**Magnetic Fusion Energy
Plasma Interactive and High Heat Flux Components**

Volume II

**Technical Assessment of the
Critical Issues and Problem Areas in
High Heat Flux Materials & Component
Development**

June 1984

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Volume II

Technical Assessment of the
Critical Issues and Problem Areas in
High Heat Flux Materials & Component Development

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Table of Contents

	<u>Page</u>
FOREWORD.....	viii
SUMMARY.....	xi
I. SOURCE CONDITIONS	
I.1 Introduction.....	I-1
I.2 The Data Base on Present Devices.....	I-2
I.3 Theoretical Understanding of Operating Conditions.....	I-6
I.4 Predictions of Operating Conditions for Future Machines.....	I-7
I.5 Assessment of Operating Conditions and Future Needs.....	I-9
References for Chapter I.....	I-10
II. SYSTEMS INTEGRATION	
II.1 Introduction.....	II-1
II.2 HHF Components Location and Description.....	II-1
II.3 Systems Integration Considerations.....	II-20
II.4 Key Issues and R&D Needs.....	II-33
References for Chapter II.....	II-36
III. MATERIALS AND PROCESSES	
III.1 Operating Conditions.....	III-1
III.2 Plasma Side Materials.....	III-5
III.3 Heat Sink Materials.....	III-10
III.4 Interface/Attachment.....	III-19
III.5 Fabrication.....	III-22
III.6 Other Issues-Tritium Permeation.....	III-31
III.7 Materials and Processes Technology Needs.....	III-32
References for Chapter III.....	III-38

IV.	THERMAL HYDRAULICS	
IV.1	Introduction.....	IV-1
IV.2	Three High Heat Flux Heat Removal Techniques.....	IV-1
IV.3	Comparison of Three High Heat Flux Heat Removal Techniques.....	IV-10
IV.4	Conclusions and Recommendations.....	IV-11
	References for Chapter IV.....	IV-15
V.	THERMOMECHANICAL RESPONSE	
V.1	Thermal Response.....	V-1
V.2	Structural Response.....	V-9
V.3	Failure Analysis.....	V-15
V.4	Design Criteria.....	V-20
V.5	Thermomechanical Testing.....	V-20
V.6	R&D Needs.....	V-21
V.7	Critical Issues.....	V-26
	References for Chapter V.....	V-29
VI.	ELECTROMAGNETIC RESPONSE	
VI.1	Introduction to the Issues.....	VI-1
VI.2	Eddy Current Codes.....	VI-5
VI.3	Coupling of Eddy Currents to Thermomechanical Response.....	VI-6
VI.4	Conclusions and Recommendations.....	VI-10
	References for Chapter VI.....	VI-13
VII.	HIGH HEAT FLUX COMPONENT TEST FACILITIES.....	VII-1
VIII.	INSTRUMENTATION AND CONTROL	
VIII.1	Introduction.....	VIII-1
VIII.2	Measurement Requirements.....	VIII-2
VIII.3	Control Requirements.....	VIII-6
VIII.4	Operating Environment and Device Interfacing.....	VIII-11
VIII.5	Instrumentation and Control Techniques.....	VIII-14
VIII.6	Conclusions and Recommendations.....	VIII-18
	References for Chapter VIII.....	VIII-22

List of Figures

<u>Fig.</u>	<u>Title</u>	<u>Page</u>
I-1	Time scale on which current decays in various devices.....	I-5
II-1	Tokamak first wall location.....	II-3
II-2	Example of configuration for water-cooled first wall.....	II-3
II-3	Examples of armor for tokamak first wall.....	II-4
II-4	Positioning and startup limiters.....	II-5
II-5	Typical single-null poloidal divertor configuration.....	II-7
II-6	Inner divertor collector plate configuration.....	II-7
II-7	Limiter configuration (1982 U.S. INTOR reference design).....	II-8
II-8	Neutral Beam Engineering Test Facility (NBETF) ion dump.....	II-9
II-9	Actively cooled heat absorption panel for NBETF.....	II-9
II-10	TFTR accelerator source grid assembly.....	II-11
II-11	EBT-S ICRH antenna.....	II-11
II-12	MFTF-U: Beam and reactor layout.....	II-13
II-13	External beam dump configuration.....	II-13
II-14	Internal beam dump configuration.....	II-14
II-15	Typical beam dump design.....	II-14
II-16	End cell configuration.....	II-15
II-17	Inner halo configuration and analysis.....	II-15
II-18	Impact of availability in experimental fusion devices.....	II-31
II-19	Comparison of limiter maintainance approaches for STARFIRE and DEMO.....	II-31
IV-1	Predictions of the exit conditions...as a function of mass velocity.....	IV-4
IV-2	Predictions of the exit conditions...as a function of mass flow rate.....	IV-7
IV-3	Predictions of the pumping and the exit wall temperature as a function of velocity.....	IV-9

V-1	Combinations of surface heat flux and pulse length required to melt the surface of an uncooled, thermally thick material.....	V-2
V-2	Heat capacity figure-of-merit. Larger numbers imply that more heat can be absorbed in a given amount of time without melting.....	V-4
V-3	Maximum heat flux needed to melt the surface of a 1 cm thick plate cooled on its backside to 100°C during steady-state conditions.....	V-5
V-4	Amount of material melted during a 20 ms plasma disruption.....	V-7
V-5	Amount of material vaporized during a 20 ms plasma disruption.....	V-8
V-6	Thermal stress figure-of-merit for transient heating.....	V-11
V-7	Thermal stress figure-of-merit for steady-state heating.....	V-12
V-8	Triplex 3-layer construction of a high heat flux component.....	V-14
VI-1	Interrelation of the three responses (heating, electro-magnetic loading, and transient stresses) of an in-vessel component to plasma disruptions.....	VI-8

List of Tables

<u>Table</u>	<u>Title</u>	<u>Page</u>
I-1	Operating conditions of existing devices.....	I-3
I-2	Predicted operating conditions in future devices.....	I-8
II-1	Inner collector fluxes.....	II-17
II-2	Halo scraper fluxes.....	II-17
II-3	Comparison of activation for various HHFC materials.....	II-25
II-4	Estimated requirements for reliability of HHF components for tokamaks.....	II-27
II-5	Effect of limiter location and coolant choice on tritium breeding ratio.....	II-32
III-1	Impurity control parameters - tokamaks.....	III-2
III-2	Property values at 800 K for low-Z materials.....	III-8
III-3	Properties of high-Z materials at 500 K.....	III-8
III-4	Candidate heat sink copper alloys.....	III-12
III-5	Candidate refractory metal alloys.....	III-15
IV-1	Comparison of three high flux heat removal alternatives.....	IV-12
V-1	Time-independent failure modes.....	V-16
V-2	Time-dependent failure modes.....	V-17
V-3	Damage mechanisms.....	V-18
V-4	Critical issues for failure analysis.....	V-19
V-5	Summary of testing of actively cooled components.....	V-22-24
VI-1	Magnetic diffusivity, κ , and skin depths, δ , for a few metallic materials.....	VI-4
VI-2	Rectangular geometry eddy current computer programs.....	VI-7
VII-1	Dedicated electron test facilities.....	VII-2
VII-2	Dedicated ion test facilities.....	VII-3
VII-3	Neutral beam and RF test stand characteristics.....	VII-4

VIII-1	Measurement requirements and potential techniques for HHF components	VIII-3
VIII-2	Potential control loops involving HHF components.....	VIII-8
VIII-3	HHF component environment related concerns for I&C systems.....	VIII-13

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 HRFMCD 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, Chief
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Office of Fusion Energy
Office of Energy Research

FOREWORD

Technical Assessment of the Critical Issues and Problem Areas in High Heat Flux Materials and Component Development

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 Plasma Materials Interaction (PMI), which is closely linked to the High Heat Flux Materials and Component Development (HHFMCD).

