma Confinement

1. Poloidal divertors or pump limiters in toroidal confinement devices. Will poloidal divertors be necessary to solve the impurity control and ash removal problems presently encountered and anticipated on longer pulse, strongly auxiliary heated devices, or can pumped limiter schemes be devised to handle these tasks? Is confinement fundamentally altered by the approach chosen for particle removal and impurity control? The answer to these questions will fundamentally affect the configuration and complexity of toroidal fusion reactors. As such, this technical area deserves the highest priority.

2. Effects of auxilliary plasma heating and alpha particles on impurity generation and control. What are the detailed effects of radio frequency heating and current drive (Ion Cyclotron, Lower Hybrid, and Electron Cyclotron), Neutral Beam Injection, and fusion produced alphas on edge impurity generation from

limiter and first wall components, and on impurity transport both through the edge and in the main plasma?

3. Alternate conditioning and operational techniques required to suppress impurities brought in with neutral beams or evolved from RF antenna and waveguide structures. Impurities which accompany neutral beam injection pose a special problem for mirror devices, where these impurities can fill in the thermal barrier in the end plugs.

4. Effects of long pulse, high power discharges on the "conditioning" of the first wall and edge structures.

5. Fueling in large, long pulse devices and reactors. It is likely that fueling via pulsed gas through the edge is unacceptable for large, long pulse devices, especially if tritium inventory must be minimized. The only viable alternative appears to be pellet fueling. Therefore, the development of quasi-steady-state tritium injectors must proceed without delay.

6. Plasma edge diagnostics. Edge diagnostics needing the most development are those capable of measuring plasma potential, and those which can give detailed three dimensional spatially-resolved information of the edge plasma properties.

7. Availability and reliability of primary plasma-supporting technologies and components. As long pulse length, high duty cycle devices are anticipated, the general reliability and availability of auxiliary components must be greatly improved. Much of this reliability will come with improved conditioning techniques for neutral beam and RF components, and these techniques are closely related to those required for first wall and limiter conditioning.

B. Plasma Edge Physics Theory and Modelling

The most important immediate needs for modelling the plasma edge (or scrape-off-layer) include:

1. The development of a combined edge plasma and neutral transport model, including the physical processes required to analyze present experimental results.

2. The development of an impurity transport model, ideally coupled to the plasma transport model, containing realistic impurity source models.

3. The development of a time-dependent model of the evolution of the device walls and limiters, including material redeposition and the structure of the reformed wall, and including such topics as the wall tritium inventory.

4. Modeling of solutions to the particle and power control problems expected in future devices.

The most critical atomic and surface physics data needed for theory and modeling, in priority order, are:

1. Reflection coefficients and reflected particle energy and angular distributions at low incident energies (1 to 100 eV), on single and multi-elemental surfaces.

2. Hydrogen trapping, diffusion, and re-emission rates from surfaces.

3. Sputtering yields, distributions, and physical properties of original and redeposited surfaces, for elemental and compound materials.

4. Radiation rates for electron impact on atoms and H_2 at low (1-100 eV) temperatures.

5. Atomic rate coefficients at low temperatures.

6. Data from comprehensive edge diagnostics on present and future devices, so that global transport models can be verified.

C. Surface Physics: Experiment and Theory

Surface physics technical issues and problem areas conveniently fall into three topical headings: hydrogen recycling, impurity generation and vacuum and surface control. For each of these topics, the priority technical issues are:

C-I. Hydrogen Recycling

1. Reflection coefficients and reflected particle energy and angular distributions at low incident energy (1-100 eV) for single and multicomponent surfaces. This is the most critical problem and is the same as is listed under theory and modelling. Improved surface physics theory is also required for this energy range.

2. Hydrogen trapping and desorption measurements by ion-impact, photons and electrons. This need also is the same as identified under theory and modelling.

3. Tritium transport in first wall and in-vessel components. This topic is critical for near-term D-T machines. Tritium permeation will be important for high duty cycle devices. A reasonable data-base and theory exists for metals.

4. Experiments and theory of hydrogen/tritium interactions with low Z materials and coatings, e.g., graphite, the carbides, etc., at relevant operating temperatures.

C-II. Impurity Introduction

1. Sputtering yields as a function of selected parameters such as angle of incidence. Also, hydrogen and self-sputtering from surfaces with redeposited material. This work is required to add important data to a general sound data base.

2. Evaporation/sublimation and thermal shock data for composite and neutron-irradiated materials. Generally, the thermal data base is good and numerical codes are available to analyze thermal response.

3. Chemical erosion rates and influence of simultaneous processes such as redeposition.

4. Arcing data for special materials and coatings such as graphite and the carbides.

C-III. Vacuum and Surface Control

1. Physics and chemistry of surface conditioning of in-vessel components, especially those subject to high heat and particle loadings. This topic is not well understood yet is central to producing clean plasmas. Both experiments in confinement machines and in the laboratory are needed.

2. Getters for use in near-term devices with the characteristic of a low regeneration temperature.

3. Desorption and secondary electron emission data and secondary electron emission suppression for special components such as RF transmission and launching structures.

D. Materials Technology

For all confinement approaches, the fundamental materials technology problems for invessel components are related to the ability of the material to survive the plasma environment while mitigating the deleterious effects on the plasma caused by the introduction of impurities. The components must be able to withstand repeated thermal cycling, physical and chemical erosion, and in some cases thermal shock caused by a sudden dump of plasma energy. Efforts are focussing on the development of low Z refractory and metallic coatings and cladding on metal substrates. For long pulse machines, coatings of 1 cm thickness may be required.

The technical assessment has identified the critical issues and problem areas to be:

1. Develop the technology to apply thicker coatings to in-vessel components or plasma-side materials. Two promising approaches for low Z and refractory materials are plasma spray deposition and bonding of monolithic overlays. For plasma spraying, it must be determined that adequate coating thickness and density can be attained.

2. Develop in-situ and self-coating materials for the intermediate and long-term.

3. For the immediate and intermediate term, candidate protective surface materials include, in non-priority (alphabetical) order, Be, BeO, C, Mo, SiC, Ta, TiC and W. Candidate materials for cooled substrates are copper alloys and vanadium alloys. Research and development should concentrate on characterizing, developing, and testing both coatings and tile-to-substrate attachment techniques. Specifically, mechanical and thermal testing of joined materials needs emphasis. Further, a better understanding of damage mechanisms is required to select the materials that will perform best in the long term. Research should also be directed at characterizing interfaces such as those between tile and substrate of a duplex structure.

4. Testing and analysis of specific electromagnetic configurations and materials under expected electromagnetic conditions. These are required to establish feasible approaches to avoiding the arcing problem.

5. Tritium permeation through the surface of the first wall and IVC's. This is a major problem area influencing on-site inventory, handling, and safety. A program to evaluate the magnitude of the permeation problem and for developing solutions to the problem, should it be significant, is required. Specifically, one must characterize: a) the production of trapping sites for tritium by neutrons; b) release of tritium through interfaces; and c) designs that act as tritium barriers.

In summary, for materials technology and for each concept or design, a development effort should:

1. Identify potentially viable coating or cladding systems for specific applications;

2. Establish the feasibility of fabricating these systems reliably by identifying key properties and relating them to process variables;

3. Determine the effects of hydrogen permeation, fatigue cycling, neutron damage, etc., as appropriate, on interface integrity in a comprehensive testing program;

4. Demonstrate the fabricability of full-scale components, including non-destructive evaluation; and

5. Perform integrated heat removal tests of prototype components as part of the component development process.

Part A

Technical Assessment of the Critical Issues and Problem Areas in the Plasma-Materials Interaction Field: Edge Plasma Physics in Confinement Experiments

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Plasma-Materials Interactions:

I. Introduction

We examine in Part A the critical issues and problem areas associated with the interplay between edge plasma physics and plasma wall interactions. The report is organized by topic, first to examine issues which are relevant to a generic magnetic confinement device and, at the same time, issues specific to the tokamak are addressed. Special mirror-related problems are examined in section X, and alternate concept specific issues are examined in section XI.

The main issues and questions which require the most attention over the next several years are enumerated as follows, roughly in order of importance:

1) Will poloidal divertors be necessary to solve the impurity control and ash removal problems presently encountered and anticipated on longer pulse, strongly auxiliary heated devices, or can pumped limiter schemes be devised to handle these tasks?

2) What are the detailed effects of radio frequency heating and current drive (ion cyclotron, lower hybrid, and electron cyclotron), neutral beam injection, and fusion-produced alphas on edge impurity generation from limiter and first wall components, and on impurity transport both through the edge and in the main plasma?

3) Alternate conditioning and operational techniques are required to suppress impurities brought in with neutral beams and evolved from RF antenna and waveguide structures.

Impurities which accompany neutral beam injection pose a special problem for mirror devices, where these impurities can fill in the thermal barrier in the end plugs.

4) Effects of long pulse, high power discharges on the "conditioning" of the first wall and edge structures.

5) It is likely that fueling via pulsed gas through the edge is unacceptable for large, long pulse devices, especially if tritium inventory must be minimized. The only viable alternative appears to be pellet fueling. Therefore, the development of quasi-steady-state tritium injectors must proceed without delay.

6) Edge diagnostics needing the most development are those capable of measuring plasma potential, and those which can give detailed three dimensional spatially resolved information of the edge plasma properties.

7) As long pulse length, high duty cycle devices are anticipated, the general reliability and availability of auxiliary components must be greatly improved. Much of this reliability will come with improved conditioning techniques for neutral beam and RF components, and these techniques are closely related to those required for first wall and limiter conditioning.

II. Wall Conditioning

It appears that problems associated with gaseous low z impurity species (H_2O,CH_4,CO_2), which are loosely bound to the first wall surfaces, have largely been solved on presently operating, short pulse, devices. Well established techniques of vacuum preparation, baking, and glow and pulsed discharge cleaning can be applied, so that within a reasonable period of time after exposure of the system to air, relatively clean, reproducible plasma discharges can be obtained. Once a sufficiently clean first wall has been prepared, the issues of wall conditioning are related more closely to the specific materials chosen for that wall: chemical sputtering in graphite; the need to develop a micro-crack structure in Ti-C coatings; recycling properties of the materials for hydrogen isotopes; physical sputtering processes.

The issue of recycling is coupled closely to conditioning, as well as to the choices of materials and fueling methods (see section IX). It may well be that surface preparation techniques can be devised to make graphites, for example, which are less able to soak up tritium. This would have important consequences when tritium inventory is examined.

In some respects, present conditioning techniques may not extrapolate to long pulse devices. Present modes of machine operation usually entail some discharge cleaning for a period of a few hours before each day's run. When pulse lengths are in the range of hours or days, it will be the hot, high density discharge itself which will "condition" the walls, and some experimental simulation of the effects of this must be undertaken. A recontamination of the walls due to the build up of the low Z gaseous impurities must be avoided, and these issues have not been addressed with sufficient care to date.

III. Impurity Control

The main issues related to the impact of edge plasma physics on impurity control are:

1) Thermal edge plasma conditions

Density, temperature and edge potentials combine to determine sputtering rates (including self-sputtering) and heat loads due to the thermal plasma.

2) Non thermal edge components

These include fast neutrals created by charge exchange, ICRF and neutral beams, fast electrons produced by LH current drive and ECH, and in the future, fusion produced alphas. Each of these components can have direct effects on impurity generation through sputtering or heating, as well as indirect effects by perturbing such things as edge sheath potentials or impurity transport into the main plasma.

3) Edge impurity transport

The details of impurity transport in and through the edge plasma have important consequences for the central impurity content which results from a given edge source influx. Redeposition of desorbed and sputtered material onto the high heat flux components may be a necessary prerequisite for the surfaces of these structures to have reasonable lifetimes, and the details of impurity transport in the edge strongly influences the redeposition process. Experimental investigations which can shed light on this transport include laser blowoff and impurity gas puffing, as well as detailed measurements of toroidal and poloidal asymmetries in intrinsic impurity densities. Laser fluorescence diagnostic techniques can make a large contribution to this effort. Edge impurity transport is a fully three dimensional problem, and future emphasis must be placed on experiments which examine the spatially localized phenomena which occur.

4) Impurity transport in the main plasma

Cross field impurity transport through the main plasma, coupled with transport through the edge and the size of the influx, determines the ultimate on-axis impurity levels which can radiate power from the core of the plasma. The areas requiring further study include: problems induced by neutral beams (especially counter injection), ICRF, Lower Hybrid current drive, and ECH; for long pulse operation, conditions leading to very long confinement must be avoided; important profile parameters which affect impurity transport (eg., q, j, T_e, T_i, n_e, potential) must be determined.

IV. Limiter/Divertor

The choice between the use of limiters or divertors to define the edge of the plasma definitely falls into the category of most critical PMI issues. While limiters are in principle a simpler technological solution for the plasma material interface, many serious problems associated with their use are still unresolved. The main advantage of the poloidal divertor appears to accrue from the fact that the plasma material boundary is further removed from the main body of the hot plasma, at least in the sense of distance along field lines. Thus, most impurities generated at the divertor plates are reionized and then redeposited on the plates before they find their way into the main plasma. There has been much theoretical effort directed towards understanding these effects (see theory section), and considerable experimental effort as well (PDX, ASDEX, DIII).

The effects of edge plasma conditions on global energy confinement have been seen in the dramatic difference between the H and L mode discharges with NBI heating. To date, the H mode has only been achieved in plasmas with poloidal divertors. The details of just what causes the improved confinement are not well understood, but the experiments on ASDEX and PDX indicate that it is a relatively small change in the electron temperature near the edge of the plasma which leads to the improved confinement mode. Correlated with this temperature change is a decrease in edge neutral density. Studies must continue on this matter, along with the attempts to achieve H mode without the use of a divertor.

The coupling of ICRF and LHRF waves across the edge plasma is an area which also falls into the domain of edge plasma physics. In the case of

LH, the only technique presently available to drive tokamak currents on a steady state basis, a cold high density edge plasma strongly damps the waves. This leads to poor current drive efficiency, and the RF power deposited at the edge can also lead directly to impurity generation problems. On the other hand, a relatively hot edge is bad from the point of view of sputtering at the limiter surfaces. The situation with a divertor, particularly in a cold gas fuelled discharge, may also be bad from the point of view of RF penetration. There have been no experiments to date on the coupling of either ICRF or LHRF through the edge of a tokamak operating with a divertor, and if the divertor option is chosen for a device such as TFCX, such experiments must be carried out before such a choice is made.

Even if impurity generation problems can be solved with limiter configurations (in the presence of high power RF and NBI auxiliary heating) there are the separate problems of particle pumping and ash removal to be dealt with. While preliminary studies with pumped limiter schemes have shown a great deal of promise, continued studies with various forms of pumped limiter are essential. In particular, nearly axi-symmetric toroidal belt pumped limiter configurations should be developed. Ideally, experiments should be performed to do as definitive and direct a comparison as possible between toroidal pumped limiter and poloidal divertor configurations.

V. Edge Plasma Diagnostics

1. Present Implementation and Suggested Improvements

The major aims of edge plasma diagnostics are to increase our understanding of the important phenomena and to diagnose plasma and wall conditions that may be deleterious or favorable to the achievement of controlled thermonuclear fusion. The techniques that are currently used in these endeavors fall into four broad categories: probe, μ wave, spectral, and photographic. These methods overlap, both in aims and in technologies.

A. Probes

Probes using surface analyses techniques, such as AES, RBS, channelling and SIMS, remain expensive to implement and difficult to employ because of the complex and slow analysis equipment that is frequently kept off-site. However, the type of data these provide have not yet been routinely produced by any "standard" in-situ diagnostic. These data include impurity density and hydrogen energy distribution. It is likely, however, that spectroscopy (especially laser fluorescence) and charge exchange diagnostics can and will produce many of these data.

Other surface type probe techniques are relatively easily implemented. These include bolometry, resistance change and thermal desorption. Their simplicity and accuracy continue to increase their use. They have extreme utility in assessing power and particle flow in the edge.

There has been a great resurgence in Langmuir probe usage on tokamaks. The techniques are being refined. Double and triple probes are now commonly used. Frequency response has been increased to study fluctuations and wave penetration.

Other types of electrostatic probes, such as gridded energy analyzers, have been used, but are not yet commonplace.

Overall, the use of probes is justifiably increasing. A main drawback has been the limited penetration depth for material probes. This is being somewhat ameliorated by the use of advanced materials and designs.

B. Microwave

Microwaves have been used for density measurements in divertor chambers as well as in the main plasma body. Since microwaves measure line integrated densities, an array is frequently necessary to unfold the local density. This has not yet been done in the plasma edge. The importance of the measurements in the edge should have encouraged a large effort, but this has not yet materialized.

C. Spectral

This category includes photon and particle spectroscopy. The main emphasis is photon spectral diagnostics has been in the plasma core. An exception is the laser fluorescence work which has been applied to hydrogen detection and to low charge states of metals. These experiments have started to unravel the complex questions of recycling and impurity generation. Another exception has been the study of molecular hydrogen. These experiments should no doubt be extended to other molecules, such as methane. Thomson scattering can also be used, in principle, to make measurements of the edge plasma, and much effort could and should be directed toward that end.

Particle spectroscopy, i.e. energy dispersive detection of edge neutrals emanating from the plasma, has been accomplished in one laboratory and is being attempted in two others. These experiments should result in a more complete picture of hydrogen recycling.

A drawback to all the spectral diagnostics is that they are local. Toroidal and poloidal asymmetries are expected, and have been observed, in the edge plasma, and present spectral diagnostics are not capable of observing many different locations. This must be corrected. Multi-spectral (channel-plate) devices may solve the problem.

D. Photographic

Still and kinetic photography of plasma and the surrounding structures have been made in wavelength ranges from infrared to x-ray. The most common, infrared and visible, are extremely useful when applied in conjunction with videotape or other instant playback methods. These allow rapid evaluation of potentially dangerous situations, such as structural damage. This work must be accelerated, because of the increasing thermal load expected in reactors.

2. New Diagnostics Required

A main improvement required for diagnostics is real-time data presentation for possible feedback control. Topics requiring extensive new efforts are:

- plasma potential

- poloidal and toroidal spectral asymmetries
- neutral atomic and molecular density
- local gas pressure

3. Tritium Impact

The main impact on edge plasma diagnostics due to the use of tritium would be in the areas of:

- maintenance and
- signal detection in a noisy (neutron) enviroment.

VI. Auxiliary Components

The auxiliary components of fusion reactors and confinement experiments are generally considered to be the systems which heat or inject particles into the plasma. At this time, the major auxiliary components include neutral beam injectors, rf systems for heating and current drive, ECRH heating systems, and gas and pellet injectors for fueling. Each component contains many subsystems whose performance depends on plasma materials interactions. The areas of concern in each subsystem for fusion are conditioning procedures and time requirements, electrical breakdowns, impurity generation and control, and lifetime limiting erosion. These problem areas will be discussed for each of the auxiliary components mentioned. Gas puffing and pellet injection fueling systems are now considered remote enough from the plasma not to be directly involved in PMI issues at this time.

Neutral beam lines contain many subsystems with severe PMI problems. The plasma generator in the ion source produces impurities which appear in the neutral beam. Oxygen from the walls and heavy refractory metals from the cathodes and grids can comprise several percent of the beam current. Cooled stainless steel sources or titanium gettered walls are now being investigated to eliminate impurity sources, along with rf plasma generators to remove cathode related impurities. Long pulse or continuous operation of the ion source decreases the impurity level in the beam through discharge cleaning of the source walls, but additional data are needed to assess the effectiveness of this direct cleaning technique.

The ion accelerator is the most critical area in neutral beam injectors, due to the high heat fluxes and large voltage gradients.

Copper grids developed at ORNL dissipate the heat well, survive serious arcs, and can be conditioned to hold high voltage gradients of about 40 - 50 kV/cm . Molybdenum grids of LBL and LLNL are known to stand off higher voltages, but fabrication and cooling is complicated. Conditioning methods for the grids have been developed empirically, with some guidelines developed through years of experience. Cleanliness during fabrication and assembly is essential, but no additional attempts at in-situ cleaning prior to use have been made. Conditioning generally consists of increasing the beam voltage and current at near optimum optics of the grids until breakdown no longer occurs. The grids are typically conditioned to beam voltage in excess of the planned operating level to provide reliable operation at the designed parameters. The breakdown arcs must contain enough energy to clean the surface, but not to damage the grid. This arc energy depends on the grid material and design, and in general has been determined experimentally for each grid set. The PDX sources require typically 1 - 2 thousand pulses of 0.1 second duration to achieve reliable operation at 50 kV. Higher voltages require more pulses, depending on the gap size and grid material. While the integrated beam time for PDX source conditioning is less than 3.5 minutes, it is not clear that longer pulse lengths will decrease the conditioning time required. Proper aperture design to minimize very high heat fluxes on the grids permits faster conditioning times with less chance of grid damage. Starting conditioned grids which have not been exposed to air requires 10 - 20 times fewer pulses depending on the vacuum conditions and time since last operation. Cleanliness and degassing of the grid surfaces is clearly required for all injectors. Relatively low power density cleaning techniques appear to be effective for conditioning.

Minimizing the time between pulses or providing cleaning discharges regularly also decreases the impurity level in the neutral beams.

The neutral beam duct leading to the target plasma must handle high power fluxes at grazing incidence, and neutral gas evolution, erosion, and impurity generation at the walls are problem areas. Conditioning of the duct walls is necessary to eliminate duct blockage from reionization of the beam particles by evolved neutral gas. The duct must also be cooled to dissipate the high power fluxes. Efforts at PPPL have succeeded in conditioning the ducts for pulsed beam systems by using high gas conductance duct designs and many beam pulses. Gettering of the duct walls to provide clean surfaces is useful, but additional cleaning techniques using rf or glow discharge methods have not been attempted. Impurity control in the duct has not yet been addressed comprehensively.

The neutral beam dumps also have PMI problems with sputtering and impurity generation. This area will be discussed under the high heat flux and materials area of this technical assessment.

Heating and current drive methods utilizing rf systems have many areas of concern related to plasma material interactions. The rf systems use antennas or waveguide structures to launch the waves. ICRF antennas are typically fed by a coaxial transmission line and inductively couple energy to the plasma through a Faraday shield. The antenna, feedthrough structure, and transmission line must hold off voltage gradients to grounded structures nearly as large as those in neutral beam grids, and breakdown is a serious problem. Proper design of these components for reasonable voltage gradients eliminates some of the breakdown problems, but cleanliness and conditioning are very important. The antenna must be located close to the plasma (< 5 cm) for efficient coupling, and the Faraday shield must dissipate the high power fluxes of the edge scrape off plasma which exists behind the main limiters or seperatrix surfaces. The short pulse experiments in progress at this time utilize passively cooled copper or stainless steel which must be conditioned like any limiter. A new program at ORNL to actively cool the Faraday shield with vapor deposited copper surfaces incorporating water cooling channels, similar to those in neutral beam accelerators, is underway to provide braze-free, cooled shields. Problems associated with cleaning, conditioning, and erosion of these surfaces during operation in the edge plasma will probably be similar to those associated with limiters and the solution will probably also be similar.