As a reflection of increased programmatic emphasis on alternate concepts, a HHFMCD Task Group has been initiated and includes members of the mirror and compact toroid communities. The chairman of this Task Group is Dr. Mark Davis, Sandia National Laboratories (SNL), Albuquerque, New Mexico. In addition, the task group on PMI has been expanded under the chairmanship of Dr. Walter Bauer, Sandia National Laboratories (SNA), Livermore, California. 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 HHFMCD Technical Assessment is part of 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 (PMI)
- Special Purpose Materials (SPM)

In the intervening six years, significant progress has been made in each of these areas. In particular, the HHFMCD area has begun in response to critical near-term needs in TFTR and for TFEX. 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 HHFMCD systems which provide the integrating function for all the separate elements of the program.

Much of the success of the HHFMCD and PMI programs 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 MFTF-B.

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 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 present Technical Assessment, will be chaired by Dr. Bauer. In the area of HHFMCD, the Technical Assessment was chaired by Professor Mohamed Abdou, UCLA, and the Program Plan will be 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 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 (e.g., fuels) or laser fusion (e.g., optical materials and ultra high frequency pulsing or ramp rates.)

SUMMARY

A technical assessment of the critical issues and problem areas for high heat flux materials and components (HHFMC) in magnetic fusion devices shows these problems to be of critical importance for the successful operation of near-term fusion experiments and for the feasibility and attractiveness of long-term fusion reactors.

Both the challenge and the need for success in resolving the problems of HHFMC become greater as progress is made toward sustaining longer plasma burns and achieving higher power density. These problems relate to many fields of science and engineering: plasma physics, surface sciences, materials, thermal hydraulics, structural mechanics, electromagnetics and neutronics. Many of these problems require extensive long-term development and should receive attention now.

A number of subgroups were formed to assess the critical HHFMC issues along the following major lines: 1) source conditions, 2) systems integration, 3) materials and processes, 4) thermal hydraulics, 5) thermomechanical response, 6) electromagnetic response, 7) instrumentation and control, and 8) test facilities. The details of the technical assessment are presented in the following eight chapters. The primary technical issues and needs for each area are highlighted below.

A. Source Conditions

A.1 Characterisation of Plasma Edge Conditions

Knowledge of the plasma edge conditions is crucial to the design of high heat flux components (HHFC). Therefore, attention must be paid to obtaining data on the plasma edge conditions from present and future devices. In addition, continuing theoretical studies are required to uncover and understand the important physical processes.

A.2 Control of Plasma Edge Conditions

There are great incentives to attempt mechanisms for controlling the plasma edge conditions in order to provide a less harsh environment for opera-

tion of the HHFC. One particularly important goal is to achieve sufficiently low plasma temperature at the neutralizer plate in toroidal confinement devices so as to use high-Z materials with low erosion and high disruption-resistance. The degree of success may vary substantially with the impurity control scheme, e.g., pump limiters or divertors.

B. System Integration

Many of the issues identified in the various technical areas translate into operational, economic and safety system issues.

B.1 Lifetime of HHFC

Present estimates predict short lifetime for many of the HHFC, particularly the limiter and divertor plates in toroidal devices and the halo scrapers in tandem mirrors under certain operating conditions. Thus, the following areas need to be emphasized: a) exploring plasma edge conditions that minimize erosion, b) development of means to eliminate or minimize plasma disruptions and other off-normal conditions, c) experimental and theoretical efforts to provide reliable estimates of erosion and redeposition of plasma-side materials, and d) development of designs of HHFC and fusion devices that permit rapid replacement of components that have short life.

B.2 Tritium Contamination of Coolants in HHFC

There are large uncertainties in predicting tritium permeation rates into the coolants of HHFC. High tritium concentrations in coolants is a safety issue and tritium removal from water is costly. Therefore, experimental data on tritium permeation under realistic conditions is needed. In addition, the development of effective tritium barriers should be explored.

B.3 Interaction of Coolants in HHFC with Other Subsystems

The simultaneous use of water in HHFC, e.g., limiter or divertor, and liquid metals in other components such as the blanket is presently perceived to entail significant safety risk. Therefore, experimental and analytical efforts are required to: a) identify viable coolants for HHFC other than water, b) determine the probabilities and consequences of accidents that involve water in HHFC and liquid metals in other components.

B.4 Effect of HHFC on Achieving Fuel Self-Sufficiency in DT Fusion Devices

HHFC, particularly the impurity control and exhaust systems (divertors and limiters), will have a substantial impact on the ability of fusion devices operated on the DT cycle to achieve fuel self sufficiency. This impact falls into two areas: a) reducing the achievable tritium breeding ratio by occupying relatively large volume of the breeding blanket and parasitic absorption of neutrons and b) increasing the required breeding ratio if the tritium recycling into the plasma is reduced.

There are substantial differences between the limiter and divertors and the various design concepts for both types of impurity control and exhaust systems. Therefore, substantial effort in the physic and engineering is required to develop impurity control and exhaust concepts that minimize the physical space requirements and maximize the tritium fractional burn-up in the plasma.

C. Materials and Processes

The critical issues for high heat flux materials and processes can be conveniently divided into near-term and long-term issues.

C.1 Near Term Issues

C.1.1 Plasma-Side Materials

There are several surface-related critical issues for plasma-side materials. In the case of graphite, the key issue is the impact of chemical sputtering on plasma operation. For beryllium, the critical issue is the response of the material to disruptions, particularly melt layer formation and erosion. For the compounds, such as SiC and BeO, the critical issue is the possible change in surface composition and properties with preferential sputtering. In the case of high-Z materials, the point where the self-sputtering coefficient exceeds unity is the most important issue. A critical issue associated with all plasma-side materials is the lack of a comprehensive bulk property database.

C.1.2 Heat-Sink Materials

The major issue associated with heat-sink materials is the lack of a comprehensive bulk property data base. Copper alloys are the leading candidates for heat-sink materials in the near term. More information is needed in the areas of elevated temperature tensile properties and fatigue/crack growth behavior before a reference alloy can be selected.

C.1.3 Duplex Structures and Attachment

A major need for HHFC is the development of fabrication and bonding techniques for structures composed of both plasma-side and heat-sink materials. Specific bonding methods have not yet been developed and the properties of the bonds are largely unknown. A secondary issue is the possible impact of bonding fabrication on the bulk properties of the heat sink. The temperature required for bonding may be in the range where the mechanical properties of copper alloys are seriously degraded.

C.2 Long Term Issues

For long-term applications, HHFC will be subjected to moderate to high neutron fluences. The critical issue with a radiation environment is whether the HHFC are sufficiently resistant to radiation damage to operate for the desired lifetime. The materials used for HHFC are considered to be new to the study of radiation damage effects, so there is very little data available. An extensive program will ultimately be required to examine the effects of radiation on the thermophysical properties, mechanical properties, and the dimensional stability of materials in a radiation environment.

The use of refractory metals in HHFC appears desirable for some operating conditions. The major issue associated with these metals is the lack of an adequate baseline and fabrication data base. The needs include the acquisition of data on all relevant bulk properties in both the unirradiated and irradiated conditions. In addition, the corrosion behavior of water and liquid metals needs to be understood. There is a possibility that refractory metals will be used in near-term devices, such as MFTF- α T. If this is the case, the development of refractory metals will become a near-term issue.

C.3 R&D Programs

The critical issues can be addressed with five major R&D programs, as follows.

C.3.1 Development and Selection of Plasma Side Materials

The surface and bulk properties of candidate plasma-side materials should be examined for temperatures ranging from room temperature to $\sim 1000^{\circ}\text{C}$. Since fabrication processes can significantly influence the properties, the materials should be tested in several fabricated forms. The goal of this program is the selection of the best plasma-side material for near-term applications.

C.3.2 Development and Selection of Copper Heat Sink Alloys

The bulk properties of candidate copper alloys will be determined for temperatures ranging from room temperature to $\sim 400^{\circ}\text{C}$. The most important properties to be investigated are the elevated temperature tensile properties and the fatigue/crack growth behavior. The thermo-mechanical treatment (TMT) will significantly affect the mechanical properties so that the properties should be determined for various TMT's. The goal of the program is the selection of a reference copper alloy for near-term applications.

C.3.3 Fabrication and Bond Development

Work should be undertaken to examine the fabrication of plasma side materials, bonds, and heat-sink materials. The properties of duplex structures (plasma-side material bonded to heat sink) are a particular concern. Duplex structures should be fabricated and the thermophysical and mechanical properties of the bonds should be measured. The goal of this program is to identify the optimum fabrication procedure for HHFC for near-term applications.

C.3.4 Irradiation Studies of High Heat Flux Materials

This program would add irradiation damage considerations to the other properties considered in the above programs. The neutron damage level of interest are ~ 10 - 50 dpa, and the temperatures of interest are room temperature to $\sim 400^{\circ}\text{C}$. The goal of this program is the selection of reference

plasma-side and heat-sink materials for long term applications.

C.3.5 Selection and Development of Refractory Metals for HEFC

Refractory metals are at an early stage of development for fusion applications. Work should be performed to establish fabrication procedures and a baseline data base for these metals. The metals of interest are tungsten, tantalum, vanadium, niobium, and molybdenum alloys. Data is needed in the areas of thermophysical properties, mechanical properties, and irradiation effects. The temperatures of interest are room temperature to $\sim 1000^{\circ}\text{C}$, and the fluences of interest are 10-50 dpa.

D. Thermal-Hydraulics

D.1 Issues

The key issues in thermal hydraulics are:

D.1.1 Critical Heat Flux

Evaluation of subcooled flow boiling (SFB) critical heat flux between 1.0 and 20 MW/m^2 for large surface area (0.1 to 50 m^2) fusion components with large L/D ratio (50 to 500) coolant channels. The lower range of the heat flux ($1\text{--}5 \text{ MW/m}^2$) appears to be the most likely for in-vessel components in future devices if present thermomechanical constraints associated with high erosion rates from the plasma-side materials are not eliminated. The higher heat flux range is important for concept improvement. Some components such as beam dumps operate at the high heat flux range but success in this area has been achieved because their operating requirements are significantly less demanding than those of the in-vessel components.

D.1.2 Stability Criteria

Establishing the thermal-hydraulic stability criteria of multiple, interconnected, and subchanneled coolant conduits is essential to near-term component design.

D.2 R&D

The prioritized engineering R&D needs for HHF SFB heat removal are:

D.2.1 Develop unconditionally stable SFB CHF and pressure loss data for single and multiple coolant channels with L/D ratios between 50 and 500 and with heat fluxes between 1.0 and 20 MW/m². Determine the quantitative effects of surfactant additives, binary mixtures, swirl flow, and helical subchannels relative to stable axial flow data.

D.2.2 Perform thermal-hydraulic stability studies to examine, e.g., the Ledinegg, flow pattern transition, parallel channel, and condensation induced instabilities. The fusion component flow-channel design requires such an analysis.

D.2.3 The mechanistic understanding of conditions leading to SFB CHF should be increased by: a) Developing engineering models for SFB CHF characterization which are based on first principles and which relate the applied heat flux to the bulk and local SFB parameters; b) performing flow visualizations to observe both hydrodynamic and thermal characteristics which will lead to an overall understanding of SFB CHF and provide guidance for engineering model development.

E. Thermomechanical Response

E.1 Existing Plasma Experiments

No critical issues were identified for existing machines with respect to thermomechanical response.

E.2 Near Term Fusion Experiments

E.2.1 Bond Integrity

The structural integrity of the attachment interface between the plasma-side armor material and the actively cooled heat sink must be demonstrated. This requires thermal fatigue testing of prototype components on ASURF and PMTF, generic component testing on ESURF and EBTF, bond strength and thermal

conductance data, and 2-D finite element thermal and stress analyses.

E.2.2 Melt Layer Stability

Loss of the melt layer due to eddy current forces during a plasma disruption must be avoided to prevent excessive surface erosion. Laboratory experiments and computer simulations must be performed to assess the severity of this problem and to facilitate design changes that may eliminate the effect.

E.2.3 Fatigue Crack Growth

Crack growth leading to a coolant leak or failure of the plasma-side material must be prevented. Basic data on fatigue crack growth rates in candidate materials and across interfaces is required for computer simulations. Thermal fatigue testing of intentionally flawed prototype components should be performed to verify the fracture mechanics analysis.

E.3 Intermediate Fusion Devices

E.3.1 Bond Integrity

Neutron radiation damage will degrade the bond between armor and heat sink due to (1) differential swelling stress build-up, (2) impurity transport to the interface, and (3) embrittlement. Long-term irradiation must be performed in FFTF, EBR-II and FMIT on bonded specimens to develop appropriate damage models that can be used in computer simulations of bond failures.

E.3.2 Erosion

Surface erosion is a critical lifetime issue that directly affects the thermomechanical response.

E.3.3 Embrittlement

Neutron-induced embrittlement of candidate high heat flux materials must be understood to avoid catastrophic fractures. Fracture toughness measurements on irradiated specimens will allow the use of fracture mechanics to provide an adequate safety margin.