Waveguides for rf coupling to the plasma, as are typically used for lower hybrid heating and current drive, do not have the shield PMI problems, but do have to deal with very high electric fields from the propagating electromagnetic waves. The waveguides must be pressurized through the region of cyclotron resonance to avoid breakdown, so that a window is required near the plasma edge to separate this pressurized region from the plasma. Breakdown at the waveguide window is induced by the high electric fields and particles or photons from the plasma. Techniques to eliminate this problem include providing very clean waveguide surfaces and surface coatings which impede emission. Complicated waveguide geometries, such as ridge waveguides for example, add to the breakdown problems due to very high power densities and rough edges. More data are needed in this area in the presence of plasma to determine the capabilities of waveguide systems.

ECRH systems are used in the heating of mirror machines, tokamaks, bumpy tori, and other machines. The high magnetic fields of relevant confinement machines will require the use of high frequency systems such as gyrotrons or free electron lasers to produce high power at the 30 - 120 GHz resonance frequencies. Breakdown and erosion in the electron guns and collectors will be a problem in the very high power (> 1 MW) systems required for reactors. However, the output window, or the output mirror in quasi-optical systems, which isolate the reactor from the 10^{-8} torr vacuum in the "tube", will be subject to severe power densities. Electrical breakdown at this surface and in the waveguides are problem areas being addressed at this time. Breakdown and cleanliness in the waveguides leading to the plasma will have to be investigated further in the presence of plasma. Quasi-optical systems eliminate the waveguides, but still have the output semi-transparent mirror related problems.

For near term experiments, impurity control is a problem for all the auxiliary components. Conditioning techniques for the power densities of present devices appear adequate to permit operation with minimum probability of breakdown, but only after a considerable number of cleaning pulses. Reliability must be improved through a better understanding of the surface conditioning techniques. As rf systems are included in more of the near term machines, antenna designs which incorporate high power limiter properties (cooling, coatings, cleaning, etc.) will be necessary. Cleaning and conditioning of these components will be necessary, and will probably be based on the techniques developed for advanced limiters. A comprehensive PMI program to understand the breakdown processes and conditioning procedures in neutral beam lines with large surface areas at high voltage is needed to eliminate the largely empirical and time-consuming methods in use today. The use of neutral beams in future experiments will require new in-situ cleaning techniques for reliable operation in reasonable times. Longer pulse beam lines for the near term machines such as MFTF-B, Big D, and TFTR-U will need advanced conditioning methods to operate reliably with fast turn on times and high availability. The output window/mirror in ECRH systems must be further developed to provide reliable high power density performance.

Auxiliary components for future machines and reactors will not only be required to operate reliably, but will need to be turned on quickly to minimize any down time. The understanding of conditioning and cleaning for all the auxiliary technologies must be improved to the point where operation at the design levels is routine.

VII. - Vacuum Systems

Edge plasma physics has a number of important effects on the details of the pumping systems required in a tokamak device. The main issues are related to the steady state pumping throughput, which will depend on the global particle confinement properties of the device, and the pressure which can be maintained at the entrance to the pumping systems. The volume, density and global particle confinement time, together, determine the required throughput of the system (particles/time). The pumping speed required to maintain that throughput is then inversely proportional to the pressure which is maintained at the pump inlet.

A large number of experiments have been performed to investigate the processes which determine the edge pressures and particle confinement times. To date, essentially all devices operate with pulse lengths which are short compared to, or at most of the same order as, the relevant pump-out time scales of the associated vacuum pumping systems. In analogy with the inertial approach for limiter structures, whereby materials heat up during the pulse, and are only cooled in the time between pulses, the pump-out of presently operating fusion devices is usually negligible during the pulse, with the gas load being removed after the pulse. As very long pulse operation is anticipated, vacuum systems must be designed to handle steady state particle throughput. The details of the systems to handle this thoroughput will depend on the edge particle transport in the device, which in turn is strongly influenced by the precise design of edge hardware. Particle dynamics with pumped limiters have been under investigation for only a relatively short period of time. In the case of poloidal divertor configurations, much has been done with the closed

(PDX, ASDEX) divertor configuration, but relatively little with the more reactor relevant open divertor configuration (as on DIII and the proposed Alcator DCT).

VIII. Fueling

In large experiments, once $\bar{n}_e \cdot a > 10^{15}$ cm^{-2}, fueling with cold gas through the edge leads to relatively flat density profiles. Pellet fueling, particularly in these high $n_e \cdot a$ situations, yields considerably more peaked density profiles, as well as apparent improvements in global energy confinement, at least for ohmically heated discharges. That discharges manage to have peaked profiles at all, rather than hollow ones, with edge fueling remains a mystery. The affects of pellet fueling on other aspects of machine operation are also not well understood at this time. If the initial Alcator C pellet experiments are a good indication, however, it appears likely that pellet fueling is the preferred, if not the only viable, method for use on future large devices. If this is the case, then the development of appropriate injectors, capable of producing a steady stream of tritium pellets of sufficient size and velocity, must receive very high priority. *Fueling via neutral beams does not appear to* be a viable alternative.

Beyond flat density profiles, several other undesirable edge plasma features are associated with the use of edge fueling. High edge neutral densities and high recycling coefficients can lead to larger inventories of tritium, and higher energy losses due to charge exchange. The cold, high density edge plasma also impedes the penetration of Lower Hybrid waves used for current drive. If a divertor is used, it becomes that much harder to fuel through the edge, due to the effective screening of incoming neutrals by the scrape off plasma, which in turn puts a larger burden on pumping systems and divertor chamber hardware, as well as increasing tritium inventory.

The question of how a reactor will be fueled is one of the major issues needing resolution in the near future for successful operation of of an ETR type device.

IX. Mirror Edge Plasma PMI Issues

1. Plasma - Wall Conditioning

In current devices, glow discharge cleaning has been used for initial conditioning, with results that are similar to tokamak results (i.e., removal of hydrocarbons). Gettering is used extensively, primarily for reduction of hydrogen recycling, and to a lesser extent, for impurity control. Experiments are necessary so that gettering can be minimized in future devices such as MFTF-B.

Other methods - such as liquid-nitrogen cooled and gettered panels, and gas baffles - have also been used to minimize the cold gas influx in the thermal barrier region (end plugs). Some experiments have used external neutral beam tanks (with active pumping and baffles) to minimize the cold gas influx from neutral beam injectors.

Several machines have performed or are planning experiments to test the influence of wall temperature. Room temperature and 100° C wall are currently in use, and 400° C walls are planned.

2. Impurity Control

Current experiments have shown that radial transport of impurities is outward; no accumulation has been noted so far. The radiated power from impurities is small, usually only a few percent of the input power. One possible problem is the injection of high energy oxygen and nitrogen impurities by the neutral beams. This can be especially detrimental, as neutral beams are used in the thermal barrier region, and these impurities could "fill in" the barrier region. As discussed in a later section, research is underway to reduce the amount of beam-injected impurities.

Some startup modes (methods of generating a target plasma for the neutral beams and other heating devices), such as stream guns, have been identified as impurity sources. Other modes of operation, such as ECRH and ICRH, have been used successfully and these do not introduce impurities (at least so far). These modes will also be used on future machines.

3. Edge Plasma - Limiter/Divertor

The edge plasma plays a very pivotal role in the current machines, and will be even more important in future devices such as MFTF and the MARS reactor design. As currently envisioned, there will be no active pumping in the central cell section of future machines. Plasma particles transported to the edge will be removed because of poor axial confinement to the end regions of the machine where they will be pumped directly or trapped in an "edge-plasma scraper" or divertor.

In present machines, the edge plasma shields the core from cold neutrals, particularly in the end-plug regions. The fueling of this edge plasma - and therby its control - is accomplished by neutral beams, ECRH, and ICRH. Models of these interactions are currently being compared with edge plasma meaaurements - Langmuir probe measurements of electron temperature and density, and spectroscopic measurements of neutrals.

In addition to the edge plasma which ultimately escapes to the end wall of the machine, some of the higher energy core plasma also escapes because of a finite axial confinement time. In particular, there may be an electron flux even though the axial ion confinement time is very long.

Several schemes have been devised to convert this plasma loss directly into electricity. From the stand point of plasma - materials interactions, this plasma loss at the end wall can be similar to the interactions at a tokamak limiter or divertor. Therefore, as long as the edge plasma provides efficient isolation in the central cell and plug regions, most of the plasma-wall interaction will be in the end regions, where the plasma is expanding. For this reason, most of the emphasis for future machines should be placed on the edge-plasma scraper or divertor, the direct converter, and pumping in the end wall regions. This is also where there may be a large overlap between tokamak and mirror issues.

4. Auxiliary Components

Auxiliary heating - ICRH and ECRH - will be very important for edge plasma control. In addition, the antennas for these devices must be constructed so that they do not introduce impurities or cause any other plasma-materials interaction problems. As mentioned above, the impurities introduced by neutral beam injectors must be controlled. Gettering of the arc chamber and magnetic separation are two techniques currently being used for impurity reduction. Another neutral beam issue is the design of a beam dump to minimize sputtering of the wall and recycling. This may be particularly important during low-density startup phases. Various techniques have been tried to reduce cold neutral gas from neutral beams: gettered beamlines, baffles, and magnesium curtains. This is another region where there may by overlap between tokamak and mirror issues.

5. Specific Issues for Machines Being Built Such as MFTF

MFTF is designed for 30 second operation, so it will be an early test of long pulse issues. This machine will operate in a regime where recycling processes are in equilibrium, so wall conditioning will be important. Thirty second neutral beam operation will test beams and dumps, and because the neutralization efficiency of 80 keV beams is relatively low, intermediate dumps with bending magnets will be required.

6. Specific Issues for Reactors (MARS Design)

As with earlier devices, the major plasma materials interactions will be in the end regions. Only very limited titanium gettering will be possible. In contrast to tokamaks, there is less concern about problems associated with disruptions. The edge plasma will again plan an important role for plasma-wall isolation and pumping of the central cell. The major edge physics issues are associated with the formation of a plasma halo. The key issues are: initial formation of the halo; sustenance of the halo in terms of input power; coupling between the halo and core, particularly in the barrier region; pumping of halo plasma flow at ends; control of recycling; and interaction between fueling and the halo.

XI. Alternative Fusion Concepts, Critical PMI Issues

1. Introduction

Many of the PMI problems are the same for the alternative fusion concepts (AFCs) and the mainline approaches, especially, for systems of equivalent system power densities. The high power density (HPD) concepts have more demanding requirements due to the higher levels of neutron, radiation, heat, and particle fluxes. The level of effort on AFCs has resulted in a data base (theoretical and experimental) which is often insufficient to address these issues in the detail required to ensure rigorous conclusions.

The problems associated with PMI and High Heat Flux (HHF) are not long term issues for some AFCs. Current experimental devices (CTX, ZT-40M, HBTX-1A, and OHTE) operate near to the conventional reactor conditions in terms of plasma beta and heat flux for pulse durations of $\leq$ 25 ms. The problems associated with equilibrium, impurity control, and thermal stresses will have to be solved before optimum machine parameters can be obtained for the devices in operation now. Table I lists some representative AFC and tokamak devices with the estimated average first wall thermal load in MW/m^2 for three specific time frames, 0-3 years, 3-8 years, and "Reactor." Wherever possible, an estimate of the thermal load for a HPD option is also given.

TABLE I

Device	Average First Wall Thermal Loading (MW/m^2)	
	LPD	HPD
TFTR	0.80	
ZT-40M (200 kA)	1.00	
ZT-40M (500 kA)	< 2.50	
ZT-P	----	~ 8.5
HBTX-1A	< 2.00	
OHTE	< 2.00	
FRX-C	?	
CTX	?	
AFT-1	0.600	
Alcator DCT	0.300	
TFCD (SC)	0.25	----
ZT-H (0.40 m, 2.0 MA)	< 1.00	----
ZT-H (0.25 m, 2.0 MA)		< 5.00
RFX	< 2.50	----
CCTX	?	?
Starfire	0.90	----
RFPR	0.68	----
SPH (PPPL)	?	?
CRFPR	----	5.00
OHTE	----	5.00
Riggatron	----	20 - 50

2. Wall Conditioning

The wall conditioning techniques for the stellarator/torsatron/heliotron (S/T/H), reverse field pinch (RFP), OHTE, and spheromaks operated in the low power density (LPD) mode should be the same as for tokamaks. The high power density (HPD) systems will have to utilize large surface areas of graphite (coated or uncoated) or copper first walls due to the higher heat fluxes. The use of uncoated graphite may necessitate a completely different type of conditioning than stainless steel, Inconel, or copper. If HHF coatings are used, a method for in situ replacement of the coating will have to be developed for future devices. In order for ZT-40M, HBTX1A and OHTE to operate routinely at their design current levels passively cooled versions of advanced first wall components will have to be incorporated into the devices; therefore, the methods for conditioning the wall components will have to be addressed in the current machines. No new techniques should be required for "flagship" machines or "reactor" level devices unless unforeseen changes in first-wall materials are necessary for long pulse or continuous operation. Wall conditioning techniques that are under investigation are the same as for tokamaks and include

1. Gettering (Ti and Cr)
2. HHF coatings (SiC, TiC, W, Mo, etc.)
3. Glow Discharge Cleaning (GDC)
4. Taylor Discharge Cleaning (TDC)
5. Vacuum Baking $\leq$ 200°C

3. Limiter/Divertor/Impurity Control[1,2]

The operational chacteristic of limiters and divertors are well known for tokamaks, however, it should be stressed that even with tokamaks, more consideration should be given to the distortion of the axisymmetry that results from the use of some limiter configurations.[1] The current profiles in the scrape-off layer can become strongly altered from the normal equilibrium profiles. The resulting change in the axisymmetric equilibrium conditions may result in a change in the plasma confinement properties.

Very little experimental or theoretical work has been done with limiter or divertors in the AFCs. As the duration of the experiments gets longer, the issues of plasma wall interactions and impurity control will become much more important. These issues may be responsible for the current difficulties in the devices operating with high thermal wall loadings (< 1 MW/m^2) such as HBTX-IA, OHTE, CTX, and ZT-40M.

Current RFP experiments have begun to use limiters in an effort to protect the vacuum liner from the high heat loads. This work is in its infancy and extensive theoretical as well as experimental work will have to be actively pursued if the current generation of devices are to operate routinely at their design current levels. The limiter systems will have to be designed so that they do not introduce plasma equilibrium or stability problems, and they will have to tolerate the high heat loads (< 100 MW/m^2) for short pulses (~ 25 ms) without introducing impurities into the plasma system. Injection of gas into the boundary layer will be utlized in ZT-40M to study the effect on the plasma parameters in the edge plasma region. Either pump limiters or divertors will have to be developed for use on the longer pulse devices (ZT-H and RFX). Again, an expertise will have to be developed to examine the impact of limiter/divertor systems on RFPs. Modeling of the edge plasma and evaluation of the stability of the plasma system are necessary elements for the appropriate design of these systems.

HPD operation will place even more stringent requirements on the limiter/divertor systems. A larger fraction of the wall will be involved in the interaction and very tight control of the plasma equilibrium and edge plasma parameters will be necessary. Even though the stresses will be higher for HPD operation, the solutions should logically result from extensions of technologies learned at the lower stress levels.

The limiter/divertor/impurity control issues for the OHTE device are very similar to those of the RFP discussed above. In the near term, the interaction of the plasma with the first wall in the area of the magnetic divertors may have to be controlled. Divertor chambers capable of handling the power and particle loads will have to be incorporated into future devices.

The RFP and OHTE HPD options require extensions of technology by factors of $\sim \times 5$ from conditions in the LPD options; however, the Riggatron requires extensions of approximately another factor of $\sim \times 5$. At the present time it is difficult to perceive the solutions to impurity control and thermal stresses for heat fluxes of this magnitude.

The three-dimensional helical character of the S/T/H presents added difficulties for the engineering of divertors or limiters which adapt to this helical symmetry. If a HPD option is identified, the problems would be similar to those for the OHTE device discussed above; otherwise, the problems should be similar to the tokamak.

Both the speromak and the FRC have natural magnetic divertors. The PMI issues will be the same as the mainline program where similar fusion power outputs produce heat flux problems similar to those which occur in the divertor chamber of a tokamak or the end cell of a mirror machine. HPD options will have correspondingly more heat flux and the divertor chamber will have to be designed accordingly. The spheromak approach[3] utilizing electrodes for injection of magnetic helicity will have to develop or identify the technology necessary to prevent the injection of impurities from the electrodes.

4. Auxiliary Components (RF, NBI, etc.)

The need for the development of NBI and RF components for plasma heating and current drive is basically the same for some of the AFCs [S/T/H, EBT/NBT, CT (some concepts)] as for the mainline programs. Current drive for the RFP (ZT-H) via F-θ pumping will use low frequency ($\sim$ 1 kHz) components which should not require the development of new technology. A careful assessment of the critical features and parameters of the conducting shell for RFPs (ZT-H) will be of major importance with respect to design of the other PMI systems. The optimum method of plasma production for the CT concept will have to be identified and developed (CCTX).

5. Vacuum Systems

The LPD options of the AFCs have the same requirements for vacuum systems as the tokamak program. With tokamaks as with the AFCs, the use of large amounts of rf power will require careful shielding of the vacuum system components from the rf energy.

The total vacuum and/or divertor pumping speeds will have similar requirements of both the HPD option and the LPD systems; therefore, the vacuum ducting may be a more dominant feature relative to the FPC size for the HPD than for the LPD systems (including tokamaks and mirrors). This requirement, along with the requirement for a larger (fractional) volume for the primary coolant ducting, may generate a difficult "real estate" problem in the vicinity of the FPC for some concepts. Those approaches that place the FPC, or a portion thereof, within a vacuum envelope alleviate this concern somewhat.

6. Fueling

Pellet refueling requirements for both the AFCs and the mainline programs are similar, as are the requirements for LPD and HPD systems. A pellet ablation scaling law that agress with experiment[4] indicates that the pellet lifetime is weakly dependent on average plasma density ($\propto 1/n^{1/3}$). The decrease in plasma radius for HPD systems more than compensates for the higher plasma density and results in similar or less stringent requirements on pellet velocity. Systems with similar energy outputs will require similar fueling rates; therefore, the pellet injection frequency should be the same as for the conventional systems.

7. Plasma Equlibrium

The design of a limiter/divertor system should insure that the currents induced in the divertor/limiter structure or the modification of the plasma currents terminated by the structures do not generate problems with the global equilibrium or introduce toroidal asymmetries which interfere with the confinement. These effects have been considered theoretically for tokamaks in certain circumstances;[1,2] however, very little experimental work has been done at reactor level beta (or plasma pressure) conditions. Due to the lower level of effort for AFCs, even less consideration has been given to these problems.

An extensive effort to combine self-consistent models for the scrapeoff layer with plasma equilibrium and plasma stability codes should be made. The effect of field errors due to gaps in the conducting shell or discrete winding configurations should be included. This is inherently a 3-D problem where departures from axisymmetry are accurately determined and modeled. Care must be taken experimentally to excite gross plasma modes which are stabilized by the magnetic configurations or can be stabilized by feedback systems. Again, there is a considerable level of effort involved in the implementation of these considerations for the AFCs, and the manpower may not be available.

HPD operation will place even more stringent requirements on the interaction of the limter/divertor systems with the plasma system. Very tight control of the plasma equilibrium and edge plasma parameters will be necessary to prevent localized heat fluxes which exceed the design conditions.

8. Disruptions

The AFCs (EBT/NBT, S/T/H) which operate in a "currentless" mode are not expected to experience disruptions. The RFP, OHTE, High Field Tokamak and the CT schemes all have toroidal currents and can experience "current terminations." The collapse of the field structures in these devices dissipates the field energy and terminates the current flow in the plasma. The current experiments do not experience localized damage of the type generated in tokamak disruptions; however, these devices do not have confinement times of the same order of magnitude as the tokamak. Once the disruption mechanisms are identified and understood for tokamaks the potential for similar behavior in the AFCs can be assessed. The differences in the field structures (relative to the tokamak) may preclude the occurrence of the "disruption" phenomena.

All operating RFPs (ZT-40M, HBTX-1A, and OHTE) experience an abrupt end to the discharge where the plasma current decreases rapidly to zero in a few hundred microseconds. Accompanying the termination of the current is a positive pulse in the toroidal voltage at the liner, indicating that poloidal flux is entering the liner from the external circuit. A single-turn voltage in the range of 0.5 to 1.5 kV occurs for plasma currents in the range of 80 - 200 kA in ZT-40M.[5] No clear indication of any voltage scaling with current exists because some low current discharges can have high voltage spikes. The physics of MHD oscillations that occur in ZT-40M just preceding the current termination and their connection to the termination are being investigated. ZT H and RFX with their bigger physical size, lower field errors and better confinement times will be very important in assessing "disruption" type phenomena in RFPs.

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Part B

Technical Assessment of the Critical Issues and Problem Areas in the Plasma-Materials Interaction Field:

Edge Plasma Physics Theory and Modelling

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INTRODUCTION

Ideally the theorist and experimentalist work together to design experiments and to interpret their results. As part of the theorist-experimentalist dialog in the study of plasma-materials interaction, we present here a survey of this field from the theorist/modeler's point of view.

Figures 1-5 present in graphical form the topics that are covered in this survey. Each section contains a description of a physical problem, recent theoretical work, plans for future work, and needs for experimental data. We note that theoretical progress is unpredictable, and it is also hard to know what the important theoretical questions will be five years in the future. Thus most of our discussion is confined to the near-term (0 - 3 years).

While this survey of physics topics is incomplete (having neglected, for example, such topics as neutron loading, getter physics, tritium wall inventory, and wall effects on polarization), we can identify from it the most important immediate needs for modeling. They include:

(1) The development of a combined edge plasma and neutral transport model, including the physical processes required to analyse present experimental results,

(2) The development of an impurity transport model, ideally coupled to the plasma transport model, containing realistic impurity source models,

(3) The development of a time-dependent model of the evolution of the device walls and limiters, including material redeposition and the structure of the reformed wall, and including such topics as the wall tritium inventory, and

(4) modeling of solutions to the particle and power control problems expected in future devices.

A data base describing the basic physical processes is necessary for the development of these new theoretical and applied models. In particular, theory and modelling require data for:

(1) Reflection coefficients and reflected particle energy and angular distributions at low incident energies (1 to 100 eV), on single and multi-elemental surfaces,

(2) Hydrogen trapping, diffusion, and re-emission rates from surfaces,

(3) Sputtering yields, distributions, and physical properties of original and redeposited surfaces, for elemental and compound materials,

(4) Radiation rates for electron impact on atoms and H_2 at low (1-100 eV) temperatures,

(5) Atomic rate coefficients at low temperatures, and

(6) Data from comprehensive edge diagnostics on present and future devices, so that global transport models can be verified.

I. THE EDGE PLASMA - INTRODUCTION

Modeling of edge plasmas in general and of divertor plasmas in particular has made substantial contributions to our understanding of edge plasmas. These models range from simple sheath models attached to one-dimensional transport codes to sophisticated models which include a three-dimensional calculation of the neutral gas transport with a large variety of atomic processes coupled to a two-dimensional calculation in realistic geometries of the plasma transport both along and across the field lines. At present, such codes give a fair description of the observed plasma parameters in divertor experiments. These models have pointed to the existence of a cool dense plasma in the divertor due to intense, localized recycling of the neutral gas in the divertor chamber. Such an operating regime offers the promise of impurity and particle control by minimizing the erosion because of the low temperature, and providing high gas throughputs due to the high plasma and neutral density.