E.4 DEMO/Commercial

E.4.1 Creep/Fatigue

The interactions between creep and fatigue damage during neutron irradiation may accelerate crack growth rates and lead to premature failure. More experimental data and modeling is required.

E.4.2 Creep Rupture

For long-pulse or steady-state fusion reactors, creep rupture may be the life-limiting failure mode, especially under the influence of radiation damage. More experimental data and modeling is required.

E.4.3 Swelling

Low swelling high heat flux materials must be developed for long-term applications because excessive deformations may prevent easy removal for maintenance and repair.

F. Electromagnetic Response

F.1 Issues

In order of priority the critical issues regarding the electromagnetic response of HHFC are as follows:

F.1.1 Present eddy-current codes have a limited capability to evaluate eddy-currents in real, non-idealized geometries for high-heat flux components. In addition, complicated heterogeneous and composite materials cannot presently be treated adequately.

F.1.2 The overall coupling of eddy-currents in all in-vessel components and their feedback on the motion of the disrupting plasma is not well understood, although it is believed to be significant.

F.1.3 The formation and stability of melt layers on high-heat flux components during a disruption remains a major uncertainty.

F.2 R&D

Accordingly, the following R&D requirements have been identified.

F.2.1 A 3-D eddy-current code needs to be developed as a component design tool. This code must provide the capabilities of treating composite structures consisting of different materials with temperature-dependent electrical conductivities.

F.2.2 This code should be integrated with a general purpose, finite-element stress analysis code; the latter should treat magnetic damping.

F.2.3 Simulation experiments on electromagnetic forces and on melt-layer stability are essential in order to test and confirm the above codes and models of the melt-layer stability.

G. Instrumentation and Control

G.1 Key Issues

The key issues have been identified as follows.

G.1.1 Instrumentation of the "Plasma Side" of HHFC components

This is an area in need of some development, primarily due to the severity of the environment in which this portion of the component must operate and be instrumented, and to the severity of the impact of any undesired performance or failure on the plasma and hence device operation. Non-contact, in-situ measurement techniques are desirable for instrumenting the plasma side of HHFC due to advantages of operation in the presence of severe E-M and radiation fields, and the response speeds for control loop application.

G.1.2 Definition of Control Loops and Algorithms Involving HHFC

Control logic for HHFC needs to be generated for each device during its conceptual design. Control loops and algorithms are not presently well defined for future devices at any stage. It is important to identify the reactor control systems so that they can be fully integrated into the total design. If engineering data for control algorithm generation is not taken on

devices like TFTR, the lack of such data could become a critical problem.

G.1.3 Radiation Hardening of Instrumentation and Control Systems

Upgrading is necessary to provide radiation hardened data acquisition and control systems and interfaces which can perform reliably in the severe electromagnetic and radiation fields of future devices. Such items as optical windows, lenses and mirrors, fiber optics, and electronics are particularly susceptible to radiation environments, and existing devices are not generally designed for operation in high neutron fluences.

G.2 R&D

The associated R&D needs are listed below.

G.2.1 In-Situ Instrumentation

(a) Development of measurement techniques for in-situ erosion and redeposition rates and the spatial distribution of these rates. This development effort must address the integration of the measurement system with the operating fusion device.

(b) Development of improved techniques for non-contact surface temperature measurements in the presence of a D-T plasma. This may involve upgrading of IR systems to reduce or eliminate the impact of emissivity changes of the object (primarily due to surface condition changes, although wavelength and temperature dependence need to be addressed) and transmission/reflectance changes of optical components (i.e., redeposition coating of windows and radiation damage to windows/mirrors/photosensitive devices. This development effort must also address full integration with the intended fusion device.

G.2.2 Control

(a) The ability to test HHFC control schemes should be incorporated into future device designs so that some of these techniques can be incorporated into the standard control logic of devices such as ETR and DEMO.

This should include both control of HHFC parameters and contributions of HHFC to the total reactor control scheme, which involve long term drift compensation of plasma parameters, gross device parameter adjustment, and

interlock/alarm functions.

(b) Development of control algorithms for device subsystem control will rely heavily on results from existing devices. Existing devices and their follow-ons should include tests which provide an engineering data base for control algorithm generation.

G.2.3 Radiation Hardening

(a) Define and perform experiments which will provide a data base for radiation effects (fusion-specific radiation environment) on HHF instrumentation components.

(b) Place more emphasis in device designs which address I&C radiation hardening (includes I&C design, placement and shielding).

I. SOURCE CONDITIONS

I.1 Introduction

The design of components to be used in high heat flux regions of fusion devices depends critically upon the operating conditions those devices will experience. These operating conditions include not only the heat and particle fluxes but also the electric and magnetic environment around the component. As the input power and pulse lengths of devices have increased, designers have been forced to pay more attention to operating conditions. This has fostered more diagnosis of the plasma edge and resulted in somewhat better understanding of the plasma edge. There remains much to be done before the level of understanding is adequate to provide the detailed understanding necessary to design components for advanced devices, but there is time to obtain this understanding.

There are three major groups of components for which operating conditions must be defined. The first group are those components which are in direct contact with the plasma, such as limiters or divertor plates in tokamaks and halo scrapers, collector plates, and direct converters in mirror machines. The second group are those components associated with auxiliary heating, such as beam dumps, wall armor, grids, and RF windows. Finally, there are first wall components. While some components fall into more than one group, there is always a primary function that is in one of the above groups. The auxiliary heating group has the best defined operating conditions. Those components in direct contact with the plasma have the greatest uncertainty in the operating conditions. The operating conditions of the first wall are well defined theoretically, but there is no direct experimental support for the calculations because reactor relevant conditions have not yet been achieved. The input to the codes change as more is learned about plasma conditions, resulting in a good deal of uncertainty in the first wall conditions.

This chapter will cover four major topics in regard to the environment of the three groups of components. Section 2 will address the present state of the data base that exists on present machines. Section 3 will briefly discuss the theoretical understanding of that data base. Section 4 will look at predicted conditions for future devices. Finally, Section 5 will discuss what is needed to improve the understanding of operating conditions and, hence, reduce the uncertainty in the predictions for future devices.

I.2 The Data Base on Present Devices

This section contains a brief description of the diagnostics used to measure properties of the edge plasma and the performance of components used in high heat flux areas. The typical operating conditions derived from these diagnostics are then summarized. The section concludes with a discussion of the quality of this data.

I.2.1 Diagnostic Techniques

Diagnostics that are used to determine conditions in the plasma edge have improved greatly in the past few years. The traditional Langmuire Probe has been improved by the use of better equipment and new techniques such as triple probes. The use of Langmuire Probes was recently reviewed by Staib⁽¹⁾. Probes are in use on Alcator-C⁽²⁾, PLT⁽³⁾, PDX⁽⁴⁾, and TFTR⁽⁵⁾. Those traditional probes have been supplemented by calorimeter probes⁽⁶⁾ which directly measure the plasma power density in the edge. Power flow to limiters and divertor plates, and energy scrape-off lengths have been inferred from the time dependence and spatial distribution of surface temperature measured using infrared thermography⁽⁸⁻¹¹⁾. These infrared techniques are also applicable to beam dumps and similar components. Calorimetry done by water flow and ΔT is done on many devices to determine total energy deposited on a device. Thermocouples and thermistors are also widely used to measure operating temperatures and to calibrate the infrared diagnostics. Hydrogen recycling and impurity transport in the edge region are important to operating conditions, but a discussion of all the diagnostics associated with these phenomena is beyond the scope of this chapter. The interested reader is referred to the excellent review article by McCracken and Stott⁽¹²⁾.

I.2.2 Operating Conditions in Present Devices

The operating conditions that are deduced from the data provided by the diagnostics mentioned above are listed in Table I-1. In several cases, the figures are only estimates made by those working on the particular device. The most striking feature of the table is the wide range of values. A significant number of these variations are due to the differences in geometry in the various machines. The geometrical differences are both in the size and

Table I-1. Operating Conditions of Existing Devices

Machine	Component	Peak Heat Flux	Pulse Length	Reference
Alcator-C	Limiter	$\sim 5 \text{ kW/cm}^2$	0.5 s	13
	First Wall	10 W/cm^2	0.4 s	
	Disruption	$80\text{--}500 \text{ kW/cm}^2$	200-300 s	
Doublet-III	Limiter	$0.5\text{--}2.0 \text{ kW/cm}^2$	0.3-1.0 s	14
	Divertor Plates	?		
	Disruption	$50\text{--}75 \text{ kW/cm}^2$	20-100 μs	
ISX-B	Limiter	$1.0\text{--}10.0 \text{ kW/cm}^2$	0.2-0.3 s	15
PLT	Limiter	$2.5\text{--}9.0 \text{ kW/cm}^2$	0.15-0.8 s	
PDX	Limiter	$0.2\text{--}3.0 \text{ kW/cm}^2$	0.3 s	17
	First Wall	6 W/cm^2	0.3 s	
	Divertor Plates	$9.75\text{--}1.0 \text{ kW/cm}^2$	0.3 s	
	Disruption	10 x normal flux	?	
TMX	Edge Scraper	$\sim 0.1 \text{ kW/cm}^2$	0.02 s	18
	Beam Catcher	?		
	First Wall	?		

shape of the component as well as changes in plasma parameters resulting from different configurations, e.g., the connection length is shorter for an axisymmetric limiter than it is for a simple rail limiter which results in different scrape-off lengths and different power loads. Other variations are due to different magnetic fields, auxiliary heating power, and impurities. While many aspects of the plasma edge are understood, there is not a complete understanding of the edge physics. The loads on the "first wall" are determined from bolometric measurements of the radiated power.

There is an additional line in Table I-1 for the tokamak devices dealing with the conditions during fault conditions such as disruptions. These fault conditions are characterized by two phases: a thermal quench phase during which the plasma cools, followed by a current decay phase during which the plasma current disappears. The first phase results in large heat loads to the limiter or divertor plates. The second phase results in large electromagnetic forces on internal components. The time scale on which the current decays in various devices is shown in Fig. I-1. These fault conditions are most common when plasma conditions are being pushed into new regimes. They are, therefore, to be expected for the beginning phase of any device that explores new plasma conditions.

The operating conditions for auxiliary heating components are also shown in Table I-1. The variations in these numbers are completely explained by differences in geometry and/or source parameters. There are some small differences that are caused by plasma parameters, but they are not a major effect on the operating conditions.

I.2.3 Discussion

While there is a significant amount of data on the plasma edge at this time, there are several shortcomings. One is the lack of a complete set of data about all processes in the plasma edge. This results in an incomplete understanding of the physics of the edge plasma. We also do not know how the edge parameters will scale as conditions become more reactor-like. This would probably even be true if our understanding of present edge conditions were complete. This means that future devices will have to be designed to accommodate a range of operating conditions. Thus, it is of critical importance to continue to study the plasma edge in order to reduce this

Figure I-1. Time scale on which current decays in various devices.

uncertainty. Since plasma conditions will continue to be pushed into new regimes, the fault conditions in tokamaks will also have to be considered as discussed above.

The heat loads on beam dumps, wall armor, and other auxiliary heating components is very well understood at this time. The data that we have now also extrapolates very well to future machines. Most of the work that must be done for such components is to improve pulse lengths, develop new sources, and account for differences in machine geometry. Even for new sources, the operating conditions should remain well defined.

As mentioned above, the operating conditions of the first wall in current devices are not necessarily representative of reactor devices. This problem is only solved by making more reactor-relevant devices such as MFTF, TFTR, or TFCK.

1.3 Theoretical Understanding of Operating Conditions

The extrapolation of the data on edge plasma conditions in current machines to future devices depends strongly on a theoretical understanding of the physics behind such conditions. In a large part, the certainty of the operating conditions for beam dumps, wall armor, etc., is due to the theoretical understanding of the physics involved in the heat source and power transport. This section will examine the models used to understand the edge plasma and their application to predicting conditions for high heat flux components.

The plasma edge involves a complex interaction between several important effects. The transport of particles and energy in the edge depends on the electron and ion temperature profiles, magnetic field, neutral density profile, recycling coefficients, plasma thermal conductivity, particle diffusivity, impurity species and density, impurity generation rates, and atomic and molecular cross sections. There are several sophisticated computer codes that can solve the plasma transport equations including all of the above effects with many options. One such code that includes the edge is the BALDUR code⁽¹⁹⁾. Most of these codes are one or one-and-one-half dimensional. They can partially duplicate such effects as the "H-mode." There are several two-dimensional codes^(20,21) that have been used to describe plasma transport into divertors and pump limiters, but they are not as rigorous as the one-

dimensional codes in their treatment of the physical processes in the edge. For the FED/INTOR studies, several simple one-dimensional codes^(22,23) were used to predict plasma conditions. Most of these codes have been compared to data taken from present machines and with each other. In all of these codes, assumptions must be made to make the problem tractable or to model unknown or poorly understood effects. In all cases, the nature of the assumptions strongly influences the values of the plasma transport coefficients that explain the data. Therein lies the primary source of uncertainty in predicting operating conditions in future machines. There are also a vast array of codes to calculate specific parts of the edge problem such as the neutral density profile or sputtering yield. In many cases, these specific codes are used by the large codes. The charge exchange flux to the first wall is also predicted by the same codes. This is very important for first wall designs.

The heat flux to beam armor or similar components is calculated using codes that include the source grid geometry, electric and magnetic fields, space charge, gas pressure, the presence of scrapers, and the plasma attenuation. These codes are very specific to a given source type. They have been very carefully calibrated and give excellent results.