Future efforts will be aimed at applying and improving these codes and models. The major deficiencies in the plasma transport models are:

(1) A knowledge of cross field transport processes,

(2) Lack of impurity treatments, and

(3) Inadequate inclusion of transport from the main plasma.

Efforts are underway to improve the calculations to include these effects. A high priority is to benchmark the calculations against experiments such as PDX, ASDEX, and D-III, for which data exists. Only in this way, can we have some measure of assurance that all or most of the important physical processes have been included in the models. The lack of an impurity treatment is a serious deficiency. A serious attempt to put impurities requires that a set of self-consistent 2-D impurity transport equations be formulated, and that the necessary impurity recombination, ionization, and excitation rates be available. The issue of including the main plasma can be dealt with by enlarging the scope of the calculation.

I.1. COMBINED EDGE-MAIN DISCHARGE MODELING

Several laboratories are beginning to extend computational models of plasma transport to two spatial dimensions. These models have an ambitious goal: the prediction of wall erosion and of plasma fuelling and contamination in tokamaks such as INTOR, TFTR-upgrade, and ASDEX-upgrade. The computational tools and atomic physics in these models are sophisticated - alternating direction implicit (ADI) methods are convergent, and multigrid schemes are under investigation; elaborate Monte-Carlo calculations of neutral transport have been interfaced with plasma transport codes; extensive lists of rate coefficients and sputtering coefficients are available, often in a computationally convenient form.

Nevertheless, there are a number of critical issues to derive a comprehensive understanding and predictive modeling capability for plasma-wall interactions. Of particular importance are two-dimensional transport of plasmas with impurities, interaction of low energy particles with walls, and interaction of neutral atoms and molecules with ionized and neutral particles.

Here we outline some of the steps necessary to improve treatment of the plasma transport problem. First, we review the neoclassical theory of 2-d collision-dominated plasma transport, and we suggest how it can be extended to account for anomalous transport along and across magnetic field lines. We also describe the straightforward extension of this theory to include various elements and isotopes in a multispecies plasma. Then we review problems with boundary conditions usually used for divertor simulations, and we suggest some improvements.

I.1.A. Plasma flows

In a sufficiently collision-dominated plasma, the ion and electron temperatures are all almost exactly equal. Under such conditions, the dominant transport of energy "radially," across magnetic flux surfaces, is classical rather than neoclassical [C. E. Singer and W. D. Langer, Phys. Rev. A 28, 994 (1983)]. The dominant radial particle transport is, on the other hand, neoclassical. That is, the $\vec{u}_{\psi a} \times \vec{B}_\theta$ Lorentz term in the toroidal momentum equation is balanced primarily by the toroidal component of the frictional force on each plasma species (where $\vec{u}_{\psi a}$ is the radial flow velocity of species a and $\vec{B}_\theta$ is the poloidal magnetic field). The dominant contribution to this toroidal component of the parallel friction is $(B_\zeta/B)\, F_a^{\shortparallel}$, where B_ζ is the toroidal magnetic field and $F_a^{\shortparallel} = \vec{F}_a \cdot \vec{B}/B$ is the total parallel friction force on each particle species. When an expression for the parallel coulomb friction is available [c.f. Hirshman, S.P., "Transport of a multi-species plasmna in the Pfirsch-Schluter regime," Phys. Fluids 20, 589 (1977)], generalization of the particle balance to a multispecies plasma is straightforward [C. E. Singer, "Towards a Predictive Model of Plasma Wall Interaction," Princeton Plasma Physics Laboratory, Applied Physics Division Report No. 23b, (October 1983)].

Anomalous transport processes are to be expected whenever collisions are insufficient to maintain a nearly Maxwellian plasma. In the theory outlined above, this occurs when transport across magnetic field surfaces in no longer sufficient to maintain a radial pressure scale height $L_\psi = p/[(\partial p/\partial\psi)|\vec{\nabla}\psi|]$ greater than a typical gyroradius, ρ. Presumably, instabilites arise to flatten the pressure gradient, for example, by enhancing the radial transport of ion energy. This can be accomplished phenomenologically by increasing the ion-ion collisionlity to

$$\nu_{ab} \rightarrow \alpha_{ab}\nu_{ab} ,$$

where, for example

$$\alpha_{ab} = \frac{1}{1 - \exp(-\alpha\rho/L_\psi)} .$$

To match typical experimental results, it should be necessary to take the Larmor radius multiplier $\alpha \lesssim 10^2$ if ρ is taken to be the toroidal gyroradius and $\alpha \lesssim 10^1$ if ρ is taken to be the poloidal gyroradius. (While the approximate choice for ρ may itself depend on the collisionality, either of these combinations for α and ρ will probably suffice for initial modeling studies.) An appropriate choice for p in the above formula for L_ψ is the total ion pressure $p_- = \sum_a n_a T$, but using the electron pressure $p_e = n_e T$ is essentially equivalent. A combination of this transport enhancement with neoclassical effects gives the axisymmetric scrapeoff scale height scaling shown in Fig. 6. There may be additional effects at very low collisionality, as indicated by the dashed lines in Fig. 6 and explained in the next two paragraphs.

At lower collisionality, the electron thermal conductivity parallel to the magnetic field in the above formula must be modified. The classical kinetics of parallel electron thermal conductivity have been recently worked out [P. J. Clause and R. Balescu, Plasma Physics 24, 1429 (1982)]. Clause and Balescu give for the heat conduction flux results which accurately treat departures from the generalized Braginskii formula quoted above, and which gives a reasonable extrapolation into the "heat-flux-limited" collisionless regime. With an appropriate empirically adjustable multiplier on the collisionless heat flux limit, obvious errors in the parallel heat transport can be avoided by incorporating these results in plasma transport codes.

How far the result of Clause and Balescu can be extended toward the collisionless regime depends on whether the electron distribution function remains stable upon approaching the collisionless regime. For the one-dimensional transport problem analyzed in most present scrape-off plasma transport codes, the stability analysis has been performed only on the linearized "Spitzer-Härm" distorted Maxwellian distribution function analyzed by Braginskii. For the approach to the collisionless regime is characterized by so-called heat flux instabilities. These instabilities can be driven either by the "heat-flux" drift of energetic electrons away from regions of high temperature, or by the drift of bulk electrons back into the region of high electrostatic potential which results from the heat-flux drift. In either case the marginally unstable modes transfer energy to the ions, generally after propagating transverse to the magnetic field. Thus, in the linear instability regime, reduction of parallel heat conduction is accompanied by an enhancement of radial energy transport.

Considerable theoretical and experimental effort is required to determine whether the processes described above are adequate to explain plain plasma transport in regions near material boundaries. For the collision-dominated edge plasmas of most interest for advanced tokamaks, the theory can be worked out in full, and this is a task of high priority. There is an enormous and largely untapped potential for application of spectroscopic diagnostics to test the predictions of such theories. For the less collisional plasmas often found in existing devices, considerable kinetic and turbulence theory is required to gain insight into possible transport mechanisms. Measurements of plasma turbulence need to be extended to monitor the expected transition from collisionless-dominated behavior.

I.1.B. The plasma-wall boundary

First, we discuss the boundary conditions on the poloidal flows and then the boundary conditions on the radial flows. For a pure hydrogen plasma, six boundary conditions are required on the parallel flows. For a completely determined solution, a convenient set of boundary conditions is the flow velocity and heat flux at the two poloidal end plates bounding a magnetic surface, and the applied potential difference and current drama between these plates. (Note that the transport equations do not give up-down symmetric solutions even in an up-down symmetric external geometry, so midplane symmetry boundary conditions are inappropriate.) An appropriate choice of values for these boundary conditions is sonic flow with energy flux $q_{\parallel} \approx 8nTu_{\parallel}^{"}$ [G. D. Hobbs and J. A. Wesson, "Heat Transmission Through a Langmuir Sheath in the Presence of Electron Emission," Culham Laboratory Report No. CLM-R61 (1966) UKAEA] no net current drawn from or to the plates, and a potential difference determined by collisionless sheath theory from the plasma "temperatures" at the plates. All of these conventional assumptions rely on a kinetic theory which does not self-consistently include the effect of collisions; and reflection and emission of charged particles from the plates is often also neglected.

In an attempt to construct or predict the results of a quantitative kinetic boundary layer theory, the role of reflection and emission at the plates is important. For example, many practical plate materials yield secondary electron emission to a point where an increase of $q_{\parallel}$ by a factor of 4/3 is probably a better choice for general transport studies. A quantitative theory of this effect will require net effective electron reflection as a function of incident energy.

While reflection and emission of most ions from a plate are known to be negligible at incident energies on the order of one keV, the lower energy range where solid surface structure becomes significant (5-10 eV) for hydrogen isotopes) is relatively unexplored. For some materials, particularly when fully covered with adsorbates of hydrogen, oxygen, water, or hydrocarbons, reflection and emission of ions might conceivably be significant. This would have the effect of reducing the speed of flow to the plate, and would also affect the approach to choosing realistic reflection coefficients for the kinetic boundary problem. Of course, direct reflection of incident ions as neutrals also has an important impact on calculation of plasma sources. The potential difference and current between the plates will also depend on the kinetic details of reflection and emission at the plates.

For multispecies plasmas, proper choice of poloidal boundary conditions is particularly problematic. One-dimensional fluid calculations suggest that a balance between "thermal forces" and ion-ion friction can lead to strong poloidal segregation of different plasma species [J. Neuhauser et al., "*Modeling of the Impurity Flow in theTokamak Scrapeoff*," Max Planck Institut fur Plasmaphysik Report. No. IPP-1/216 (1983)]. However, ion-ion friction effects become comparable to thermal effects only at the transition from collisional to collisionless ion behavior. Thus, a confirmation of the implications of these interesting calculations will also require a kinetic treatment. An arbitrary choice of boundary conditions consistent with the above collision-dominated theory unfortunately does not include such effects, so exploration of a range of impurity flows to the boundary may be necessary. For closed flux surfaces, these boundary conditions have to be replaced by appropriate poloidal continuity conditions.

For a pure hydrogen plasma, the one-fluid transport equations also require six radial boundary conditions. For example, the density, temperature, and poloidal or parallel ion flow velocities can be specified on given internal and external flux surfaces. The external flux surface can generally be placed in a region where interaction with neutrals allows setting boundary parameters to convenient, negligible values. The internal flux surface is more problematic. Ideally, the poloidal distribution of density, temperature, and flow velocity can be measured experimentally in a region of high collisionality. Often, however, the inner boundary condition must be assigned theoretically. For a low-β, low aspect ratio tokamak, the neoclassical particle and energy fluxes are proportional to $\cos\theta$, being outward at the largest major radius and inward at the smallest major radius. Various types of experimental evidence suggest that anomalous plasma outflux is similarly concentrated on the larger major radius side of the plasma column, but this is not certain. An anomalously enhanced transport with poloidal variation similar to neoclassical will probably suffice.

For radial impurity transport, the choice of inner boundary conditions is almost as uncertain as for parallel boundary conditions. Recent experiments on the "predisruption state" in ASDEX are compatible with a similar poloidal variation to that noted above.

Finally, additional difficulties with boundary conditions inevitably arise where divertor or limiter plates are not normal to magnetic surfaces. A kinetic treatment is again required for resolution of this problem. Toroidal asymmetries introduce further complications both in interpretating the data and to extrapolating results to configurations which could handle a large power load. It is likely that large toroidal asymmetries will and should be avoided in design of most future tokamak experiments. For non axisymmetric plasma devices, the theory is complicated and progress will be relatively slow.

I.2. SCRAPEOFF POTENTIALS

The plasma potential in the scrapeoff consists of two related potentials: the sheath potential at the limiter surface and the radial potential in the plasma.

I.2.A. Sheath potential

The limiter potential is usually calculated by requiring that the electron and ion charge fluxes flowing along field lines to each point on the limiter surface are equal (local ambipolarity). The resulting potential is approximately three T_e which means that each field line is at a different potential and, therefore, a radial electric field exists in the scrapeoff. It is equally plausible to assume that a constant potential (independent of radius) exists. In this case, the electron flux will exceed the ion flux for field lines near the limiter tip while the ion flux will exceed the electron flux for field lines which lie deeper in the scrapeoff. The potential assumes a value which assumes that the net charge flux to the entire limiter is zero (global ambipolarity). The resulting local charge imbalances are removed by current flowing within the limiter, i.e. electrons flow from the limiter tip region to the region of excess ions near the wall. For global ambipolarity, no radial electric field is present in the scrapeoff plasma.

The theoretical problem is to determine to what extent the flow to the limiter obeys load or global ambipolarity. It is feasible that an intermediate condition exists where the potential is not constant but the flow is not locally ambipolar. There is some experimental evidence that such an intermediate situation has been observed [Strawitch, C., and Emmert, G., Nucl. Fusion 21, 1291 (1981)].

The potential which results with the global ambipolarity assumption is less than the peak potential obtained from local ambipolarity. The lower potential results in a lowering of the energy gained by plasma ions in the sheath and a resultant lowering of the sputtering rate of the limiter surface. Thus, improved understanding of whether the limiter is locally or globally ambipolar is important for estimating limiter lifetimes.

The constant potential which is obtained from global ambipolarity results in a flatter $T_e(r)$ profile in the scrapeoff layer. Measured scrapeoff T_e profiles are usually flatter than the profiles predicted using local ambipolarity. Global ambipolarity may improve the agreement of the model T_e with measured values.

I.2.B. Radial potential

A group from RPI is presently measuring the radial dependence of the plasma potential with respect to the vacuum vessel wall in ISXB. The potential drop in the scrapeoff must be known for a proper interpretation of these measurements. The radial field in the scrapeoff is probably determined by the requirements of ambipolar flow both along (see above) and across field lines as well as by the radial force balance of the ions. An understanding of the physics of the radial field is also necessary for deciding whether the parallel flow to the limiter is globally or locally ambipolar.

I.3. ERGODIC SCRAPEOFFS

The scrapeoff of the torsatron ATF, to be built at ORNL, will have an ergodic scrapeoff. Since the magnetic field in the scrapeoff does not form closed flux surfaces, new methods for treating the flux of thermal plasma parallel and perpendicular to the magnetic field will need to be developed. Monte Carlo techniques for plasma transport may be necessary. In addition, the toroidally and poloidally asymmetric flux surface configuration to the main plasma wall requires a Monte Carlo 3D neutral treatment to realistically calculate the neutral flux distribution on the wall. The wall diffusion model will also need to be 3D, as discussed above for tokamaks. The fundamental wall physics will be the same as for tokamaks, however. Development of these models into a complete plasma and neutral transport model will involve substantial theoretical effort and improved computer facilities.

I.4. DATA NEEDS

Experimental or theoretical knowlege of the plasma edge conditions such as density, temperature, and flow velocity, is the most important input for almost all aspects of plasma-material modeling.

Many transport codes exist which contain edge plasma models [J. Ogden et al., IEEE Trans. Plasma Science, PS-9, 274 (1981)]. What are missing from the modeling process are good sets of measured data. Even the best diagnosed machines (from an edge physics point of view) have Langmuir probes, for example, at only a handful of sites. The modeler then may have only one or two data points to match. What is needed is better coordination between the experimentalists and the modeler in order to best design the diagnostics (probes, pressure guages, bolometers, spectroscopic diagnostics, and so on) on each experiment.

II. HYDROGEN RECYCLING - INTRODUCTION

Plasma which diffuses from the confined region into the scrapeoff flows along field lines to a limiter or divertor where it is neutralized and reemitted as neutrals. The resulting flux on the limiter is large enough to saturate the limiter surface within about 10 ms after discharge initiation so that all the incident flux is immediately returned to the plasma as neutrals. Some of these neutrals are ionized while others charge exchange with plasma ions, creating energetic neutrals which may escape to the wall. Neutrals which are not reflected from the wall represent the major loss of particles from the discharge; this is the loss which is replaced by external fueling. The particles desposited in the wall may diffuse back to the surface, recombine and desorb as molecules. For a long discharge, the wall concentration and resultant desorbed flux increase so that the desorbed flux approaches he deposited flux and the overall recycle coefficient approaches unity. The characteristic time for wall saturation for a typical tokamak discharge is several seconds. There are several uncertainties in this chain of events which constitutes the recycle model.

II.1. THREE DIMENSIONAL DIFFUSION

The neutral flux from the plasma is incident on the wall adjacent to the regions where cold neutrals first enter the plasma, i.e. around the limiters and gas puff port. Thus, the wall neutral flux and resulting wall concentration are intrinsically asymmetric poloidally and toroidally. If toroidal and poloidal variations in the wall concentration profile are to be properly accounted for, a 3D wall diffusion model should be used. A 3D plasma neutral model would be needed to calculate the flux distribution on the wall and this model would need to be incorporated into a plasma transport code to follow the temporal evolution of the discharge. With this model, the earlier saturation of the regions of the wall nearest the limiters could be properly accounted for. Such a model would be particularly attractive for TFTR and JET since it would produce a complete accounting of the tritium inventory in the wall.

Three-dimensional neutral transport treatments do exist [D. Heifetz et al., J. Comput. Phys. 46, 309 (1982); J. T. Hogan, J. Nucl. Mat. 111-112, 413 (1982); D. Reiter and A. Nicolai, J. Nucl. Mat. 111-112, 434 (1982); P. Gierszewski, Ph.D. thesis, M.I.T.], however the computer time needed for repeated calls to the neutral codes may be prohibitive on present computers. All present codes use Monte Carlo algorithms which do not benefit significantly from vectoring computers such as the CRAY-I and CDC-205. Future code work should include whatever forms of optimization possible.

II.2. NEUTRAL-PLASMA PROCESSES

Important in the modeling of the transport of H^o, He^o, Ne^o, and higher Z impurities is a complete catalog of the most important neutral-plasma reactions, together with accurate experimental or, lacking that, theortical data for their cross-sections.

An elementary catalog of these reactions can be gathered using cross-sections found in a number of standard references [R. L. Freeman and E. M. Jones, CLM-R137, Culham Laboratory, Abingdon, Berkshire (1974); E. M. Jones,CLM-R175, Culham Laboratory, Abingdon, Berkshire (1977); K. L. Bell et al., CLM-R216, Culham Laboratory, Abingdon, Berkshire (1982); K. Takayanagi and H. Suzuki, IPPJ-DT-48, Nagoya University (198)]. These data are, however, deficient in a number of ways:

(1) Not all cross-sections are known over the entire energy range of 1 -

25,000 eV found in present and coming machines,

(2) Atomic excitation and molecular viabrational excitation models are lacking, and

(3) Molecular dissociation processes are not well documented in the low (1-50 eV) energy range.

Because of (1), we cannot know which are the most important neutral-plasma reactions. Excitation models are necessary for modeling H_α and other radiative emissions which are presently experimentally measured. Finally, molecular dissociation may be an important term in the energy balance in devices where high plasma recycling occur.

One recent work [R. Janev et al, "A Survey of Atomic Collision Processes Important at the Plasma Edge," Symp. on Energy Removal and Particle Control in Toroidal Devices," Princeton (1983)] is an extensive effort to collect and evaluate available data. Cross-sections are estimated where no experimental or theoretical data exist. Special emphasis is placed on the low energy range of 2-50 eV. This work, which is still in progress, will greatly improve the neutral-plasma reaction data base.

It will be necessary to periodically review and update this work as experimentalist improve and expand their data, and as the needs of the modelers evolve.

II.3. NEUTRAL-NEUTRAL PROCESSES

In modeling the neutral density distributions, neutral-neutral scattering must be taken into consideration where neutral pressures reach the few mTorr range.

At present, however, there is virtually no data on total cross-sections, σ_{total}, and $d\sigma/d\theta$ for H^o and D^o onto anything for energies above 2 eV and below 300 eV. Many of these quantities can be measured by the scattering experiments now in progress utilizing PLT as a neutral source and the Low Energy Neutral Spectrometer (LENS) as a detector. Scattering-geometry-independent σ_{total}'s (from experimentally determined V(r)) of 25 to 1800 eV D^o and He^o on He, Ne, Ar, and Kr, will soon be published [D. Ruzic, PhD thesis, Princeton U.]. Determination of σ_{total} for H^o, D^o, and He^o on H_2, D_2, CH_4, and other molecules can also be done with this experimental apparatus.

II.4. WALL REFLECTION MODELING

A fraction of the neutral flux emitted from the limiter is desorbed hydrogen molecules. In addition, molecules are desorbed from the wall and introduced externally by gas puffing. When these molecules dissociate, a flux of several eV neutral atoms is created and approximately half of these particles are immediately incident on the wall. A major uncertainty in the neutral recycling model is the reflection coefficient of these low energy neutrals.

Some theoretical and experimental work has recently been begun (by Baskes at SNL and Bohdansky at IPP, Garching, among others). Improved understanding of low energy reflection is needed to formulate a complete recycle model.

Further, in present models the wall is assumed to be a smooth uniform metal surface. In reality, the first several hundred angstroms of the wall consist of a deposited, heavily oxidized rough layer of tritium, carbon, and other materials which is dubbed "Tokomakium" by the surface physics community. The diffusion properties of this surface layer are poorly known and the enhanced chemical and physical trapping of hydrogen in this layer need to be considered.

II.5. FIRST WALL PARTICLE AND POWER LOADINGS

While the wall particle and power fluxes cause minor structural damage in present machines, the fluxes expected in future devices may cause significant erosion. Thus designers need to know the magnitude of these fluxes throughout the system.

The ion fluxes can be calculated using present transport codes. However, an analysis of the physics of the pre-sheath and sheath regions, such as in [R. Chodura, J. Nucl. Mats. 111-112, 420 (1982)], is necessary in order to predict the actual power loadings and wall erosion due to the impinging ions.

Neutral transport in an INTOR pump limiter design has been modeled [D. Heifetz et al, Jrnl. Nucl. Mats 111-112 (1982) 298], and the results show that the charge-exchange flux is confined mainly to the limiter face, while the load on the first wall is confined to the area behind the limiter tips. This is due to the fact that the mean-free path length for a neutral born at the limiter is 5-10 cm, while the limiter face is 1.5-2 m wide. Thus most of the first wall receives a negligible flux, however, the limiter may be subject to heavy erosion.

Thus future reactor designs must include an analysis of the effect on the walls of neutral particles, done much in the same spirit as neutronics calculations.

III. IMPURITY TRANSPORT - INTRODUCTION

Impurity transport along field lines is an important new topic for the modeler. The impurity screening efficiency of divertors for wall generated impurities, and the ability of a divertor to both not generate impurities and keep the ones that are generated confined to the divertor chamber are clearly important topics for study which will achieve attention in the future. Extended calculation of impurity transport requires the formulation of the problem, the atomic physics rates, and good sputtering and desorption rates. Reasonable codes for these studies are likely to be constructed in the next two years.

III.1. WALL MATERIAL EROSION AND REDEPOSITION

The surface of an impurity control device in a fusion device can be subject to large rates of erosion and redeposition [J. N. Brooks, J. Nucl. Technol./Fusion 4, 33 (1983)]. These rates may vary from several cm/yr to ~ 100 cm/yr depending on the plasma edge conditions. Redeposition arises when sputtered neutrals are ionized in the scrape-off zone, or the plasma proper, and return to the surface along field lines. The redeposited material can result in substantial self-sputtering. Both limiters and divertors in tokamaks, and halo scrapers in mirrors, are subject to the erosion/redeposition process. For low-Z surface coatings redeposition tends to decrease the net surface erosion and decrease the plasma contamination. The effect is the same for nonlow-Z coatings ($Z \gtrsim 13$) at low plasma edge temperature ($T_e \lesssim 50$ eV). At higher edge temperatures ($T_e \gtrsim 50$ eV) redeposition may result in an unstable self-sputtering cascade, according to predictions. The lifetime and performance of impurity control systems thus depends critically on the erosion/redeposition process, for most plasma regimes.

The behavior of surfaces subject to large amounts of erosion and redeposition is uncertain. For lack of data to the contrary, current calculations treat the properties of the redeposited material as being the same as the original material. This is probably a better assumption for metals than for nonmetals and/or compounds. Experimental data are urgently needed in any case, in order to design impurity control systems with the proper confidence level. The key properties of interest, for redeposited material, are as follows: sputtering coefficients (deuterium, tritium, helium, and self-sputtering), adhesion strength, stress buildup and possible cracking and flaking, chemical composition, morphology and surface topology, and physical properties (e.g., thermal conductivity, electrical resistivity).

Experiments on redeposited material could take these complementary forms: (1) laboratory simulation experiments with controllable, CW beams (H^+, He^{++}, $Z^{(+N)}$, e^-) irradiating carefully prepared surfaces; (2) experiments using a CW plasma, e.g., from a mirror discharge, to simulate fusion reactor edge conditions; and (3) actual in-situ tokamak experiments. The latter approach obviously provides the most realistic conditions but is limited in total fluence, for present and near-term devices.

In addition to materials properties, more work is needed in predicting erosion/redeposition phenomena in general, and in characterizing specific plasma/surface conditions. These include the sheath region, angles of incidence, energy and angles of sputtered particles, charge-exchange neutral data, etc. Continuing modeling efforts, together with characterization of these phenomena by experiments, would appear to be highly advisable.

III.2. MOLECULAR PRODUCTION AND TRANSPORT

Light ion impurities can enter a plasma from the wall or limiter as molecules as well as atoms. There are differences in the transport of molecular and atomic species due to their different reactions with the bulk of the plasma.

In a plasma-surface interaction, chemical reactions occur in which an incident atom or ion reacts with a surface atom to form a molecule. The presence of a carbon surface in a hydrogen plasma results in the formation of methane and acetylene. Furthermore, there is evidence that methane and water are produced on stainless steel and titanium surfaces when carbon and oxygen impurities strike them.

Preliminary studies of the penetration of methane [W. D. Langer, Nucl. Fusion 22, 751–761 (1982)] and water in [M. Tendler and O. Agren, Plasma Physics (1983) in press] into plasmas reveals that there are qualitative and quantitative differences between their transport and that of atomic carbon and oxygen. The most significant difference is that some fraction of the carbon and oxygen released from the breakup of their precursor molecules penetrates further into the plasma than if these light impurities have been released in atomic form.

A Monte Carlo, two-dimensional treatment of the transport and breakup of molecular impurities should be developed in the near future. Data needs for such a model include:

(1) A compilation of data for the interaction of molecular impurities (in atomic and ionic form) and the bulk of the plasma,

(2) Transport coefficients for these and the atomic species in the plasma, and

(3) Data on the production and type of molecular impurities produced at the wall.

IV. PUMPING SYSTEMS

Various pumping systems have been used in order to control mass during a discharge. Gettering systems are included in many present machines; however, they are not practical in reactors since they do not pump He. Prototypical pump limiters have been used on PDX and ISX, and will be operating soon on TEXTOR and PLT. A belt pump limiter designed as a reactor prototype is planned for TEXTOR in 1985.

Modeling of the pumping performance of these devices is useful both for design and data analysis purposes. Some of this work has already been done [C. D. Boley et al., "Neutral Transport in the ALT-I Pump Limiter," Symposium on Energy Removal and Particle Control in Fusion Devices, Princeton (1983); R. Conn et al., "An Advanced Pump Limiter Experiment of Large Toroidal Extent-ALT-II," Center for Plasma Physics and Fusion Engineering, University of California, Los Angeles, CA (1983); R. Budny et al., "Results From the Scoop Limiter Experiment in PDX," Symposium on Energy Removal and Particle Control in Fusion Devices, Princeton (1983)].

Two large unknowns in predicting the pumping performance of large scale pumping systems are first the effects of plasma recycling on the neutral pressure and on the plasma inflow rate, and second the difference between hot atomic and cold molecular conductance through pumping ducts.

According to [M. Petravic et al., Phys. Rev. Letts 48, 326 (1982)], we may expect a region of high-plasma density and low-plasma temperature near the neutralizer plate in a pump limiter for an INTOR-size machine. One consequence is that the neutral pressure would be enhanced at the plate. Since the plate would typically be close to a pumping duct opening, the pumping rate would be higher than it would be in the absence of the high recycling region. However, this model could not predict the effect of this recycling on the plasma edge. It is not known whether the entire edge would be so cooled as to choke off the flow into the pump limiter. Thus a self-consistent model of the main discharge plus pump limiter is necessary to evaluate the pump limiter's pumping capacity.

Some calculations using wall reflection models which include dependence on incident angle and energy, have shown an energy dependence in the transmission probability of an atom through a duct [R. Conn et al, ibid], due to the specular reflection of energetic atoms incident on walls at grazing angles. If this effect can be experimentally shown to exist then it could be exploited in future designs.

A separate issue in pump design is the difference in pumping of hydrogen versus such impurities as He. Ideally, relatively more impurities would be removed from the system than the DT fuel. Whether this is possible, due perhaps to differences in atomic physics or wall reflection behavior, is largely unknown. Some theoretical work has been done in the past, but should be re-done as more complete plasma/neutral transprot models become available.

V. PLASMA-MATERIALS INTERACTION IN TANDEM MIRRORS

The edge plasma boundary in a tandem mirror device encompass the end-loss plasma, the cold plasma at the end wall, and the halo plasma which extends throughout the length of the plasma. The end-loss plasma are the ions and electrons that leaked out of the endplugging magnetic and electrostatic bottle. These particles has a distribution that is highly non-Maxwellian with energies (especially the ions) of several keV. The cold plasma at the end wall is created from ionization of the background gas by the end-loss plasma. The cold ions does not have enough energy to overcome the electrostatic endplugging potential and therefore reside in the end tank region. The cold electrons, however, are accelerated by the same potential to the center cell region. This effect can lead to a substantial increase in axial power loss. The halo plasma flowing along the near-axial field lines in a mirror device has a natural divertor pumping action that shields the core plasma from impurities and gas influx. The formation of a dense halo, maintained by heating of the electron and by recycling the ions at the end, is presently under study. The recycling is accomplished by a halo recycler, patterned after pump limiters, operated in a high recycling mode.

In order to assess the effects of the edge plasma on bulk plasma confinement we need to properly model each of the following areas:

(1) Cold end wall plasma. The interaction of the cold plasma with the end-loss plasma and the wall must be studied in detail. The important question here is the sheath which can cause a large power drain. The formation of the sheath must be treated self-consistently in view of the presence of two plasma components (Maxwellian cold plasma and non-Maxwellian energetic end-loss plasma) and secondary electron emission from the wall.

(2) Halo plasma particle and heat flow. A kinetic treatment of the axial flow and the recycling of the halo is needed to understand the particle and power fluxes. This analysis must treat the power flow from a semi-collisional region in the upstream, through a highly collisional region in the region of the recycler, and into a convective flow region at the sheath. The theory must also treat the radial power transport by diffusion, charge-exchange, and radiation and direct coupling to auxiliary heating sources.

(3) Impurities in the halo. A model must be created to study the production, the buildup, and the ultimate fate of impurities that are trapped in the halo. If the concentration of impurities becomes large in a long pulse device, then radiative energy loss from these impurities must be accounted for in the power balance of the halo plasma.

FIGURE CAPTIONS

FIG. 1. An overview of plasma-material interactions in a fusion device.

FIG. 2. Plasma flow and neutral particle transport in a divertor/pump limiter.

FIG. 3. Neutral particle recycling processes.

FIG. 4. Particle-wall processes.

FIGL 5. The plasma-wall transition region.

FIG. 6. Schematic of classical and gyroradius-limited radial scale heights for plasma pressure.

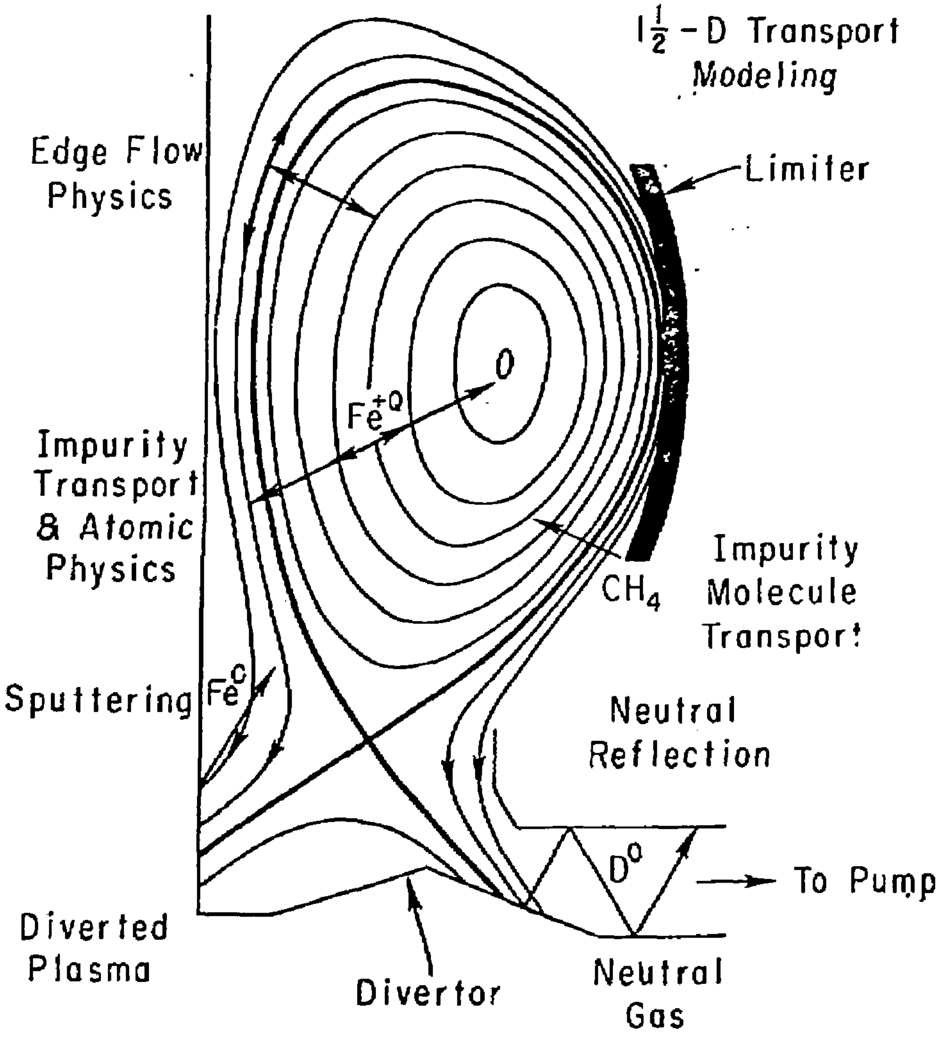

$1\frac{1}{2}$-D Transport
Modeling
Edge Flow
Physics
Limiter
0
Fe^{+Q}
Impurity
Transport
& Atomic
Physics
CH_4
Impurity
Molecule
Transport
Sputtering
Fe^0
Neutral
Reflection
D^0
To Pump
Diverted
Plasma
Divertor
Neutral
Gas

FIG. 1

#81P0177

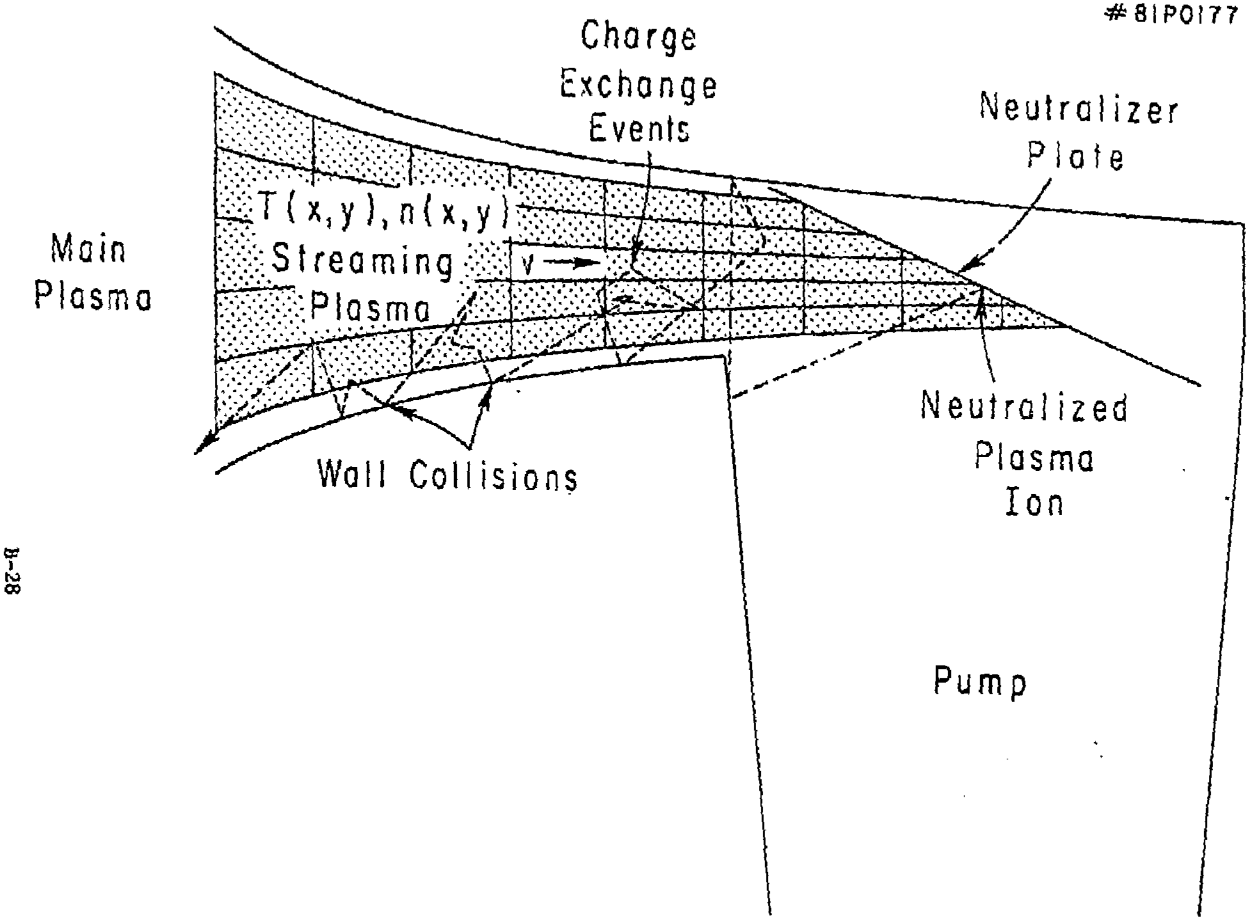
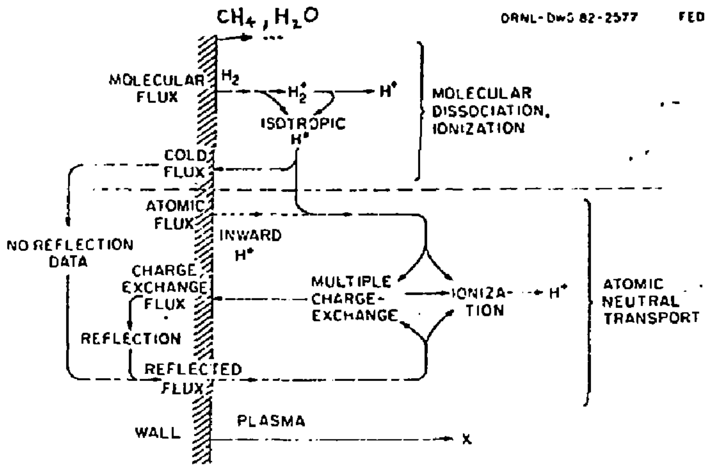


FIG. 2



CH4 , H2O
ORNL-DWG 82-2577 FED
MOLECULAR FLUX
H2
H2+
H+
ISOTROPIC H°
MOLECULAR DISSOCIATION, IONIZATION
COLD FLUX
ATOMIC FLUX
INWARD H°
NO REFLECTION DATA
CHARGE EXCHANGE FLUX
MULTIPLE CHARGE-EXCHANGE
IONIZA-TION
H+
ATOMIC NEUTRAL TRANSPORT
REFLECTION
REFLECTED FLUX
WALL
PLASMA
X

FIG. 3

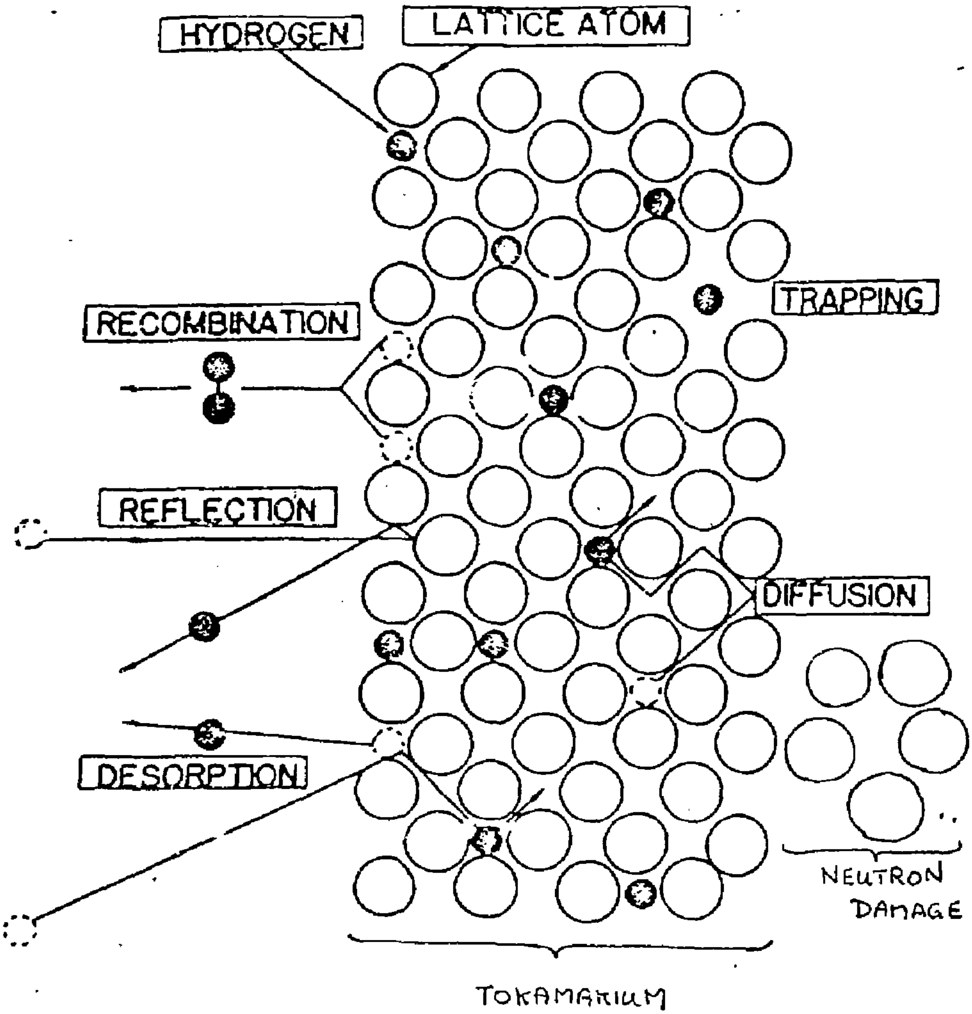

HYDROGEN
LATTICE ATOM
TRAPPING
RECOMBINATION
REFLECTION
DIFFUSION
DESORPTION
NEUTRON
DAMAGE
TOKAMAKIUM

FIG. 4

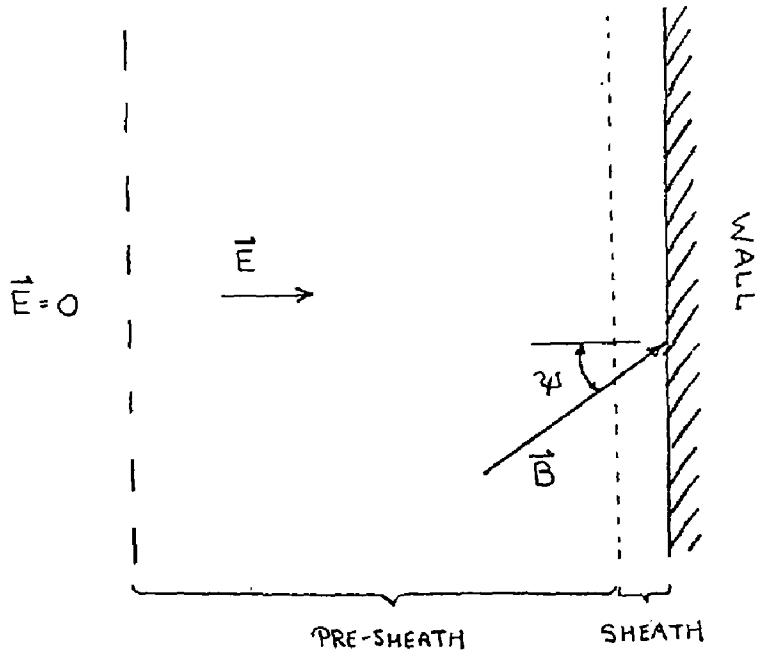
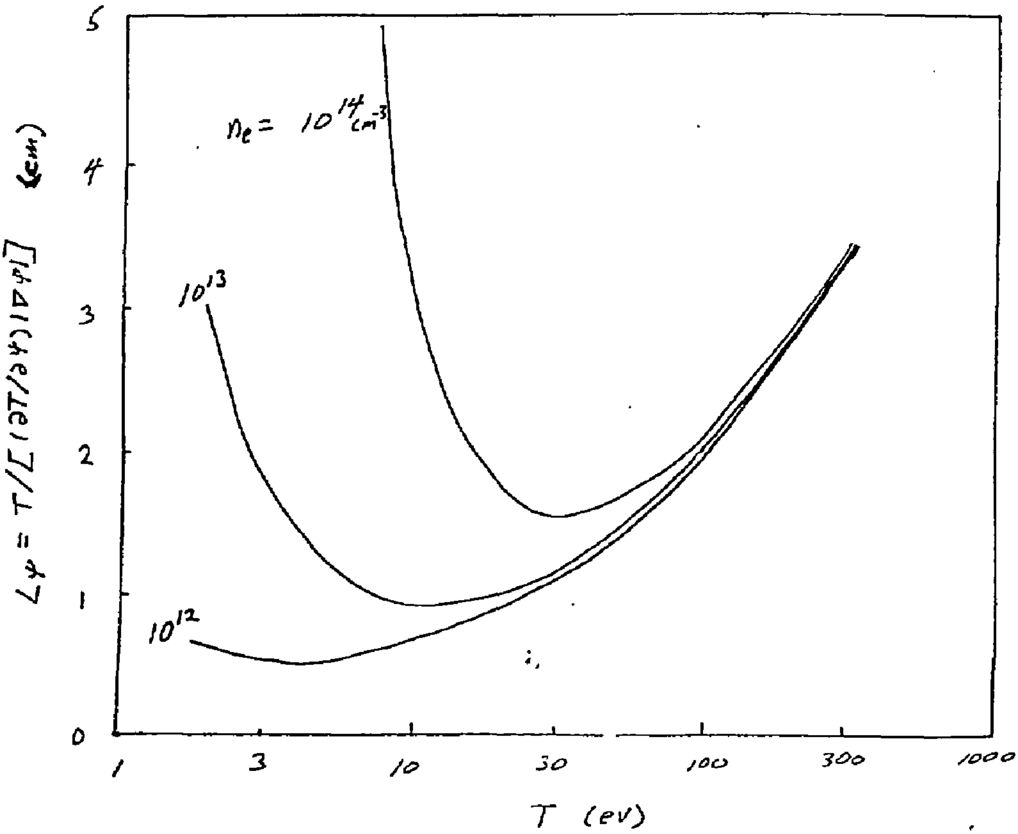


FIG. 5



$L_\psi = T/[\partial T/\partial \psi)|\nabla\psi|]$ (cm)
$n_e = 10^{14}$ cm^{-3}
10^{13}
10^{12}
5
4
3
2
1
0
1
3
10
30
100
300
1000
T (eV)

FIG. 6

Part C

Technical Assessment of Critical Issues and Problem Areas in the Plasma Materials Interactions Field:

Surface Physics

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I. INTRODUCTION

Surface physics issues that impact plasma-materials interactions can be conveniently divided into three main topics:

(1) Hydrogen recycling effects;

(2) Impurity introduction; and

(3) Vacuum and surface control.