I.4 Predictions of Operating Conditions for Future Machines

The codes discussed above have been used to project operating conditions for devices that are just beginning operation, or under construction, as well as longer term devices such as TFCX, INTOR, or MARS. Table I-2 lists the results of such predictions. This table does not list every device which has been considered, but it does contain a representative sample of the general classes of devices that have been studied. In general, those machines which are the closest to being constructed (or actually under construction) have received the most careful consideration in terms of evaluating the uncertainties and are also the closest to present edge conditions. Predictions for long-term devices like DEMO are not adequate to do a detailed design of such a device, but data from intermediate machines will solve that problem. It is interesting to note that most of the peak heat fluxes for components in contact with the plasma are below 5 MW/m^2 . This is in part due to the perception that this is an acceptable steady-state heat flux. Devices

Table I-2. Predicted Operating Conditions in Future Devices

Device/Component	Normal Operation Heat Flux (kW/cm ²)			Duration (s)	Off-Normal Heat Flux (kW/cm ²)	Duration
	Lowest	Typical	Highest			
<u>TFTR</u> (24)						
Rail Limiter	2.0	4.5	7.0	0.25-1.0	100	0.002- 0.005
Bumper Limiter	0.3	0.65	1.0	1.5	100	0.002- 0.005
Beam Armor	0.4	0.6	0.8	1.5	5.6	0.01
First Wall	—	0.03	—	1.5	—	—
<u>MFTF-B</u> (25)						
Beam Dumps	2.0	3.0	4.0	0.5-30	—	—
End Domes	0.03	0.25	0.45	0.5-30	—	—
Halo Scraper	—	0.2	—	0.5-30	—	—
<u>TFCX</u> (20)						
Limiter, Divertor	0.2	0.35	0.5	100	—	—
First Wall						
<u>INTOR</u> (27)						
Limiter, Divertor	0.1	0.3	0.5	100	10	0.02
First Wall	0.01	0.05	0.1	100		
<u>Alcator-DCT</u> (30)						
Limiter	0.2	0.35	0.5	90	20-50	0.001
<u>RFP</u> (31)						
Limiter	0.2	1.8	3.9	0.05-0.25	10	0.003
First Wall	0.05	0.3	0.6			
<u>OHTE</u> (31)						
Limiter		0.6				
First Wall		0.2				
<u>RIGGATRON</u> (31)						
Limiter	2.0	3.5	5.0			
First Wall		0.5				
<u>EBT</u> (31)						
Limiter		0.25				
<u>DEMO</u> (28)						
Limiter, Divertor	0.5	0.75	1.0	Steady- State		
First Wall	0.05	0.075	0.1			
<u>MARS</u> (29)						
Halo Scraper	0.2	0.25	0.3	Steady- State		
Beam Armor	0.19	0.35	0.4			
First Wall	—	—	—			

having a shorter planned lifetime, such as the compact tori, have the highest predicted heat fluxes. This is the result of a trade-off between lifetime, peak, heat flux, and machine size. The heat fluxes for beam dumps are higher but the uncertainties are smaller and such devices are not in direct contact with the plasma. Impurity levels have a strong influence on operating conditions, and impurity control is, thus, very important.

I.5 Assessment of Operating Conditions and Future Needs

The following conclusions can be reached from considering the present state of knowledge concerning operating conditions.

- Recent developments in and attention to edge plasma diagnostics have fostered a better understanding of the plasma edge.
- Theoretical understanding of the edge is keeping pace with the data base.
- Infrared temperature diagnostics are shedding light on the interaction of the plasma with high heat flux components.
- Operating conditions for auxiliary heating components are well understood.

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CHAPTER I

SOURCE CONDITIONS

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II. SYSTEMS INTEGRATION

II.1 Introduction

For purposes of discussing systems integration aspects of high heat flux components (HHFC), magnetic confinement fusion reactor concepts will be separated into three categories: two mainline concepts--tokamaks and tandem mirror reactors--and a field of alternative concepts. As illustrated by the parameter tables in Chapter I, the types of HHFC needed and the requirements imposed on those components vary widely among the three reactor categories, and within each category as well when the range of needs from present-day experimental devices through power reactors are considered.

High heat flux components are required at many locations throughout fusion experimental devices and power reactors to perform a variety of functions. Most components are located in areas with limited access and must operate for long periods without failure. The number of cycles and heat flux level strongly affects the time between required replacements. Materials choices are driven by restraints on plasma contamination, radiation damage, and impinging particle flux and energy levels. All components must operate in a high quality vacuum which implies very low leak rates ($\lesssim 10^{-8}$ std cc/s). Additional factors affecting the design of each component are energy recovery requirements, method of structural support, and design integration requirements for other reactor subsystems.

Although there are a large number of different HHFC components required for each reactor type, only the more important ones will be discussed in the following sections. The issues and R&D needs indicated for these components are expected to be valid for the remaining components as well.

The remainder of this chapter concerns the description and location of HHFC for each of the three reactor types (Sec. II.2), discussion of the most important systems integration considerations affecting HHFC designs (Sec. II.3), and a brief listing of key issues and related R&D needs (Sec. II.4).

II.2 HHF Components Location and Description

The environment for plasma-facing HHFC in tokamaks is relatively severe. For most projected plasma conditions, surface heat fluxes and particle fluxes

will be high. Solutions are needed for these problems that will also be satisfactory from the standpoints of economics and safety. The problems for specific HHFC are discussed below.

II.2.1.1 First Wall

The first wall of a tokamak is exposed directly to the plasma. The basic first wall is shown in Fig. II-1 for both a divertor and pumped limiter configuration.

Active cooling of tokamak first walls will be required for devices beyond TFTR. Water coolant has often been proposed⁽¹⁻³⁾ for these devices. As an example, Fig. II-2 illustrates typical structural configurations proposed for FED. In present devices such as TFTR and Doublet III, the first wall has generally been fabricated as an entity separate from any shielding structure. In future devices where tritium breeding is required, it is likely that the first wall will be made as an integral part of the tritium breeding blanket to reduce first wall thickness and decrease neutron capture. This integration results in a number of important design constraints for the first wall, e.g. added restraint to thermal expansion.

There are often other HHFC attached to the first wall, and localized areas requiring special attention as discussed below.

Protective Armor - Protective armor is used to protect the first wall against damage during plasma disruptions or from neutral beam shine-through. The TFTR utilizes bumper limiters at the plasma interior to absorb the energy of a plasma perturbation or disruption. The protective plates at the plasma exterior are used to protect against neutral beam shine-through. These locations are indicated in Fig. II-3 for TFTR in a cutaway view of the vacuum vessel. The design utilizes TiC coated graphite tiles mounted to a water-cooled support panel. Some typical armor attachment methods are shown in Fig. II-3.

Positioning Limiter - In near term machines such as TFTR, a positioning limiter is used to locate the plasma. In TFTR⁽⁴⁾ this is accomplished by the limiter shown in Fig. II-4(a). The conceptual design for a moveable startup limiter for FED is shown in Fig. II-4(b). This limiter generally receives the most severe heat loading of any HHFC in the device.

Fig. II-1. Tokamak first wall location.

Fig. II-2. Example of configuration for water-cooled first wall.

TFTR PROTECTIVE ARMOR LOCATION

PASSIVELY COOLED ARMOR TILE

Fig. II-3. Examples of armor for tokamak first wall.

(a) Positioning limiters in TFTR.

(b) Movable startup limiter for FED (conceptual design).

Fig. II-4. Positioning and startup limiters.