These issues are, of course, interrelated and, in turn, depend on edge plasma physics, materials selection, and component design. The issues also change in importance, depending on the time frame. Since the area of surface physics effects in plasma materials interactions has been recently reviewed in great detail [1]; this section will only attempt to assess the critical problem areas for the magnetic fusion energy program.

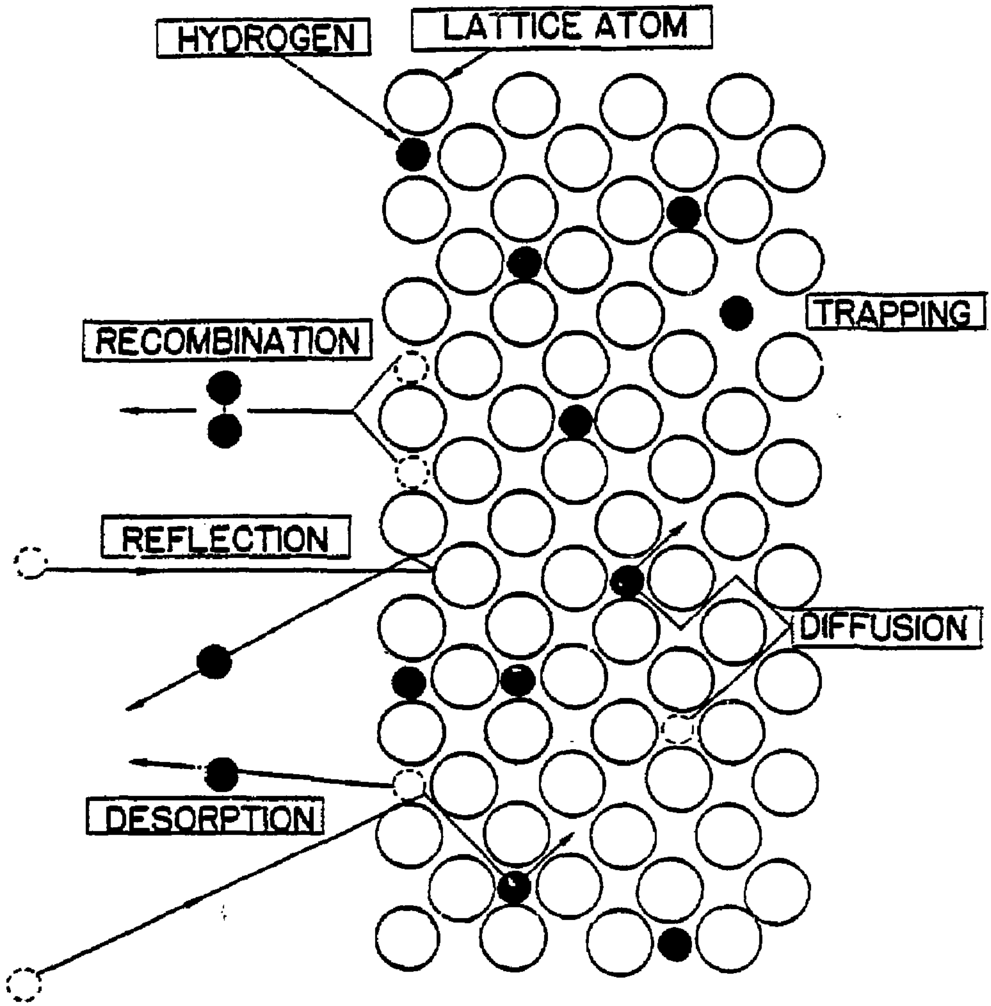
II. HYDROGEN RECYCLING EFFECTS

Hydrogen recycling effects refer to the repeated interchange of hydrogen fuel between the plasma and first wall of a magnetically confined plasma, as illustrated in Figure 1. Hydrogen ions escape from the plasma by diffusional or charge exchange processes, and bombard the first wall. Hydrogen subsequently leaves the wall as neutral atoms and molecules that are ionized in the plasma edge region. As demonstrated by hydrogen isotope exchange experiments, hydrogen recycling from the wall is a controlling source of plasma fueling in current tokamaks with shortburn cycles. In the next generation of magnetically confined plasma devices, such as TFTR and JET, a second major concern is wall tritium inventory. A third hydrogen concern is tritium permeation through the first wall of advanced D-T devices such as FED, INTOR, or DEMO. Excessive tritium permeation would require the implementation of permeation barriers or additional tritium recovery equipment.

Knowledge of the processes of hydrogen recycling and permeation is, therefore, important not only for the understanding of the operation of present day devices but for the optimization of future reactor designs as well. There are three main mechanisms for hydrogen recycling from first walls: (1) reflections; (2) photon, electron and ion desorption; and (3) hydrogen trapping, diffusion and molecular recombination.

FIGURE 1

RECYCLING MECHANISMS



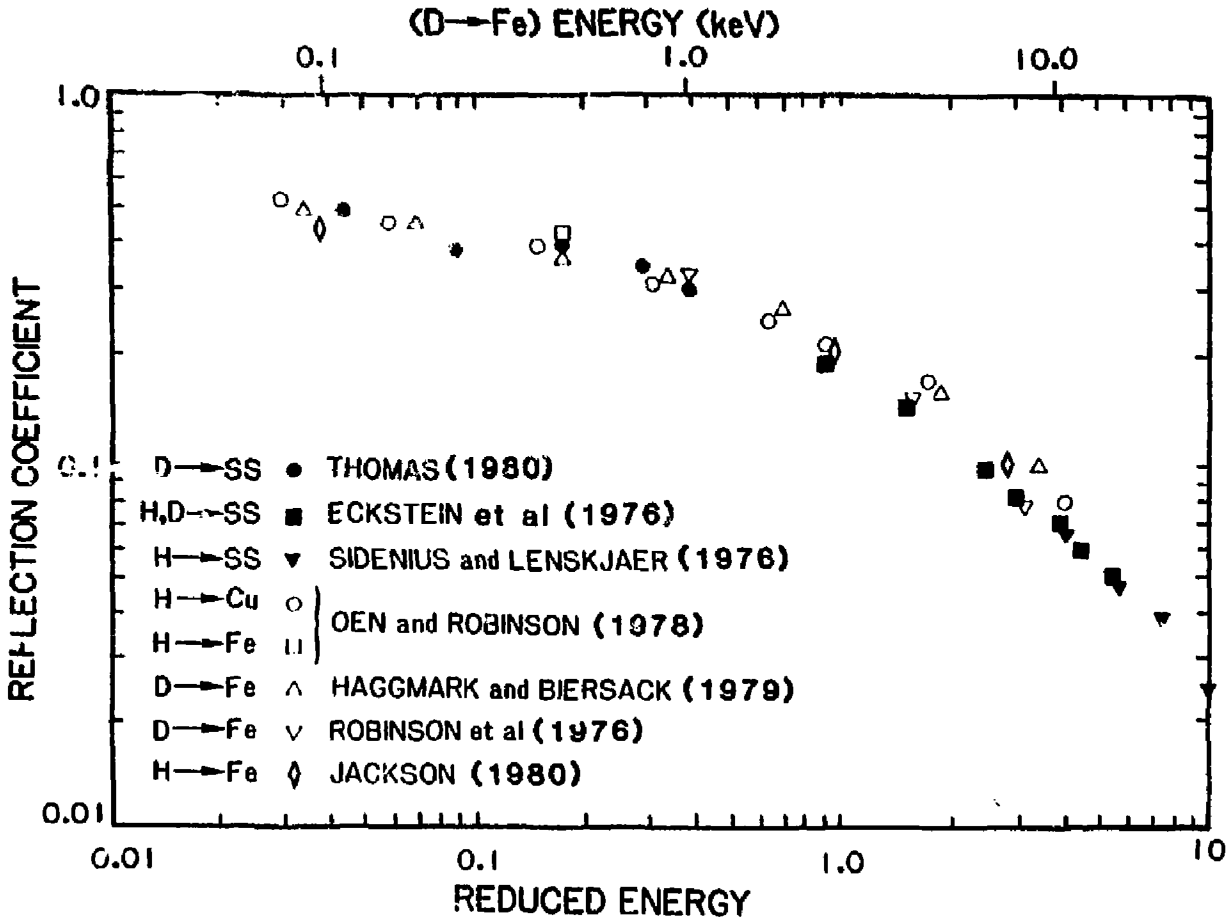
II.1 Reflection

Reflection of ionic and neutral atomic hydrogen is an important physical phenomenon in fusion devices; the measurement and understanding of the physical processes associated with reflection has immediate and critical applications in the design and construction of both first wall components and equipment and devices connected to the first wall.

Reflection coefficients of hydrogen have been measured only down to energies of ∿100 eV as compiled for stainless steel in Figure 2 [2]; at lower energies measurements are difficult because of dramatic loss in primary beam currents in the ion sources currently used to study ion reflection. In addition, at these lower energies, detection of the scattered particles becomes especially difficult since most particle detectors, such as electron multipliers, suffer an exponential loss in gain with decreasing particle energy.

Measurements of the energy and number reflection coefficients down to energies of ∿100 eV show an increase in the reflection coefficients with decreasing energy. Below ∿100 eV, the reflection coefficient is assumed to further increase with decreasing energy. For example, the empirical fit to reflection data at higher energies made first by Seki, et al [3], has been used at energies below ∿100 eV in much modeling of energetic particle transport, including the Princeton group of Post, Heifetz, et al [4]. The Seki model extrapolated towards zero energy predicts a numerical value of reflection coefficient which is approximately unity at an energy of ∿1 eV. At energies below ∿10 eV, chemial effects between the incident particle and substrate can be expected to affect the reflection coefficient, resulting in possibly significant deviations of the

FIGURE 2


(D→Fe) ENERGY (keV)
0.1
1.0
10.0
REFLECTION COEFFICIENT
1.0
0.1
0.01
D→SS ● THOMAS (1980)
H,D→SS ■ ECKSTEIN et al (1976)
H→SS ▼ SIDENIUS and LENSKJAER (1976)
H→Cu ○
H→Fe □ OEN and ROBINSON (1978)
D→Fe Δ HAGGMARK and BIERSACK (1979)
D→Fe ∇ ROBINSON et al (1976)
H→Fe ◊ JACKSON (1980)
0.01
0.1
1.0
10
REDUCED ENERGY

coefficient from the smoothly varying behavior now assumed as input to particle transport codes. Molecular dynamics calculations (e.g., by Baskes and Daw[5]) indicate a decrease in the reflection coefficient at ∿10 eV as a result of the particle-solid attractive forces. It is clear that the low energy picture afforded by the Seki model as input to the particle transport codes cannot be physically correct, and that it is essential to investigate the consequences of using reflection coefficients predicted by models such as those developed by Baskes and others on the transport code predictions.

The critical gap in the hydrogen reflection database, therefore, is in both the theoretical modeling of, and experimental measurement of, reflection coefficients at energies below ∿100 eV. This information is necessary as input to particle transport codes, which are used in particle pumping applications to codes which model plasma-wall interactions. Reflection data thus has immediate applications and, since the models are used to help design planned components such as pump limiters, these same data have essential uses in devices planned for the future. Without these data, it is possible that some components can be designed, but it is unrealistic to believe that the design thus produced will be as efficient or as economical as one based on a trustworthy database.

II.2 Athermal Desorption Processes

Desorption of hydrogen (and its isotopes) from plasma-side materials can occur by athermal (i.e., ballistic, electronic) or thermal mechanisms. This section will discuss the first two of these mechanisms.

Ballistic desorption results from surface impact of energetic atoms and ions escaping from the plasma. At plasma-edge energies, this process is reasonably efficient (cross section $\gtrsim 10^{-16}$ cm^2 [6]) since hydrogen desorption by particle impact is most efficient when incident particles and desorbed species are similar in mass. In high flux regions, hydrogen desorption via a ballistic mechanism can be substantial if sufficient hydrogen exists on plasma-side surfaces. The amount of surface hydrogen in turn depends upon the balance between hydrogen replenishment due to gas phase adsorption as well as bulk to surface diffusion of implanted hydrogen on the one side and hydrogen depletion due to desorption processes and other effects, such as molecular recombination, on the other.

Desorption resulting from electronic excitation of surface atoms may also occur during electron or photon irradiation. The cross sections for these electronic processes are orders of magnitude less than for ballistic desorption, so electronic mechanisms appear to play a minor role in hydrogen desorption.

Desorption is intrinsically associated with hydrogen recycling.

In the near term (TFTR), desorption will cause transient recycling effects in devices having moderate pulse lengths and low duty cycles. In longer pulse devices that approach equilibrium conditions, steady-state desorption will affect hydrogen loading of plasma-side materials. In the long term (ETR) desorption effects may influence the performance of auxiliary components, such as direct converters, rf guides, and sensors for disruption control.

Laboratory measurements have studied in some detail individual aspects of desorption, but only for few materials. Little or no data exists for hydrogen desorption from important materials such as B_e, C, and refractory compounds. Machine based experiments have given an indication of some overall desorption effects. What is needed is to obtain data on desorption from actual vessel materials and to couple together what is known about the individual mechanisms in order to form a general model applicable to the machine environment. Some basic questions to answer along this path are: What is the balance between ballistic and thermal desorption in startup and steady-state? Does surface topography influence desorption? How is surface coverage related to bulk hydrogen concentration in a plasma environment? Do strong rf or B fields affect desorption? Addressing these issues will help to identify the critical effects of desorption upon overall recycling and transport of hydrogen.

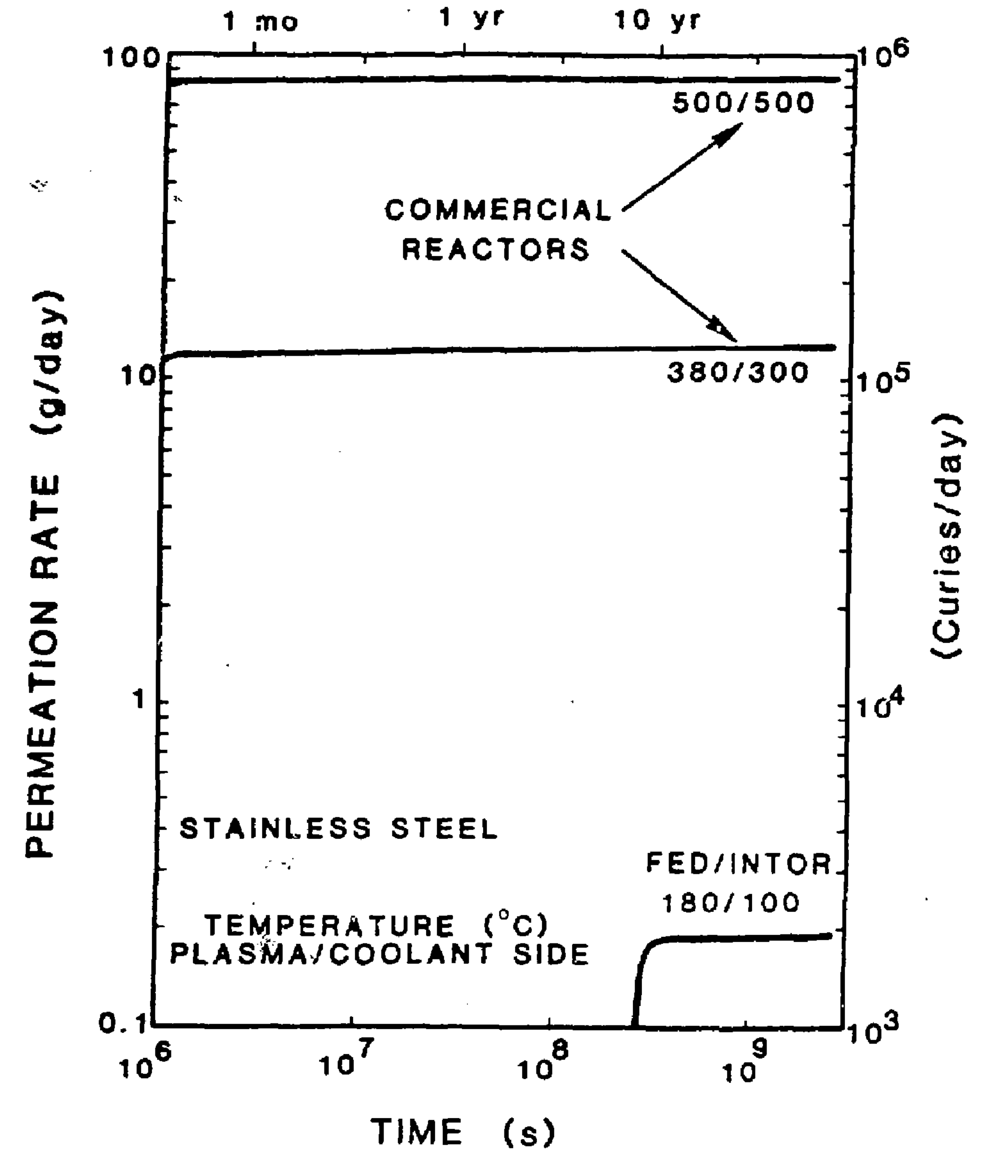
II.3 Solid State Transport and Thermal Release

Hydrogen transport refers to the behavior of hydrogen atoms that come to rest in a first wall, limiter, or divertor material. In general, hydrogen has three possible fates: (1) it can diffuse to the surface and be recycled back to the plasma by athermal or thermal processes; (2) it can be retained in the material in traps or in thermodynamic solid solution, thus contributing to the hydrogen isotope inventory; (3) it can permeate through the material and be released into the component's cooling system. Hydrogen retention in a material is generally described by a formalism involving hydrogen diffusion in the presence of trapping sites, with a recombination limited kinetics boundary condition for the thermally activated release of hydrogen molecules. Hydrogen retention results from trapping of hydrogen at radiation damage or intrinsic defects, and from bulk hydrogen solid solution formation or second phase (i.e., hydride) precipitation. The complete characterization of hydrogen trapping and release must, therefore, include detailed information on mobility, solubility, and trapping for hydrogen as well as the hydrogen-solid phase diagram. In nonmetals, chemical interactions between hydrogen and lattice atoms (e.g., H-C bonding in TiC) must also be considered.

Hydrogen transport has a direct impact on wall conditioning, vacuum, fueling, and radioactivity characteristics. Hydrogen transport has significant effects on fueling in current devices since steady-state first wall re-emission characteristics are rarely achieved during the relatively short ($\sim$1s) discharges. Startup recycling characteristics are also important in mirror devices. In the time frame of 5 to 8 years, the long-pulse lengths will result in steady-state wall re-emission behavior.

Tritium inventory concerns for D-T will probably peak in this period. Trapped tritium concentrations could reach the 100g (10^6 curies) level, representing a major fraction of the allowable on-site inventory. With the advent of high duty cycle of ETR, full nuclear machines, there will be significant tritium permeation problems. Calculations have suggested that permeation rates exceeding 10 g/day (10^5 curies/day) might occur in this time frame. Figure 3 shows several calculated tritium permeation scenarios for stainless steel first walls.[5] Because of the elevated component temperatures, tritium inventories will not rise as dramatically in this time frame, and allowable on-site inventories will be much higher. The tritium permeation problem must be solved before commercial machines can be economical sources of electrical power.

A reasonable database exists for fundamental hydrogen properties (i.e., diffusion, solubility, trapping, and phase changes) in structural metals such as stainless steels, molybdenum, vanadium, etc., but not for low atomic number metals such as beryllium. The rate of molecular recombination at the surface has been measured for only a few selected metals. Another deficiency in the metals database involves measurements for hydrogen diffusion in the large thermal gradients (Soret effect) that will exist in first walls, limiters, etc., of fusion reactors. Dynamic interactions between hydrogen and neutron-produced radiation damage are poorly understood. Hydrogen transport in nonmetals is, in general, very poorly characterized.



1 mo
1 yr
10 yr
100
10
1
0.1
10^6
10^5
10^4
10^3
10^6
10^7
10^8
10^9
500/500
COMMERCIAL
REACTORS
380/300
STAINLESS STEEL
FED/INTOR
180/100
TEMPERATURE (°C)
PLASMA/COOLANT SIDE
PERMEATION RATE (g/day)
(Curies/day)
TIME (s)

FIGURE 3

III. IMPURITY INTRODUCTION

Impurity introduction has a dual impact on fusion. Wall and limiter atoms injected into the plasma by plasma-materials interactions can adversely affect device performance, as evidenced by the hollow PLT temperature profiles caused by radiation of tungsten impurities from the limiter. In future reactors operated with high duty cycles, limiter and divertor plate designs are dominated by erosion lifetime considerations.

This section summarizes seven major impurity introduction mechanisms: (1) impurity desorption; (2) physical sputtering; (3) evaporation; (4) disruptions; (5) chemical erosion; (6) arcing; and (7) blistering. A discussion on the redeposition of the eroded material is also included.

III.1 Impurity Desorption

Desorption is the release of an adsorbed species from a surface due to irradiation or thermal excitation. Under irradiation, desorption can result from sputtering of the adsorbed species (ion-induced desorption) or from electronic excitation due to electron or photon bombardment. Table 1 compares these mechanisms.[1] Thermal desorption is the thermal release of adsorbed atoms or molecules from a surface. Thermal desorption often occurs following surface recombination and is a necessary step in the reactive sputtering process. Impurity desorption is also important in wall conditioning.

Desorbed oxygen is the dominant impurity in many of today's fusion plasmas. Desorption of oxygen, however, may become less important in future high-duty-cycle, long-pulse devices. Wall fluxes and desorption cross-sections are usually sufficient to remove an adsorbed monolayer in less than a second. If the desorbed product is pumped away and not replaced, then clean wall conditions will result. An important factor in impurity control will be the ability to prepare sufficiently clean surfaces to minimize impurity desorption during plasma initiation.

Besides adsorption of residual gases, two other impurity sources for desorption are segregation of bulk impurities and redeposition of eroded material. Impurity segregation can occur in areas of thermal stress such as found in arcing and disruptions. Redeposition will be a continual process as plasma interactive components undergo erosion.

There are limited data available on desorption cross sections for a variety of targets and incident particle beams. These data and the knowledge of surface conditions in fusion devices are generally

TABLE 1

Impurity (C and O) release due to desorption from stainless steel.[1]

	Incident Particle		
	D	Electron	Photon
Impurity Yield (atoms/particle)	2	5×10^{-3}	4×10^{-4}
D-Desorption Cross-Section (cm^2)	10^{-16}	10^{-17}	10^{-20}-10^{-18}
Incident Flux (cm^{-2} s^{-1})	10^{16}	10^{16}	10^{17}
Released Impurity Flux (cm^{-2} s^{-1})	2×10^{16}	5×10^{13}	4×10^{13}

insufficient to make reasonable estimates of desorption contributions to recycling and impurity release. Additional measurements of desorption phenomena could be essential in improving wall-conditioning, arc suppression, and impurity control procedures.

III.2 Physical Sputtering

Physical sputtering is the energetic removal of a near surface atom from a solid target (wall) as a result of an atomic collision sequence initiated by an energetic incident particle. In fusion devices a small reaction of energetic particles, both charged and neutral, escape the plasma edge region and impinge on the first physical barrier of the plasma chamber, e.g., first wall, limiter or divertor. Wall atoms sputtered from the surface may be ionized in the plasma edge region and subsequently escape the plasma and impinge on the wall and cause further sputtering. Thus, one must consider sputtering by primary plasma particles, viz., D, T, and He, impurity ions such as oxygen in the plasma, and self-ions which were previously sputtered from the wall.

In fusion devices, sputtering contributes to impurity introduction into the plasma and to erosion and surface modification of plasma side materials, particularly on limiter/divertor surfaces. The sputtering process is extremely sensitive to surface properties of the wall material since sputtered atoms originate within the first few monolayers of the target surface.

Current experimental observations and predictions from reactor design studies indicate that physical sputtering will be an important erosion problem for a limiter or divertor and a dominant source of plasma impurities. In fact, erosion caused by physical sputtering of the limiter/divertor is generally considered as a primary feasibility problem. As such, physical sputtering has been given considerable attention in the fusion community.

Physical sputtering in current machines (0-3 y) is projected to be a dominant source of plasma impurity. For flagship machines (3-6 y) erosion of limiters/divertors is projected to be a serious problem. In subsequent machines, physical sputtering may be the primary consideration for defining acceptable plasma edge conditions, limiter/divertor lifetime, and material selection for the limiter/divertor.

Physical sputtering yields are known to be dependent on the energy, angle of incidence and mass of the incident particle, as well as the target (wall) material and the condition of the wall. The physical sputtering data base is fairly well developed for most materials of interest.[7] Most of the experimental data are for normal incidence with light ions. Theoretical and empirical models have been developed to predict physical sputtering yields for materials and conditions where data do not exist. It is generally believed that sputtering yields can be predicted within a factor of 1.5-2 for most conditions of interest. This is considered adequate for the current uncertainties in plasma edge physics (e.g., sheath potential effects, edge temperature and angle of incidence). Additional data are needed in selected areas to reduce uncertainties, e.g., angular dependent yields, low energy yields, yields for redeposited material, self-sputtering yields for selected materials, and yields for selected compounds and multicomponent materials, including solute segregation effects.

Because of the critical impact of physical sputtering in future reactors, it will most likely be necessary to refine physical sputtering data in selected areas when the operating scenarios are better defined. Also, innovative approaches to control sputtering, such as surface modifications, should be investigated.

III.3 Evaporation/Sublimation/Melting

Under normal operating conditions, evaporation/sublimation/melting can be significant for components onto which the heat flux is concentrated (i.e., limiters, divertor plates, and beam dumps). The energy deposition on the first wall is assumed to be uniform and low enough not to cause major thermal problems. Under fault conditions like major thermal plasma disruptions or runaway electron discharges, excessively high heat loads which cannot be dissipated without substantial evaporation/sublimation/melting have to be expected on the limiters and parts of the first wall. To avoid damage to the vacuum vessel due to melting and evaporation, the critical locations must be protected with limiters and/or wall armor which can either tolerate higher heat fluxes than the vacuum vessel or function as sacrificial elements. In high duty cycle devices, evaporation/sublimation cannot only lead to impurity release, but also to severe erosion problems.

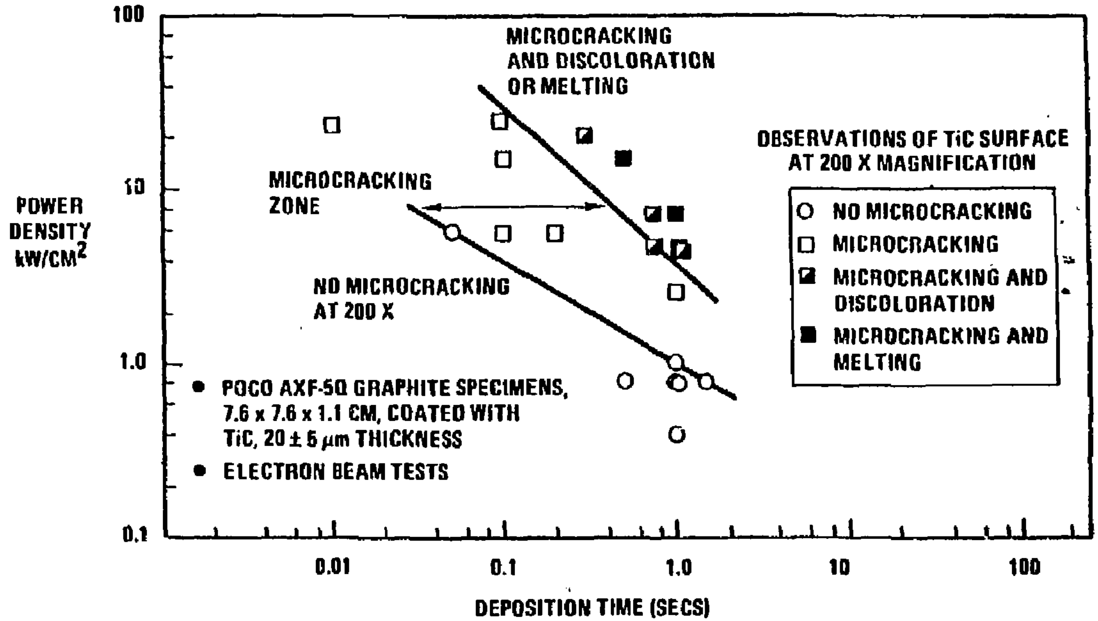
Evaporation/sublimation/melting impacts impurity control, the design of limiters, divertors, RF-antennas and NBI-components, and becomes the dominant erosion mechanism during disruptions. Impurity introduction by sublimation can be quite substantial (e.g., on inertia-cooled graphite limiters). For example, if the surface temperature reaches 2000^{o}C, the vapor pressure is $2 \cdot 10^{-5}$ torr, corresponding to a sublimation rate of $4 \cdot 10^{15}$ cm^{-2} s^{-1} which constitutes a significant impurity source if the limiter has the size of several square meters. Melting and resolidification can also significantly change the surface characteristics of the material. During major plasma disruptions the heat load becomes localized and reaches values such that no protective elements can dissipate the energy without substantial evaporation/sublimation/melting.

In currently operating devices like PLT or TFTR the limiters are inertially cooled and the surface temperature usually increases monotonically during the discharge. Although the pulse lengths are only on the order of 1s, the high loads in the kW/cm^2 range cause high surface temperatures and corresponding evaporation/sublimation. Figure 4 shows laboratory tests for TiC coatings subjected to various power densities and deposition times.[8] In some cases, temperatures close to 3000°C have been measured on graphite limiters. In this case, sublimation rates in excess of 10^{20} s^{-1} must be expected and impurity introduction by sublimation is orders of magnitude higher than that due to sputtering. In future machines, the heat-dissipating components will have to be designed for steady-state operation, including active cooling. This calls for power loads in the range of a few hundred watts/cm^2 and operating temperatures well below the onset of substantial evaporation. Thus, evaporation/sublimation/melting and corresponding erosion should mainly occur during disruptions only for most fusion reactor materials. In full nuclear machines with substantial neutron fluxes the present data base might not be applicable any more because the thermal data are likely to change under neutron irradiation.

Major uncertainties in designing heat dissipating components are due to the lack of reliable models that predict heat loads as a function of space and time. This is particularly the case for plasma disruptions. A major unknown is the effect of vapor shielding. This phenomenon could substantially impact vaporization rates by reducing energy deposition onto the wall.

The thermal data base on the basic materials used in fusion devices is

FIGURE 4



100
10
1.0
0.1
POWER DENSITY kW/CM^2
MICROCRACKING AND DISCOLORATION OR MELTING
MICROCRACKING ZONE
NO MICROCRACKING AT 200 X
OBSERVATIONS OF TiC SURFACE AT 200 X MAGNIFICATION
NO MICROCRACKING
MICROCRACKING
MICROCRACKING AND DISCOLORATION
MICROCRACKING AND MELTING
• POCO AXF-5Q GRAPHITE SPECIMENS, 7.6 x 7.6 x 1.1 CM, COATED WITH TiC, 20 ± 5 μm THICKNESS
• ELECTRON BEAM TESTS
0.01
0.1
1.0
10
100
DEPOSITION TIME (SECS)

quite good. Vapor pressure data are available for all relevant metals considered for in-vessel components. Extensive numerical codes are available and being used to analyze and predict the thermal response to given heat loads. The present data base needs expansion in the following areas:

a) Response to thermal shock

b) Thermal data under neutron irradiation

c) Thermal data of composite materials.

Furthermore, an assessment needs to be made on the effects of the surface topology on evaporation/sublimation. Response of candidate limiter and wall materials to thermal shock as experienced in disruptions needs to be studied with high priority. These data are necessary input for the design of the next machine. The data base for composite materials needs to be established in parallel to their development. Thermal properties under neutron irradiation need to be known for the design of the first full nuclear machine (ETR, late 1990's).

III.4 Disruptions

A plasma disruption will lead to a rapid deposition of the plasma energy on the first-wall/limiter (or divertor) regions of a fusion reactor. This energy deposition is not expected to be uniform on the first wall, but will most likely be highly localized with very high energy deposition rates in certain areas. Effects of a disruption will be strongly dependent on the total plasma energy deposited, the length and time dependent profile of the pulse, the area over which the energy is deposited, the properties of the wall material, and the condition (i.e., temperature) of the wall.

The disruption may lead to vaporization of a part of the wall, melting of surface regions, possible erosion/spallation of the melted layer, and changes in chemical, physical or mechanical properties of the melted or heat affected regions. Electromagnetic interactions (induced current in melt layers with magnetic field) could greatly affect the stability of any melt layer formed during a disruption.

Models have been developed to quantitatively predict the effects of projected disruption scenarios, i.e. the extent of vaporization and melting. The calculated results have been correlated with limited experimental results from simulated disruptions. Effects vary significantly for different types of materials and for different disruption scenarios, e.g., different energy deposition rates and different pulse lengths. Effects of electromagnetic interactions are very difficult to predict.

Disruptions in current fusion devices have caused severe surface effects on limiters and selected first-wall regions. The destructive effects of disruptions are projected to increase in severity as the plasma energy of the fusion devices increases. Reactor design studies have generally concluded that no fusion rector wall can withstand a large number of severe plasma disruptions. However, it is probably necessary to ensure that the integrity of the first wall components will not be severely compromised by a modest number of moderate disruptions. A major uncertainty exists in the severity of a disruption in a power reactor. Probably the greatest uncertainty exists in the response of a melt layer formed during a disruption. Calculations indicate that this is a major erosion problem for many materials if the melt layer formed is not stable, i.e., is displaced. Clearly, early detection techniques for disruptive behavior coupled with preventative measures are essential for power rectors.

III.5 Chemical Erosion

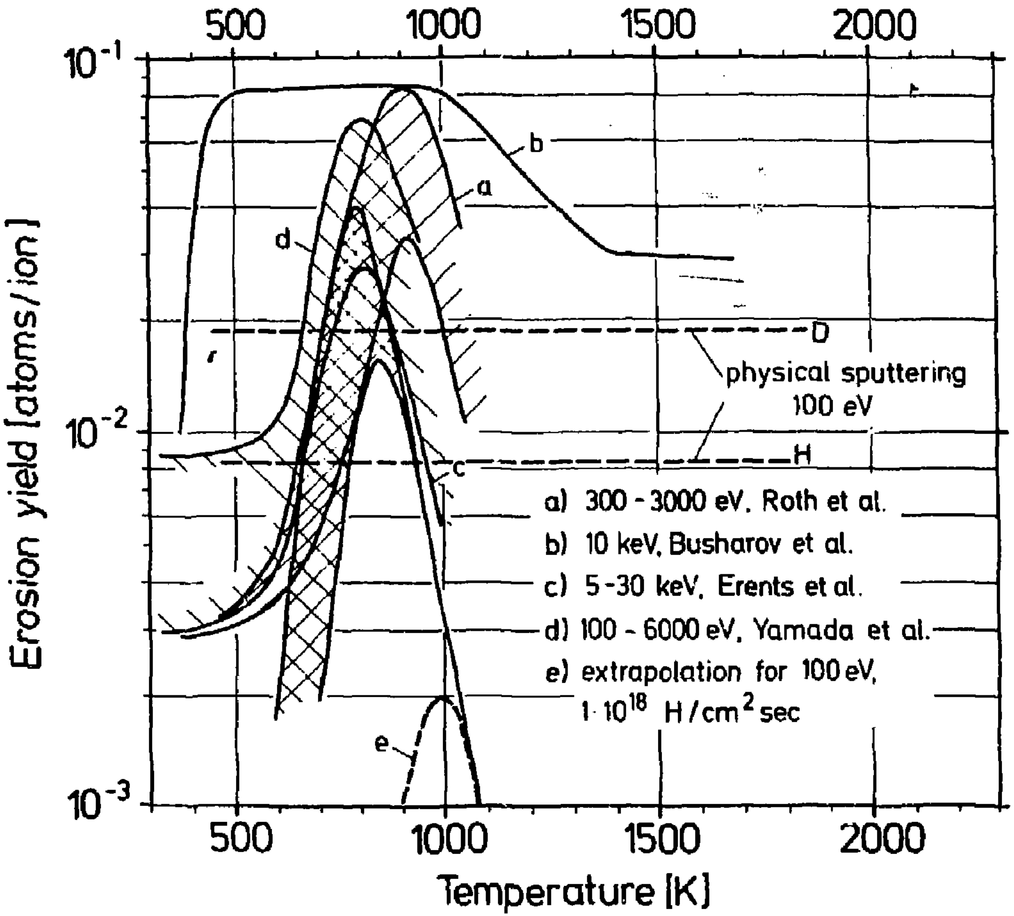
Chemical erosion refers to a broad class of phenomena in which the erosion of a surface under irradiation is influenced by chemical changes induced in the near surface region as a result of irradiation. These phenomena generally fall into two categories, chemically-influenced physical sputtering and reactive sputtering. In chemically-influenced physical sputtering, a surface chemical change (such as oxidation) affects the sputtering yield through, for example, a change in the surface binding energy. Particles released by this process have energy distributions characteristic of knock-on sputtering. Reactive sputtering involves the formation of volatile molecular species under irradiation with subsequent thermal release. Both processes can result in order of magnitude changes in erosion yields in comparison to physical sputtering.

Chemical erosion is often temperature dependent and requires a chemically-active surface and irradiation with reactive ions in a reactive atmosphere. These conditions generally occur in the fusion environment, and the corresponding impact on impurity release, component lifetime, and wall condition can be significant. Chemically-influenced physical sputtering of oxide surfaces which frequently form in fusion devices, often result in a large reduction in the total sputtering yields and changes in the charge state distribution of the sputtered particles as compared with the corresponding clean metal surface. Phase changes and segregation at the surface can also influence erosion yields. An outstanding example, reactive sputtering of fusion materials is hydrogen bombardment of graphite, where irradiation near 900 K results in erosion yields due to methane formation which are ten times the physical sputtering yield as shown in Figure 5.[9] Reactive sputtering is also important for wall conditioning procedures which often rely on the enhanced removal of surface impurities such as C and O by bombardment with reactive ions.

Chemical erosion will be important in both near-term and future fusion devices. In the near term, reactive sputtering and surface chemical effects such as oxidation will influence impurity introduction and materials choices. In the longer term, chemical erosion will become an issue in component lifetime. The influence of phase changes and segregation on erosion yields will become increasingly important for higher wall-temperature and duty factors.

The current level of understanding of chemical erosion is limited. In many cases, there is sufficient data to estimate the potential impact of particular chemical process under well-defined conditions, but there is little fundamental understanding of the process. For example, the near-term use of graphite in fusion devices may depend critically on the presently unknown influence of multiple particle fluxes and impurity deposition on the reactive sputtering yield. Similar gaps appear in the understanding of surface chemistry, phase changes, and segregation in other fusion materials. Accordingly, it is not possible to extrapolate the existing data base on chemical erosion to the fusion environment. Realistic estimates of chemical erosion in fusion devices will require additional research on the fundamental processes involved and on the influence of the plasma surface interaction.

FIGURE 5



500
1000
1500
2000
10^{-1}
10^{-2}
10^{-3}
Erosion yield [atoms/ion]
Temperature [K]
a
b
c
d
e
D
H
physical sputtering
100 eV
a) 300 - 3000 eV, Roth et al.
b) 10 keV, Busharov et al.
c) 5-30 keV, Erents et al.
d) 100 - 6000 eV, Yamada et al.
e) extrapolation for 100 eV,
$1 \cdot 10^{18}$ H/cm^2 sec

III.6 Arcing

Metal surfaces exposed to plasmas with densities of $n_e \simeq 10^{11}$ cm^{-3} and electron temperatures of $T_e \simeq 10$ eV can be subject to extensive surface erosion with a pattern similar to that of vacuum arcs. The arcs causing this erosion appear to be driven by the space charge potential between the plasma and the wall and require only one electrode. For this reason, they are called 'unipolar arcs.'

Arcing processes impact wall-conditioning and impurity control. One single arc can release 10^{16} - 10^{18} impurity atoms, enough to cause a major disruption if the impurities consist of high-Z material. Therefore, it is crucial to control arcing. In present day machines wall-conditioning by discharge cleaning, as well as by regular plasma operation seems to be very efficient in cleaning up potential arcing sites.

In currently operating machines arcing has been identified as a transient phenomenon which normally does not occur during the steady-phase of the discharge after the initial conditioning phase of machine operation. It is not clear now whether or not future devices with higher plasma densities and temperatures can be conditioned similarly.

It is generally assumed that the cathode processes of unipolar arcs are similar to those of common vacuum arcs and the basic cathode phenomena like arc initiation, current density, cathode spot stability, arc velocity or erosion rates have been studied extensively and are well covered in the literature.[10] The susceptibility for arcing is a function of material and surface conditions of the electrode as well as the plasma parameters. All metal and graphite surfaces examined so far show conditioning effects such that the final arcing rate is only a small fraction of the initial rate.

After the conditioning phase present day devices show arcing mainly in periods of enhanced MHD activity: current rise, disruptions, end of discharge, introduction of impurity by gas puffing or laser blow-off, and during runaway discharges. The observed cathode spots can be divided into two different types with different velocities and surface erosion: type 1 spots burn mainly in evaporated contaminants, are more easily initiated, move faster and have lower erosion rates than type 2 spots, while type 2 spots burn in the vapor of the electrode material proper, produce larger craters and move more slowly. It is conceivable that conditioning only works for type 1 arcs by cleaning up the surface contaminants. When the plasma density and temperature get high enough to initiate type 2 arcs directly, arcing might occur continuously and even in the steady phase of the discharge.

Erosion rates are approximately one surface atom for every 20 electron-charge units of cathode current for electrodes based on Fe, Ni, Cr and Cu, or for every 30 electrons of current to the refractory metals Mo, Ta and W. The eroded material can be lost as ions, neutral vapor and/or macro particles. The erosion pattern of the cathode spot allows certain conclusions concerning the direction of motion. Regular vacuum arcs and unipolar arcs in high transverse magnetic fields move in the "retrograde" direction, opposite to the Lorentzian force $j \times B$. The velocity of arc motion depends on arc current, magnetic field strength, cathode material and surface conditions. Typical velocities for tokamak conditions are 20-200 m/s.

The data base on arc parameters as covered by the vacuum arc literature is quite extensive. Two areas which are specific for arcing in fusion devices need attention:

a) Arcing properties of special materials (e.g., graphite, coatings) in strong external magnetic fields.

b) Conditioning of arcing in future devices with higher electron density and temperature.

III.7 Helium Trapping and Blistering

Blistering is the formation of surface deformation or exfoliation by the accumulation of insoluble gas beneath the surface. While hydrogen blistering has been reported for certain low solubility metals (e.g., Ci, Mo, Be) implanted near room temperatures and for certain nonmetals (e.g., TiC), it is unlikely that significant erosion will result under reactor operating conditions. Alpha particle blistering is considered to be a more significant problem under certain reactor operating conditions. Blistering can occur on any surface exposed to high energy (>1 keV) alpha particles. Since helium is insoluble in any metal, it immediately precipitates out of solution at the end of implant range. Small helium bubbles develop and eventually interconnect. If the interconnected layer occurs without any pathway to the surface, then surface deformation (blistering) will occur to release the gas pocket. In the temperature range of 0.3 to 0.5 of the melting temperature, the entire surface layer will exfoliate, exposing a fresh, new surface. Multiple exfoliation of micron thick surface layers has been demonstrated under certain well-defined laboratory conditions (polished surfaces, normal incidence, monoenergetic energy). Calculations indicate that reactor first walls may blister from the accumulation of unconfined 3.5 MeV alpha particles. Blistering may also be a concern in the direct converters of mirror devices.