CX Areas - First wall areas near the divertor and pumped limiter are subjected to increased erosion and higher surface heat fluxes due to a higher concentration of charge exchange (CX) neutrals. These areas are noted in Fig. II-1.

II.2.1.2 Impurity Control

Pumped limiters and poloidal divertors (single- and double-null) are being considered for impurity control. Each system requires high heat flux components for removing the energy from the impurities.

A typical single-null divertor design is shown in Fig. II-5. It consists of an inner plate and outer plate that are aligned to intersect with magnetic field lines from the plasma. The angle of incidence determines the heat load the plate must absorb. This angle is limited by the ability to tailor the magnetic field. The plate is subject to erosion from impingement by energetic particles from the plasma and generally incorporates a sacrificial protective surface (Fig. II-6). This surface is generally conceived to be a low Z material such as Be or C. The substrate is generally copper, vanadium or other material with relatively high thermal conductivity. The coolant is typically water.

A pumped limiter design for INTOR is shown in Fig. II-7. This particular design is located at the bottom of the plasma chamber and consists of a double-edged heat absorption surface. The leading edge is moved away from the plasma to reduce heat load and limit the throughput of the vacuum system. Materials considered for the limiter are generally the same as for the divertor plate.

II.2.1.3 Beam Dumps

Neutral beam heating of the plasma requires that ions be stripped from the beam and absorbed in a vacuum pumping system. The ions strike a surface that absorbs their energy. This surface must be actively cooled. The location of an ion dump in a neutral beam test facility is shown in Fig. II-8. Designs that have been used for this heat absorbing surface include swirl tubes and flat plates. The flat plate design shown in Fig. II-9 can absorb 3000 W/cm^2 continuously.

Fig. II-5. Typical single-null poloidal divertor configuration.

Fig. II-6. Inner divertor collector plate configuration.

Fig. II-7. Limiter configuration (1982 U.S. INTOR reference design).

Fig. II-8. Neutral Beam Engineering Test Facility (NBETF) ion dump.

Fig. II-9. Actively cooled heat absorption panel for NBETF.

II.2.1.4 Ion Source Grids

Neutral beam ion sources utilize a grid rail system to direct the particles. These grids must absorb large heat fluxes. Although current neutral beams are operated in a pulsed mode, beams in the future must be designed for continuous operation. The accelerator source grid used in the TFTR ion sources is shown in Fig. II-10.

II.2.1.5 RF Antennas

RF antennas must be located close to the plasma for effective coupling. Thus the antenna system must absorb energy levels comparable to those for a limiter. A system used for EBT-S is shown in Fig. II-11. It consists of a loop antenna radiating element and a Faraday shield. The loop antenna must absorb $\sim 50 \text{ W/cm}^2$ but the Faraday shield is exposed directly to the plasma.

II.2.2 Tandem Mirror Reactors

The tandem mirror is an inherently steady-state device with an open magnetic configuration. These two characteristics have an influence on the requirements for high heat flux components, their location in the device and on their design.

II.2.2.1 Beam Dumps

Beam dump requirements for tandem mirrors are predicted to decrease as the systems evolve, for several reasons. First, the reliance on beams is decreasing. MFTF-B uses beams for plasma heating, charge exchange pumping and density profile control. In the reactor configurations, only a sloshing ion beam for density profile control is needed. Moreover, the particle energies are increasing. This causes the same power to be delivered at lower current which simplifies the design. Development work has been done on molybdenum and vanadium-coated copper dumps up to $8,000 \text{ W/cm}^2$. However, at the levels envisioned, copper or AMZIRC dumps are preferred. The MFTF-B⁽⁵⁾ has short pulses of 0.5 to 30 s. The dumps are designed for 5×10^4 cycles. All later machines will have a much lower number of cycles. Other beamline components, notably collimators, will be subject to moderately high heat fluxes of $< 1,000 \text{ W/cm}^2$.

Fig. II-10. TFTR accelerator source grid assembly.

13-5720

Fig. II-11. EBT-S ICRH antenna.

Figure II-12 shows the layout of one version of MFTF-U⁽⁶⁾, a beam-driven mirror without charge exchange pumping. Each beam has both an internal (ion) and external (neutral) dump. Schematics of external and internal dumps are shown in Figs. II-13 and II-14 respectively. The cross section of an internally-finned, water-cooled copper dump is shown in Fig. II-15. In all designs, the option exists to angle the dump with respect to the incidence angle of the beam. This lowers the heat flux but raises the sputter rate.

II.2.2.2 Charge Exchange Areas

Charge exchange from low energy beams causes a distributed flux of energetic neutrals on the walls near the injection point. The angular distribution of these neutrals depends on beam energy and plasma density. At higher densities and lower beam energies, the neutrals are primarily backward peaked which causes more difficult design problems. This source of high heat fluxes does not occur in reactor-like configurations.

II.2.2.3 End Tank/Direct Converter

The end cell of a tandem mirror is used for direct conversion and particle removal. Present designs of advanced mirror systems such as MARS⁽⁷⁾ have four components used for heat removal in the end cell. These are the inner and outer collector plates and the inner halo (scraper) and outer halo (vented plate). These are depicted in Fig. II-16. The collector is segmented because different potentials are applied. The halo scraper is analogous to a tokamak pumped limiter. Particles scatter off the scraper back into the halo plasma. Once reionized they pass through the vented plate and are pumped by the vacuum system. Thus, the scraper is subject to high heat fluxes and erosion rates. Present designs use a thick beryllium coating to provide sputter lifetime. This is depicted in Fig. II-17 together with results of an analysis that shows how the 12-year sputter lifetime for MARS was determined. The heat flux on the outer halo is believed to be below that considered as a high heat flux component (100 W/cm^2); however, these estimates could be revised upward.

The distinction between FPD⁽⁸⁾ and MARS end cell designs is the desire for high quality heat in MARS which led to a coolant outlet temperature of 320°C and sufficiently high pressure ($\sim 3000 \text{ psi}$) to suppress boiling. The MARS structure is TZM, the only material that satisfies all constraints. FPD

Fig. II-12. MFTF-U: Beam and reactor layout.

Fig. II-13. External beam dump configuration.

Fig. II-14. Internal beam dump configuration.

Fig. II-15. Typical beam dump design.

MS-1012-028PP

Fig. II-16. End cell configuration.

MS-1012-003PP

Fig. II-17. Inner halo configuration and analysis.

operation would be at much lower temperatures and pressures. Copper alloys, steels, nickel alloys and vanadium would all be suitable.

The end cell components are subjected to a mixed environment of D, T, He and electrons. The most severe environments occur on the inner collector and halo scraper. The particle currents and energies for the inner collector are given in Table II-1 and those for the inner halo in Table II-2.

II.2.2.4 ICRH Antennas

Mirror reactors have ICRH heating in the end cell magnetic anchor to provide the high beta required for stability. Heat flux levels have not been defined but are expected to be comparable to those of a tokamak.