Blistering erosion could not be a serious concern until the ETR time frame, since a high concentration of helium ($\sim$0.3 He/metal) must be achieved to produce any likelihood of blistering. This condition should occur only in a high duty cycle D-T machine. If it proves to be an important erosion source for ETR, then steps can be taken to eliminate blistering for the commercial reactor phase. Surface roughening, cold work, and bulk porosity have been demonstrated as effective cures for blistering in laboratory studies.

Helium blistering in metals has been studied in great detail. One major gap in the database is information on the more subtle change in mechanical properties (i.e., fatigue, crack growth) that occur from helium bubble precipitation and interconnection. Another area involves the determination of synergistic effects of hydrogen and helium under realistic limiter conditions. Finally, additional research of helium effects in non-metals is also warranted.

III.8 Redeposition

Redeposition of material eroded from the wall (or limiter/divertor) of a fusion reactor is expected to occur under two different conditions. During normal operations material sputtered from the first wall or limiter/divertor by energetic plasma particles (ions or neutrals) will redeposit on selected regions of the first wall, limiter or divertor. During a disruption vaporized or melted material from the first wall or limiter/divertor may deposit in a region different from that where it originated.

The conditions for redeposition are not well defined. However, several possible consequences are projected for anticipated scenarios. Properties of the redeposited material may be significantly different than those of the original material. If different materials are used within the plasma chamber, alloying or compound formation may occur. Also, a change of stoichiometry or composition of compounds may occur, and solute segregation of alloys may result. For certain materials such as graphite, a change of structure to amorphous carbon may occur. All of these effects can change the properties of the wall materials and produce changes in subsequent responses of the plasma-wall interactions.

Effects of redeposition have been observed in existing devices. These effects are projected to become more severe as the plasma energy is increased in subsequent devices. Since the effects will be device and design dependent, only generic studies can now be conducted to investigate the severity of anticipated effects and possible solutions. Currently, very little pertinent information in this area is available. Operation of long pulse devices should provide increased understanding of the magnitude of this problem. In the near term, it will be necessary to conduct carefully controlled laboratory experiments to evaluate and predict the effects of redeposition that may occur in long pulse, high power fusion devices.

IV. VACUUM ISSUES AND SPECIAL COMPONENTS

Control of vacuum conditions plays critical role in the successful operation of present day devices. While many wall conditioning problems may be alleviated in high duty cycle or steady state reactors, start-up scenarios will remain intimately connected with vacuum issues. This section summarized two present day options: (1) wall conditioning techniques; and (2) gettering.

Special components often introduce new plasma-material interaction issues. As an example described in this section, RF heating has required the development of low secondary electron emission coatings to handle high power loadings.

IV.1 Surface Conditioning

Surface conditioning refers to the sum of the physical and chemical processes that are applied to first-wall components prior to plasma operation. Conditioning is necessary to minimize impurity influx by thermal and particle-induced outgassing and other plasma surface interactions. In addition, conditioning also affects the hydrogen recycling behavior and the susceptibility to arcing of first-wall surfaces.

Successful conditioning methods have been documented in the present generation of fusion devices, which primarily use stainless-steels as the first-wall material. The conditioning involves careful pretreatment of the material to insure good vacuum qualities (i.e., low outgassing surfaces, bulk material free of large voids, defects, and cracks that can serve as leak paths). The material is usually physically and chemically cleaned to remove macroscopic contamination, and vacuum baked to degas the bulk material. Pretreatments vary in effectiveness; however, pretreatments cannot stand alone as a conditioning method. In situ conditioning, involving extensive interaction with hydrogen plasmas or atomic hydrogen, is necessary to minimize plasma-wall interactions. The most successful form of in situ treatment, as demonstrated on the present generation of operating devices, is some form of hydrogen discharge cleaning: usually low temperature pulse discharge cleaning or dc (or rf-assisted) glow discharge cleaning, performed at elevated wall temperatures (100-350°C). Successful wall conditioning is defined as the point at which low-Z impurity introduction from first-wall surfaces is negligible, the hydrogen recycling properties are well-characterized and static, and the tendency for unipolar arc formation is minimal.

The methodology described above will continue to be relevant for the first-order vessel conditioning of the next generation of fusion devices, especially following device commissioning, atmospheric vents, and major changes in operation cycles. However, the more important surface conditioning problem, both for the devices presently coming on-line (TFTR, JET and JT-60) and the next generation, will be an understanding and optimization of the conditioning process(es) required for plasma-contacting surfaces such as limiters and divertor plates. This problem represents a transition of the surface conditioning process from basically a problem in surface chemistry to a problem concerned with the response of materials to high plasma ion fluxes and high heat loads.

In contrast to the conditioning of stainless-steel vacuum vessel surfaces, the conditioning of high flux limiter surfaces remains an open issue. As an example of the problem, graphite and carbide-coated graphite limiters have been used in many presently operating devices (PLT, PDX, Alcator-C, ISX, TFR, etc.). Much operation time has been spent in these devices to condition the limiter surfaces so that the limiter has a minimal effect on plasma operation, i.e., tolerable impurity influx and static recycling properties. The task remains of quantifying what sufficient limiter conditioning means in terms of the chemical and structural changes that occur at the surface (or near-surface) layer of the candidate limiter materials. Secondly, once the conditioning process is understood, it should be optimized to minimize impact on device operations.

Looking toward reactors, the need for in situ surface conditioning will tend to diminish, as the high heat loads and much higher duty cycle will accelerate the conditioning process. However, pretreatments of materials, and judicious choice of materials for plasma-contacting surfaces will dominate the conditioning problem in fusion reactors because erosion and redeposition processes will govern the nature and composition of first-wall surfaces.

IV.2 Gettering

In the past, gettering has been used in many tokamaks as a means of reducing impurities and controlling density. The gettering process involves the insertion of one or more sublimation sources into the vacuum vessel and the deposition of a fresh layer of the getter material. Titanium has been used almost exclusively because of its well-known gettering properties and the commercial availability of suitable sources. Other metals under discussion are aluminum and chromium which are both different from titanium in that they show no significant diffusion of hydrogen isotopes into the bulk.[11] This minimizes the hydrogen content in the getter film and, consequently, the tritium inventory in future DT-devices. Another alternative for impurity control with getters at minimum tritium inventory are Zr-Al bulk getters as installed in TFTR.[12] They are installed ready with heater elements for temperature control. At moderate temperatures (200-400°C) the bulk getters provide high pumping speeds for impurities and hydrogen isotopes while at high temperature (500-700°C) hydrogen (tritium) is outgassed and can be fully recovered, whereas the impurities diffuse into the bulk to provide a fresh surface for the next pumping cycle.

Gettering impacts wall conditioning, impurity control, vacuum conditions and fueling of the plasma. Wall conditioning and impurity control are affected in two ways: the freshly deposited metal film (1) changes the adsorption/desorption characteristics of the wall, and (2) covers the wall and with it the impurity source. The combination of both effects provides for a very efficient impurity control. Since titanium also pumps hydrogen isotopes, the effective recycling of fuel particles is reduced and the external plasma fueling has to be increased correspondingly. Gettering is also very effective in achieving low base pressures in fusion devices. After a typical getter cycle the total pressure usually decreases by one to two orders of magnitude.

The utility of gettering beyond TFTR seems to be limited. Surface gettering has been proven to be useful in pulsed, low-duty cycle operation while it does not seem to be practical in long pulse, high duty cycle or even steady state operation of future devices. Depending on the gas load, getter films tend to spall off the substrate after reaching a thickness of a few tens of microns. Bulk getters provide a high capacity for impurities, and the experience to be gained on TFTR will allow us to assess their utility in future machines.

Studies on gettering in contemporary devices have revealed that titanium gettering entails a short-term and a long-term effect. The short-term effect consists of a strong reduction in hydrogen recycling and persists for a few discharges only, i.e., until the titanium is saturated with hydrogen. The long-term effect can persist for at least ten times as many discharges and manifests itself by low oxygen content. It has been found that even after saturating the getter film with hydrogen isotopes, the impurity pumping capability is still maintained until all the hydrogen is replaced with oxygen.

The data base on titanium gettering is well established. For chromium gettering, which might have some applications in DT-devices, the vacuum physics data base is sufficient, but the behavior under energetic particle bombardment needs to be investigated. Furthermore, studies need to be undertaken on adhesion properties and critical film thickness beyond which flaking begins. Gettering seems to be a temporary solution to impurity problems rather than a permanent one. Therefore, the priorities are near-term and have to be set as impurity problems evolve in the present generation of machines.

IV.3 RF Heating

The injection of RF energy directly into fusion plasmas appears very attractive for plasma heating and current drive for fusion reactors. There are a number of possible advantages of RF heating methods in comparison to neutral beam injection (NBI) heating methods, when both techniques are extrapolated to the requirements of fusion reactors: (1) the RF power sources are more easily designed for steady-state operation than NBI ion sources; (2) can be located remote from the fusion device; (3) the energy transport systems (waveguides) are generally simpler in design and less constrained in dimension and location than the vacuum ducts required for NBI beam lines; and (4) the cross-section of the RF antenna on the fusion device wall is generally small.

An important problem area in the extrapolation of RF heating and current drive technology for reactor needs is the development of antenna and waveguide launchers that can handle higher RF powers at eventual steady-state power loadings. The development for these RF components for reactors requires significant attention to choice of materials, surface treatment, and plasma-surface interactions. Only in the last few years have surface physics issues been addressed in the design and treatment of RF components. As examples, first the design of the ICRF antenna arrays for TFR and PLT required the use of graphite and alumina insulators and extensive in situ hydrogen discharge cleaning to enable the designed power levels to be handled; and, secondly, the LHRH waveguide launchers have been designed and successfully tested on PLT with low secondary emission carbon coatings that allowed 400 kw input power to be sustained without arc-over. A similar surface-coating development program is being performed for the

LHRH launchers to be used in ASDEX.*

The reduction of the secondary emission characteristics of the antenna/launcher structures is a demonstrated and well-defined goal. The development work must be continued in a manner that takes into account the integral effects of plasma-surface interaction and plasma-wave interactions on RF component breakdown characteristics. In addition, RF components face the same development problems of other first wall components, i.e., consideration of the constraints imposed by plasma erosion and neutron irradiation damage.

The goal is to determine the practical limits for RF power loading for a given size antenna or launcher so that the technology can be extrapolated to the next generation of fusion devices, and eventually to use in reactors.

V. SUMMARY

Considerable surface physics research has been conducted since the last plasma-material interaction technical assessment. Some research areas (e.g., physical sputtering) have become quite mature, while new topics (e.g., RF components) have surfaced. In regards to laboratory experiments, additional research on fundamental processes are needed as well as refinements of surface physics data in critical areas. The priority of further surface physics research in actual devices must evolve with experience gained during operation. In particular, operation of long pulsed devices should provide increased understanding of the issues where erosion and redeposition process will ultimately govern the behavior of plasma-interactive surfaces.

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FIGURE CAPTIONS

Figure 1. The fundamental mechanisms involved in hydrogen recycling.

Figure 2. A summary of experimental (solid) and theoretical (open) reflection coefficients for H—stainless steel.[2]

Figure 3. A calculation of tritium permeation through stainless steel first walls of various temperatures.[5]

Figure 4. Electron beam testing of T.C.[8]

Figure 5. A summary of chemical erosion of graphite.[9]

Part D

Technical Assessment of the Critical Issues and Problem Areas in the Plasma-Materials Interactions Field:

Materials Technology

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1. INTRODUCTION AND SCOPE

The success of magnetic fusion depends on the availability of engineering materials that can perform under high fluxes of particles and heat. The problem is a synergistic one in that the simultaneous resolution of a number of issues in plasma physics, surface physics, materials science, and engineering design is required.

As part of a technical assessment in the area of plasma-materials interactions, this chapter concentrates on "plasma-side" materials. Structural properties such as the selection of alloys for actively cooled components are discussed in the High Heat Flux Task Group's assessment. The problems concentrated on here involve:

a) the ability of materials to survive in the plasma environment, and

b) mitigation of deleterious effects on the plasma by the introduction of impurities.

Both the survivability of materials for in-vessel components (IVC's) and the mitigation of deleterious effects on the plasma are central to the commercial success of fusion. In the mainline tokamak and mirror approaches, the plasma is surrounded by a relatively low-power-density structure. Contact with the plasma is maintained by high heat flux systems that are a small fraction (~ 7% for the STARFIRE conceptual design) of the first-wall area. In compact systems such as the Reversed Field Pinch, IVC areas are nominally the same. The materials technology required of the compact systems, therefore, differs little in principle from that projected for the mainline systems; the EMF components are of comparable size (area and volume), but the inefficient low-power-density systems have been eliminated, leading to systems that are reduced in total volume by 10-30 while generating similar total power. The magnitude of the peak critical heat fluxes remains the same for both. Furthermore, the ratio of particle to neutron fluxes for both mainline and compact systems are similar; both would reach the same IVC erosion fluences for the same neutron irradiation fluence limit. The impact on plasma performance of the higher impurity influx is greater for the compact systems, however, showing a stronger coupling between a) and b) listed above.

In present and future experiments, and in toroidal fusion reactors, impurity atoms produced by the contact of the plasma with materials must be controlled to permit long, stable discharges and high plasma temperatures to be achieved. Similarly, in large mirror machines and mirror reactors, impurities due to the erosion of material in the center cell and end fan regions can limit the achievement of long-pulse, high-temperature plasmas. All approaches to impurity control in tokamaks place severe demands on materials. Limiters and divertor collector plates must be able to withstand repeated thermal cycling and undergo physical and chemical erosion processes. In addition, tokamak in-vessel components such as limiters, armor tile and RF antennas are subjected to thermal shock due to plasma disruptions. Corresponding demands are placed on materials by mirror technology (direct convertor collector plate, limiter, RF antenna, etc.).

The mechanisms reponsible for the introduction of impurities into the plasma are discussed extensively in the Surface Physics section of this assessment and will not be analyzed in detail here. These problems are (1) physical sputtering, (2) chemical erosion, (3) impurity desorption, (4) arcing, (5) blistering, (6) evaporation, and (7) disruptions. The Surface Physics section also includes a discussion on redeposition of eroded materials.

To reduce radiation losses due to ions that are products of erosion, materials with low atomic number have been chosen. This has led to efforts to develop low-Z refractory and non-refractory coatings and claddings on metal substrates that meet specific requirements with respect to hydrogen recycling, hydrogen (tritium) retention (inventory), hydrogen (tritium) permeation, as well as degradation from thermal, hydrogen, and radiation effects. For long-pulse machine applications, extensive erosion may require that coatings be of the order of 1 cm thick to give adequate component lifetimes.

It is important to note that materials needs have been identified often by conceptual design studies. With high power density, long pulse duration machines now being constructed and in various planning stages, considerably more is required than for conceptual designs. Difficult materials choices must be made on time scales compatible with the projects, and an expanded materials data base as well as improves estimates of subsystem performance are required.

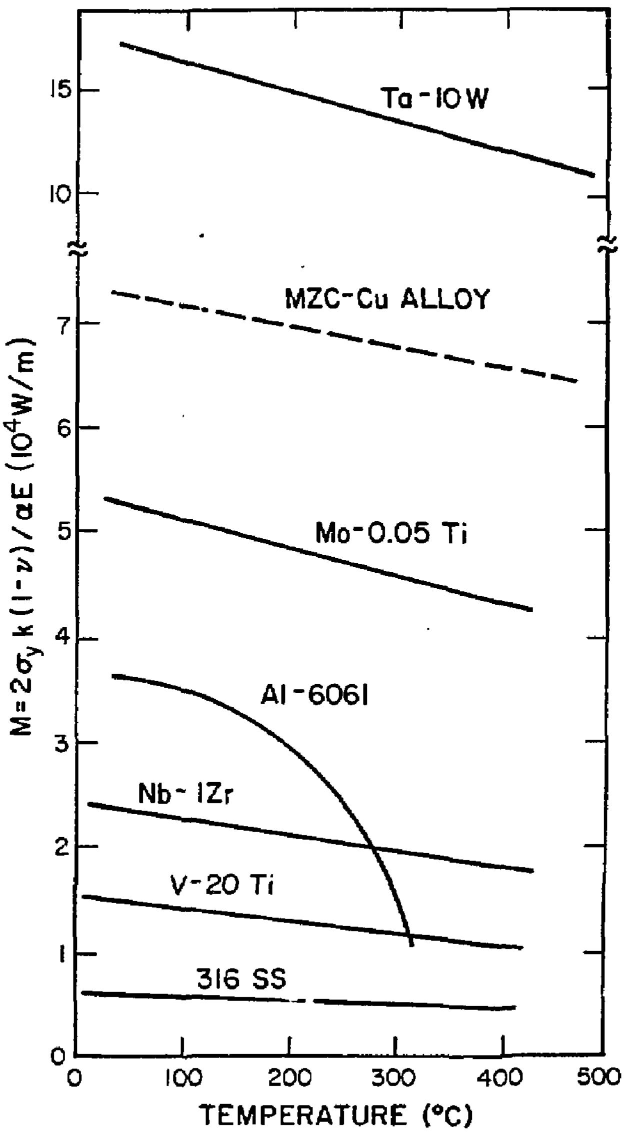
In the next section, we summarize armor fabrication techniques and problem areas. The following sections then focus on specific assessments for three time frames: 0-3 years, 3-8 years, and 8 years and beyond. The final section contains a summary and conclusion.

2. MATERIALS FABRICATION

Generally, a majority of IVCs will be fabricated from structural alloys, preferably with a HHF capability, that in some applications would be coated with a low-Z armor. The materials technology required of the IVCs therefore, can be categorized according to coating armor versus heat-sink functions.

2.1 Heat-Sink Material

Austenitic stainless steel represents a common choice for a structural alloy. When used in a HHF environment, however, alloys with a better thermal-stress response will be required. The available choices are best ranked in terms of the thermal stress parameter, $M = q^*\delta$, where $q^*(MW/m^2)$ represents the heat flux through a thickness δ required to bring the HHF surface to the yield point. Figure 1 gives M as a function of temperature for a range of alloys. Copper, vanadium, molybdenum alloys show superior performance in this respect. The ultimate choice of heat-sink material, however, must be made on the basis of fabricability, cost, and compatibility with the coating/armor material. Other crucial issues for these newer, improved alloys are related to the development of a creep/fatigue data base that will allow certification for quality-assured operation in the fusion environment. This topic is addressed in greater depth in the High Heat



Ta-10W
MZC-Cu ALLOY
Mo-0.05 Ti
Al-6061
Nb-1Zr
V-20 Ti
316 SS
$M=2\sigma_y k(1-\nu)/\alpha E$ (10^4 W/m)
TEMPERATURE (°C)
0
100
200
300
400
500
1
2
3
4
5
6
7
10
15

FIGURE 1.

Flux Group's assessment.

2.2 Coatings and Armor Material

There are two alternative approaches to fabricating surface armor of plasma side materials for high heat flux components. One is to make a preformed cladding that is subsequently bonded to the heat sink by processes such as brazing, diffusion bonding or explosive welding. The other is to deposit the plasma side material directly onto the heat sink by coating processes, such as vapor deposition or plasma spray, or by use of composite materials in which a low Z material segregates to the surface under plasma bombardment.

Cladding is potentially most attractive for applications requiring very thick armor (several millimeters or even centimeters of plasma side material) applied to flat or nearly flat surfaces, such as divertor collector plates. The critical fabrication issues for cladding are related to the attachment of the cladding to a heat sink.

Coating offers distinct advantages over cladding in three areas: (1) surfacing of complex shapes, e.g., curved limiter blades, (2) coating of large area components, e.g., first wall, and (3) in-situ repair of damaged or eroded surfaces. At present, the primary technology for coating high heat flux components is chemical vapor deposition (CVD) of thin (< 20 μm) layers of TiC onto graphite components. This technology has worked well in ISX-B and Doublet III, and TiC coated graphite will be used in TFTR.

Chemical vapor deposition is suitable for a large variety of materials, including metals (Be, Mo, Ta, W, Re), semiconductors (B, Si), carbides (TiC, TaC, WC), nitrides, borides, and beryllides. The primary disadvantage of CVD (and other vapor deposition techniques) for future applications is the low deposition rate, typically only microns per hour. Thus, it is not practical to build coating thicknesses commensurate with predicted erosion rates that range up to millimeters and even centimeters per year. Many CVD processes also produce corrosive by-products and require high process temperatures that can adversely affect metal substrates.

Plasma spray coating is a promising technology to overcome the limitations of vapor deposition. Industry experience has shown that nearly all metals spray well and can generally be deposited to thicknesses ranging from several millimeters to more than one centimeter. For example, beryllium and tungsten have both been deposited to thicknesses greater than one centimeter. Ceramic compounds that have a stable molten phase (BeO, B_4C, MgO, Al_2O_3, TiB_2, TiC, VC, etc.) can also be plasma sprayed, but they are sometimes more difficult to apply and residual stress limits maximum coating thicknesses of pure ceramic coatings to a few millimeters or less. Materials that sublime or dissociate before they melt, such as graphite, cannot be plasma sprayed unless they are co-deposited with at least 20% of a second sprayable material to form a composite coating. Composite coatings of ceramics and metals (cermets), such as SiC/Al and SiC/Ni, have been plasma sprayed and look very promising for high heat flux applications. However, these materials are still in the development stage. Graded mixtures of the heat sink and plasma side materials can also be spray deposited to fabricate a smooth transition zone between the substrate and surface coating. This may alleviate stress that is caused by the typically large mismatch in thermal expansion between candidate heat sink and plasma side materials. Segregation of mobile solutes, which tend to concentrate at sharp interfaces between dissimilar materials, may also be reduced.

In the past, the primary disadvantages of plasma sprayed coatings have been high porosity (typically 80 to 95% of theoretical density) and poor adhesion in some cases. Recent advances in technology to spray coatings in an evacuated chamber (so-called low-pressure or vacuum plasma spray) have produced coatings ranging up to 99% of theoretical density with excellent adhesion.

An alternative coating approach to cope with high erosion rates is to develop technology for in-situ vapor deposition or plasma spray coating inside a fusion device. This offers obvious advantages, since periodic recoating could be used to repair eroded or damaged surfaces. It would also permit the use of thinner coatings that would result in lower surface temperatures and reduced thermal stresses. Japanese and European researchers have begun preliminary experiments to investigate possible in-situ vapor deposition of TiC or carbon. However, it appears that a long development effort will be required, and the prospects for success are open to question. At present, in-situ coating is perhaps best viewed as a very uncertain, long-range technology that has great potential benefit.

A variation of the in-situ coating techniques is found in the use of self-coating materials, i.e., composite or alloy materials in which the thermal and radiation environment results in the preferential formation of a thin layer of one of the components at the surface. Typically, materials such as Cu-Li or V-Al alloy would be used to form a low-Z over-layer on a thermally conductive/refractory substrate consisting of the high-Z element. By using such materials, thermal mismatch problems at the interface would be eliminated and the erosion of the high-Z component would be reduced. The low-Z coating would not have to be replaced and damaged regions would be self-healing, provided that the low-Z material would be brought to the surface rapidly enough to replace eroded surface material. The low-Z transport rate is a sensitive function of the temperature and radiation spectrum, and a full determination of the behavior in a reactor environment has not yet been made. It is suggested that a series of experiments be performed in devices which simulate plasma edge conditions using weight loss and post-irradiation analysis techniques. For preliminary experiments, single species mono-energetic irradiation at the relevant temperatures will suffice but, since the process is inherently synergistic, it would ultimately be necessary to provide a full spectrum of energies and particles including neutrons.

In summary, erosion is the primary issue that drives coating fabrication requirements. Since the accuracy of erosion predictions cannot be adequately evaluated on the basis of existing experimental data, the only prudent course of action is to continue to develop technology to apply thicker coatings of plasma side materials.

fabrication requirements. Since the accuracy of erosion predictions cannot be adequately evaluated on the basis of existing experimental data, the only prudent course of action is to continue to develop technology to apply thicker coatings of plasma side materials.

Vapor deposition is a proven coating technology that has worked well in existing machines, but the prospects to extrapolate vapor deposition techniques to fabricate extremely thick coatings are poor. Plasma spray coating has been successfully used to apply very thick coatings in industry. However, the materials of primary interest are not widely sprayed in industry, and the limited data that are available do not reflect recent advances in spray technology. It must be confirmed that these materials can be sprayed to adequate thicknesses. Due to the unique microstructures of coatings, measurements of coating properties and testing in fusion relevant environments are also needed to establish the suitability of these coatings for high heat flux surfaces. Technology for non-destructive testing (NDT) of both coatings and claddings must also be improved to assure component reliability.

In-situ coating and self-coating materials may be important for advanced fusion machines. However, it is too early to clearly evaluate the prospects for successful development of this technology.

3.0 0-3 Years

This section contains a review of those materials that are presently being used as plasma side materials in existing machines. In addition, a brief overview of experiments that are being planned for execution during the 0 to 3 year time frame is presented. It should be noted right at the beginning of this section that experiments that can be carried out that would influence materials choices in this time frame are restricted to small area limiters, divertors, or protective plates. This is because the lead time required to design and construct large area devices is of order 2 to 4 years depending on the complexity of the device. As an example, the design and construction of the 20 m^2 bumper limiter for TFTR required 3 to 4 years. This is mainly due to the complexity of such large area devices. This should not be taken to mean that no useful experiments can be done in this time frame because there are several significant small area experiments planned during this time frame.

The materials that are presently being used as plasma side materials cover a wide spectrum. There has been a strong shift away from high-Z materials in the last few years. Even medium-Z materials are used less extensively than in the past. This list of materials being used will include those presently in use and/or those that were used within the last year.

Bare graphite is used as limiters on both PLT and PDX under conditions of intense auxiliary heating (both neutral beam and

RF). Titanium carbide coated graphite is used on ISX-B (moveable limiter and protective armor), PDX (axisymmetric limiter and protective armor), and TFTR (moveable limiter). All of these applications (except TFTR) of TiC/graphite have included strong auxiliary heating. Silicon carbide coated graphite is used as a limiter on Alcator-C. A carbon-silicon carbide coating on graphite is used as part of the Doublet-III protective armor. Vanadium clad copper (with a stainless steel inner layer) was used as the neutralizer plate in the PDX pump limiter experiment. Stainless steel and/or Inconel and/or titanium is used as fixed limiters on D-III, ISX-B, PDX, and Alcator-C. These materials are also extensively used in mirror devices. Molybdenum is the only high-Z material that has been used recently (in Alcator-C).

All of these materials have shown damage caused by one or more of the following: arcing, spalling, melting, and evaporation. The data concerning these phenomena is not sufficient to determine which is most important under different circumstances. The details of these phenomena are discussed in the Surface Physics assessment.

Experiments planned in the near term include one new material and some new designs for old materials. The new material is beryllium that will be tried on ISX-B. Since beryllium has been proposed as an important limiter material for reactors, this experiment should provide important data. There are no other new materials that are scheduled for testing in the near term.

There are several new designs for graphite or coated graphite limiters or pump limiters that are being planned for the 0 to 3 year time frame. These may shed light on damage mechanisms.

In summary, there are a large number of plasma side materials in use. A small number of new materials are planned to be tested in the near term. Such testing in the near term is confined to small devices by design and construction considerations. A better understanding of damage mechanisms is required to select those materials that will perform best in the long term.

4. 3-8 YEARS

In this time frame, materials needs will have to be met for Doublet III-Big D, Alcator DCT, TFCX, later stages of MFTF-B and compact devices such as ZTH. The materials requirements for DCT and TFCX are typical of what is required in this category.

a) Alcator DCT

Alcator DCT is a proposed machine that would produce pulse lengths of the order of one minute with the possibility of being operated steady-state. The preliminary design study includes configurations for a pump-limiter and for internal and external poloidal divertors. For all three the heat flux and sputtering regimes are found to be reactor prototypical. In terms of the power levels deposited on them, heat removal structures fall into three classes:

(1)	direct plasma contact	2-3 MW/m^2
(2)	recycling plasma contact	< 1 MW/m^2

(3) radiation and charge exchange

neutrals from main plasm $\quad .2 - .4\ MW/m^2$

The first category describes divertor plates or limiters. Designs currently being considered include graphite tiles in contact with the plasma, brazed to an actively cooled substrate made of copper alloy. An alternative would be a tile graded in composition from graphite to copper, produced by plasma spraying. As indicated in Section 2., there is a need to evaluate the possible attachment schemes. Good contact between the armor tiles and the substrate is required to keep surface temperatures low; this indicates brazing or diffusion bonding to be appropriate. A possible difficulty is that the high temperatures required in these fabrication techniques can lead to loss of strength of the copper alloy substrate.

Lifetimes of the high heat flux components will be determined either by material loss due to melt layer formation in disruption or by sputtering and redeposition rates. Important uncertainties lie in the parameters that determine these processes. Erosion due to disruptions depends on the initial temperature of the surface, the uniformity of the mel ed layer, the morphology of the surface (redeposited, previously melted, etc.) and the quantity of melt layer lost. Estimates of sputtering and redeposition rates depend strongly on the physics of the scrape-off layer.

b) TFCX

Impurity control system components in TFCX will be exposed to high fluxes of energetic particles, resulting in high heat fluxes. However, limiter (or divertor plate) surface and first wall conditions are very uncertain for TFCX at this point. Therefore, it is necessary to consider a broad range of heat fluxes. Limiter (or divertor plate) heat fluxes are expected to be in the range between 1.0 and 10 MW/m^2 during equilbrium burn, with the design heat flux for the DCT-8 limiter being 7.0 MW/m^2. Disruption heat loads are expected to be in the range from 2.0 to 15 MJ/m^2 for durations from 0.5 to 20 ms. The nominal heat loads for the DCT-8 limiter are 6.0 MJ/m^2 during the 1-ms duration thermal quench phase of the plasma disruption, followed by 5.4 MJ/m^2 during the 10-ms duration current quench phase.

Erosion lifetime of the limiter (or divertor plate) surface is identified as a critical issue in impurity control system performance. The peak erosion rate of the DCT-8 limiter surface due to sputtering and disruption-induced melting and vaporization is about 10 mm per full-power-hour of operation. These conditions have led to the consideration of a duplex structure for the limiter (or divertor plate) surface with a protective surface tile, which is directly exposed to the plasma environment, attached to a cooled substrate. To increase the limiter (or divertor) lifetime, the protective surface tile thickness must be maximized, consistent with handling the high heat flux. Key factors affecting tile thickness are material properties (e.g., thermal conductivity and maximum allowable temperature) and thermal conductance between tile and substrate.

R&D efforts for developing high heat flux components are required that concentrate on characterizing, developing, and testing protective surface materials and tile-to-substrate attachment techniques. Candidate protective surface materials for TFCX include Be, BeO, C, SiC, Mo, Ta, and W. Candidate materials for cooled substrates are copper alloys and vanadium alloys.

Two promising approaches to developing thick armors of low-Z and refractory materials are: (1) plasma spray deposition and (2) bonding of monolithic overlays.

Plasma spraying involves high deposition rates, so that thick coatings (several mm) can be applied over large areas and complex shapes. The process can be used for many low-Z and refractory materials, and might be used for in-situ coating or repair. Sprayed coatings can be made with varying degrees of porosity or graded composition to improve adherence for a substrate and to improve thermal shock, fatigue and interfacial degradation properties. The relatively new technique of low pressure plasma spraying (LPPS) generally produces better coatings than conventional methods. There is presently underway a development effort to produce by LPPS thick coatings of SiC in a metal matrix. Initial results are promising, but the research is still at an early stage.

Where very thick armor must be applied to relatively simple geometric shapes, brazing or diffusion bonding of monolithic overlays as well as explosion bonding, can be advantageous. These techniques will provide better heat transfer than mechanical attachment schemes and reduce the likelihood of cracked pieces falling off in service. However, relatively little is known about brazing or bonding to many of the low-Z refractory ceramic carbides, borides and nitrides. A development effort is therefore required that emphasizes mechanical and thermal testing of joined materials.

c) Doublet III - Big Dee

Although present plans for Big Dee call for 1.5-second pulse operation at plasma currents of 5 MA with 14 MW of auxiliary heating, the design provides for possible upgrades. The vacuum vessel will have the capability of withstanding 20 MW of heating for 10-second operation; limiters and armor could be upgraded for larger pulses and higher power by modular addition.

To meet these requirements, the vessel is an all welded chamber made of Inconel 625. The walls are a corrugated sandwich construction which achieves high strength and high resistivity; in addition, the corrugations serve as passages for cooling water. Analysis of thermal and disruption stresses showed the need for a heat shield in local areas and in the mid-plane for long-pulse operation.

The limiters and armor fall into five categories: primary limiters, disruption limiters, neutral beam armor, divertor armor, and the thermal armor described above. The present design is adequate for 5 MA and 14 MW of auxiliary heating; upgrades would require modular additions.

The two primary limiters will each consist of a 0.32 m^2 surface of graphite tiles coated with TiC or SiC attached to a water-cooled Inconel blade. Disruption or back-up limiters will consist of conduction-cooled Inconel bars or tiles located poloidally around the vessel at several toroidal locations. Active cooling is not provided since these units absorb the full plasma power for very short times (milliseconds). The armor material (including the divertor) is basically the same as that for the limiters. Initially, the divertor tiles can be Inconel, as for the disruption limiters. As heating power and pulse lengths increase, a coated graphite tile structure is envisaged.

d) Mirror Devices

High heat flux components in upgrades of MFTF-B, such as MFTF − B + T and MFTF − α + T include the direct convertor, halo scraper, beam dumps and possibly the wall itself. The highest power densities encountered are several MW/m^2, i.e., in the same range as for tokamak limiters of this generation of devices. Special features that place demands on materials development include the need to survive ion irradiation of direct convertor elements without blistering and the need to control reflux of low energy neutrals from material surfaces.

e) ZT-H

Both pumped limiters and magnetic divertors are being considered for impurity-control and first-wall protection schemes for ZT-H. It is expected that ZT-H will achieve reactor-like average heat loads (2-4 MW/m^2) when operated in the high-current-density (~ 10 MA/m^2) regime, and hence a greater fraction of the first-wall will serve a limiter function. The RFP topology favors a toroidal array of poloidal pumped limiters; magnetic divertors based on the minority field lead to a preference for toroidal bundle divertors. As for the tokamak, sputter erosion is identified as a crucial issue for both the pumped limiter and magnetic divertor, with the partition between radiative versus particle transport losses being an important uncertainty in the design of both. Generally, materials solutions being proposed for DCT and TFCX should be applicable to the ZT-H. Tiles and coatings can be used for short-pulse ($<$ 1s) experiments, with an electrically-conducting first-wall shell being desirable to stabilize local MHD modes. To develop an experience base with a reactor track, coated copper alloy IVCs are preferred.

5. BEYOND 8 YEARS

a) Interface Related Effects

Components for first walls and impurity control systems in the Engineering Test Reactor (ETR) and commercial reactors will be exposed to high fluxes of energetic particles from the fusion plasma and fusion products. As reviewed elsewhere, it is possible to construct these critical components with duplex structures, which can handle high heat fluxes using active cooling techniques and high erosion rates using protective coating or claddings. These components will be tested in near-term devices such as TFTR, DIII-D, MFTF-B, TFCX, DCT, MARS, JET, and JT-60, and the best components, constructed with duplex structures or other advanced techniques, will be selected based on the results.

The severe reactor environments will include high duty cycle, long-pulse heat loads from plasma bombardments and neutron irradiation and, occasionally, transient heat loads from plasma disruptions or arcing. The heat fluxes from the high-cycle, long-pulse loads can be 5 MW/m^2 for several hundred seconds or longer; those from the transient loads, 100-500 MW/m^2 for 5-20 ms.

Actively cooled heat sinks or substrates, constructed of materials with high thermal conductance (such as copper alloys, vanadium alloys, and the austenitic stainless steels), will be used to handle the high-cycle, long-pulse heat loads. However, the thin walls of the heat sink would be rapidly eroded by sputtered plasma particles with energies of up to a few kilo-electron volts and by ablation caused by disruptions and arcing. Thus, the heat sink must be protected from direct exposure to the plasma by a coating/cladding surface. Materials for such a surface are those that are suitable for impurity control and have fulfilled the requirements for low sputtering and erosion. Candidate materials are B, Be, BeO, C, SiC, Ta, TiC, and W. However, most of these materials are not suitable for use as support structures. Consequently, a duplex structure of a protective coating or tile attached to a structural heat sink or substrate is considered for the impurity control system components.

In such a duplex structure, the protective coating can be applied to the heat sink by plasma spraying or chemical vapor depositions; tiles can be attached by diffusion bonding, brazing, or bolting. If tiles are bolted on, the interface thermal conductance between the tile and the heat sink will be relatively low, the tile temperature relatively high, and the thermal stresses and fatigues to the heat sink relatively low. Except for the limiter and divertor plates, which are subjected to extremely high heat fluxes, the bolt-on method could be practicable for first wall applications. With the other attachment methods, the high interface thermal conductance results in relatively low tile temperature, but the different thermal expansion coefficients of materials in the duplex structure may raise the thermal stresses in the heat sink to an unacceptable level. This drawback can be relieved by using a composite coating for the plasma spraying deposition, by reducing the width of the tile or coating with maximum allowable thickness, or by introducing a compliant layer in the bonding or brazing interface zone. The materials, fabrication techniques, and designs for these structures will be tested and evaluated for long component lifetime and low maintenance cost.

In the reactor environment, the lifetime of duplex structure components tends to be short because of the rapid erosion of the protective surface. Under high heat flux conditions, the maximum tile thickness is typically about 2 cm. The principal limiting factors are the maximum allowable tile temperature, the maximum allowable thermal stresses and fatigue, and the radiation effects (bulk swelling and conductivity reduction). In addition, the high duty cycle heat fluxes and the transient heat fluxes should be considered in the design and fabrication of these components. It may be possible to achieve the desired long lifetime using duplex structures or other novel techniques. Eventually, the impurity control system components for a fusion reactor should possess the following characteristics.

1. The component lifetime should be several years, or as long as that of the reactor.
2. The thermal conductivity and contact conductance of the materials should be high enough to handle the high heat fluxes of the reactor environment.
3. The variation of thermal expansion of the tile, interface, and heat sink should be minimal for all heat loads.
4. The thermal stresses should be evenly distributed over the heat sink by the interface material and fabrication technique.
5. The interface attachment should be strong enough to protect the tile or coating from the induced electromagnetic force or torque caused by plasma disruptions.
6. The reduction in contact thermal conductance caused by neutron irradiation should not make the component lifetime shorter than that of other reactor components.
7. The contact thermal conductance should not be affected by thermal shocks, high cyclic heat loads, or galvanic erosion during the component lifetime.
8. The design, interface materials, and fabrication methods should minimize debonding of the duplex structure, which may be caused by interface embrittlement, brittle fracture, fatigue failure, and radiation damage.
9. The normal peak stresses of the components should be within

safety limits that will prevent plastic or creep deformation, ductile bursts, and creep rupture.

Characterization of material properties and tests of fabrication methods will be carried out in near-term devices. Practical lifetime monitors for the coating or tile should also be developed. An additional concern is the damage caused by neutron irradiation to the material and bond properties of components in a reactor environment.

b) Neutron Effects

In addition to the severe thermal and erosion problems discussed above, the first wall and impurity control system components in ETR and commercial reactors will be subjected to intense neutron irradiation at levels far beyond those encountered in existing or near-term fusion devices. Expected irradiation damage to materials includes reduced thermal conductivity, reduced mechanical properties, bulk swelling, internal fracture, or interface embrittlement. Low-energy (< 1-MeV) neutrons in fission ractors are being used to study these effects. The basic physical, chemical, and mechanical properties obtained in operating environments will be used in selecting materials. Eventually, material selection will also be based on performance in the areas of high-temperature compatibility, component lifetime, ease of fabrication, tolerance to continuous thermal loads, and levels of residual radioactivity. These qualities will be evaluated during enhanced radiation tests in near-term devices.

The nonmetallic, low-Z materials now being considered for the protective surface next to the plasma are C, SiC, and BeO. The thermal conductivities of these materials drop to about one-half or one-quarter of their original values when they are exposed to the fluence levels of irradiation neutrons expected in fusion reactors. This loss of conductivity will raise the temperature of the tile or coating, which may increase the erosion rate and shorten the component lifetime. In addition, the radiation from high-energy neutrons will cause microcracking (from displacement damage in the bulk) and internal fractures (from irradiation distortion and swelling). Hence, these materials may not be suitable for use in ETR or commercial reactors. However, development and testing of these materials should be pursued with the objective of increased lifetimes.

Another low-Z material under consideration is Be, which has good thermal conductivity, high heat capacity, and a relatively high melting temperature. Its thermal conductivity is not sensitive to neutron irradiation, which is a favorable characteristic; however, helium accumulation in neutron-irradiated Be will result in bulk swelling and ductility reduction. Thus, the components lifetime may be shorter at high fluence levels. Factors in favor of Be as a protective surface material include its low self-sputtering coefficient, the ease of applying it by plasma spray coating, the fact that the original coating and redeposited coating have the same material properties, and its low activation/afterheat.

High-Z materials, such as tungsten and tantalum, are characterized by low D-T sputtering, high resistance to disruptions, and immunity to neutron irradiation. They may be considered for use in a Be coating to reduce melting damage by disruptions and arcing. Fabrication methods need to be developed, and the performance of new coatings or tiles must be tested experimentally.

c) Tritium Effects

There is a concern that for ETR or commercial reactor devices tritium permeation through the surface of first wall and impurity control components into the coolant will lead to contamination of the coolant to unacceptable levels. Tritium inventory in these advanced devices is a less serious issue because much larger inventories will exist in other components such as breeding blankets. Safety issues associated with large on-site tritium inventories and provisions for handling tritium-contaminated components must be addressed, regardless of the inventory in first wall and impurity control components.

Serious deficiencies in the materials data base make precise calculations of tritium permeation rates impossible at this time. Scoping studies have indicated that the range of uncertainty in the materials data base leads to coolant conditions that range from the case where no tritium processing is required up to a potential for large economic expenditures for tritium processing equipment. An R&D program for evaluating the magnitude of the permeation problem and for developing solutions to the problem, should it be significant, is required.

Several of the data base needs, covering areas such as hydrogen reflection and desorption from plasma side surfaces and material properties relevant to transport of tritium, are discussed in the Surface Physics Section and are not discussed here. To adequately define the magnitude of the tritium permeation problem and to develop solutions, significant progress must also be made in character[illegible]ing and defining: (1) production of trapping sites for tritium by neutron irradiation of the materia; (2) release of tritium through interfaces (e.g., from the material surface to the coolant); and (3) designs that act as tritium permeation barriers. These three issues are discussed below.

Neutron irradiation can produce voids, transmutation products, new phases, etc., that can act as trapping sites for tritium. Calculations indicate that these traps can often delay the time to permeation breakthrough beyond the component lifetime. Clearly, these trapping parameters must be verified experimentally with samples that have been exposed to neutron irradiation. In addition, potential dynamic effects such as γ-induced desorption, mobile hydrogen-defect complexes, etc., must be studied with in-pile permeation measurements.

The kinetics of tritium release into a water coolant must be identified, and the role of naturally occurring oxides as permeation barriers must be assessed. Research should also be directed towards the characterization of interfaces such as those between tile and substrate sections of the duplex structure required for impurity control components.

In instances where tritium permeation can be identified as a serious problem, solutions must be developed. Coatings at the coolant side could eliminate permeation in theory, but in practice the performance is often limited by defects in the coatings. A second potential solution would be to tailor the plasma side coating microstructure to enhance release back to the plasma. This effect would be accomplished, for example, by a plasma-sprayed porosity or by micro-cracking of the surface.

Until this R&D program is completed, meaningful calculations of tritium permeation cannot be carried out. There are too many unknown factors that could lead to a complete suppression of permeation during an ETR operating schedule. However, it must be stressed that for commercial reactors operated at more elevated temperatures, the existing data base can already be used in convincing arguments for a serious tritium permeation problem in many design scenarios.

d) Electromagnetic Interactions

During plasma disruptions in tokamak devices, currents are induced in the structures surrounding the plasma. The interaction of the magnetic fields present in the device with these forces depends upon the electromagnetic configuration of the components (e.g., continuous toroidal current path or components toroidally segmented into electrically isolated sections). These forces can lead to failure of the first wall or limiter/divertor structure and release of coolant, cracking of the surface tile or coating, debonding at the interface between the tile and substrate, or loss of the melt layer which may be formed by the heating of the surface during the disruption.

To characterize these forces, the electrical properties of the materials surrounding the plasma must be known. Although these properties are well established for the candidate materials, property changes caused by neutron irradiation may be important and need to be evaluated for ETR or commercial reactor applications.

Arcing caused by the presence of nearby surfaces at different electric potentials can occur during plasma disruptions in tokamaks or during operation of direct convertor components in advanced mirror devices. Arcing must either be avoided by designing appropriate electromagnetic configurations or be tolerated by providing designs and materials that are capable of withstanding arcing conditions. Conditions (e.g., voltages, surface conditions, particle densities, ...) that cause arcing between adjacent structures in a plasma environment should be investigated in an R&D program. Testing and analysis of specific electromagnetic configurations and materials under the expected electromagnetic conditions are required to establish feasible approaches to avoiding the arcing problem.

6. SUMMARY AND CONCLUSIONS

This review has concentrated on materials technology requirements for in-vessel components exposed to high fluxes of heat and particles. For each concept or design a development effort should:

1. identify potentially viable coating or cladding systems for specific applications;
2. establish the feasibility of fabricating these systems reliably by identifying key properties and relating them to process variables;
3. determine effects of hydrogen permeation, fatigue cycling, neutron damage, etc., as appropriate, on interface integrity in a comprehensive testing program;
4. demonstrate fabricability of full-scale components, including non-destructive evaluation.
5. perform integrated heat removal tests of prototype components as part of the component development process.

It appears that materials technology requirements can best be met by integrating materials development into the design process early enough for viable solutions to be developed and alternate approaches to be explored.