II.2.3 Alternative Fusion Concepts

The development of magnetic fusion energy is being pursued via two main-line programs, the tokamak and the tandem mirror. At a considerably lower level of effort, a number of less developed approaches are being pursued as alternative fusion concepts (AFCs). These approaches may eventually lead to a simpler, less expensive, or more desirable method of producing fusion energy.

The great variety of AFC types, and their associated problems, will be discussed only briefly here through use of some of the more prominent concepts. Reference 9 contains a detailed discussion of AFCs and HHFC requirements.

A realistic grouping of the most prominent AFCs follows:

- o ELMO or Nagoya Bumpy Torus (EBT/NBT)
- o Stellarator/Torsatron/Heliotron (S/T/H)
- o Reversed Field Pinches (RFP), OHTE, High Field Tokamak
- o Compact Toroids (CTs) [Field-Reversed Configurations (FRCs) and Spheromaks]

The last three concepts (and others, see Ref. 9) have both a "conventional" or low power density (LPD) design point and a "compact" or high power density (HPD) design point. These "compact" systems are characterized by a fusion power core (FPC) of reduced size/mass with system power densities approaching that of a light water fission reactor ($\sim 19.8 \text{ MWt/m}^3$).

Table II-1. Inner Collector Fluxes.

Particle	Current (A)	Energy (keV)
α	20.6	3862
α	50.6	942
D	9.6	800
D/T	73.8	353
e	1754	23.7

Table II-2. Halo Scraper Fluxes.

Particle	Current (A)	Energy (keV)
α	1.4	3500
α	3.4	600
α	208	50
D/T	1790	46
D/T	4.9	107
e	116	74

Many of the HHF problems are the same for the AFCs and the mainline approaches, especially for systems of equivalent system power densities. Therefore, the HHF requirements of the AFCs will be discussed relative to the requirements of similar mainline and/or HPD components. In general, the HPD concepts operate under more stressed conditions due to the higher levels of neutron, radiation, heat, and particle fluxes. The high heat loads require that a substantial fraction of the first wall serves as the limiter or be covered by limiter components. This dual role, limiter and first wall, may impose additional restrictions on the designs.

The problems associated with HHFC are near 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 < 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 presently in operation.

II.2.3.1 First Wall

Reactor concepts based on the FRC utilize plasmoids which translate through a burn region which has a resistive first wall and a flux conserving shell. The pulsed nature of this scheme results in a potential thermal fatigue problem for the first wall.

Although some spheromak reactor scenarios involve translating the oblate configuration, translating these plasmoids through a long-linear burn chamber will not maintain an equilibrium.⁽¹⁰⁾ These equilibrium constraints lead to stationary spheromak reactors surrounded by a loosely fitted conducting shell. The HHF characteristics of this first wall will be similar to those of the RFP.

For the RFP, a careful assessment of the critical features and parameters of the conducting shell is necessary. If the first wall must also be the conducting shell, additional complications may be imposed on the design. Wall armor and/or limiter components must be designed which are compatible with the plasma equilibrium and stability. The HPD systems will have to utilize large surface areas of graphite (coated or uncoated) or copper first walls due to the higher heat fluxes. If HHF coatings are used, a method for in situ replacement of the coating will have to be developed for future devices.

The wall loadings for the EBT and S/T/H are the same or less than for the tokamak. The three-dimensional helical character of the S/T/H presents added engineering difficulties for the first wall, and the continuous use of large amounts of ECRH may necessitate the development of first-wall components with high electrical conductivity for EBT.

II.2.3.2 Limiter/Divertor/Impurity Control(11,12)

Very little experimental or theoretical work has been done with limiters or divertors in the AFCs. As the duration of the experiments get longer, the issues of plasma materials interactions (PMI) and impurity control will become much more important. These issues may be responsible for the present difficulties in some present devices operating with high thermal wall loadings ($> 1 \text{ MW/m}^2$) such as HBTX-1A, 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 present 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 ($> 100 \text{ MW/m}^2$) for short pulses ($< 25 \text{ ms}$) without introducing impurities into the plasma system. Injection of gas into the boundary layer will be utilized 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).

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 peak stresses may be higher for HPD operation, the solutions should logically result from extensions of technologies learned at the lower stress levels. Both pumped-limiter and magnetic divertor particle control schemes are being considered; however, more experimental and theoretical work has to be done before these designs can be finalized.

The RFP and OHTE HPD options require extensions of technology by factors of ~5 from conditions in the LPD options; however, the Riggatron requires extensions of approximately another factor of 5.

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 those of the tokamak.

The limiter/impurity control issues and solutions for the EBT concept are very similar to those of the mainline programs.

Both the spheromak and the FRC have natural magnetic divertors. The PMI and HHF issues will be the same as the mainline program; HPD options will have correspondingly more heat flux, however. The spheromak approach⁽¹³⁾ 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.

II.2.3.3 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-O pumping will use low frequency (~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 and HHF systems.

II.3 Systems Integration Considerations

II.3.1 Safety

In principle, nearly all safety concerns relevant for the first wall and blanket are relevant for high heat flux components. The severity of any potential safety problem and the importance of the associated safety considerations are strong functions of material choice, component design, and location

in the reactor. Any analysis of safety concerns involving high heat flux components must necessarily consider the component in the larger reactor context, i.e., how the component influences the overall reactor performance.

For the purpose of this brief overview of possible safety concerns, safety considerations can be divided into two areas, (a) radioactivity and chemical toxicity control and (b) interaction with the surrounding reactor.

II.3.1.1 Radioactivity and Chemical Toxicity Control

Several considerations related to radioactivity and chemical toxicity are apparent: the induced radioactivity in the component, chemical toxicity of component materials, the vulnerability and potential for release of this activity/toxicity, permeation of tritium through component structure into the coolant, and the tritium inventory in the component.

Material activation (discussed in Sec. II.3.2) is actually several issues, e.g., occupational maintenance exposure, accidental release, afterheat generation and waste disposal/material recycle. Material preferences are a function of which of these issues are more important. For example, Nb is in disfavor due to its waste disposal problem of high long-lived activity; however, at short times its activity is lower so that accidental exposure per unit material is less dangerous than several other elements, e.g., irradiated Ni.

Nonetheless, some general guidance for structural material choice is possible. The most preferred are Si and C. The second general class is Mg, Al, and V. The third general class can be subdivided, with (a) Ti relatively more attractive, (b) Zr and Cr less attractive, (c) Cu, Fe, and W still less attractive, and (d) Mn. The final class, which should be avoided, is composed of Co, Ni, Mo, and Nb. Mo and Nb are the worst choices from the waste disposal standpoint. In addition, Mo oxidation products are quite volatile and form at a high rate. Ni and Co are the worst structural choices from the accidental exposure from a stoichiometric 316 SS/PCA release would come from Co isotopes induced in the Ni constituent in austenitic steel. The actual hazard from an alloy depends on its composition, including impurities. For "low activation" materials like SiC and Al alloys, the impurity concentration becomes quite significant.

Among possible coolants, Na and K should be avoided due to their activation. Activated corrosion products could be a concern if the metal/coolant interface temperature was high enough to cause significant corrosion activation product formation and release.

In terms of chemical toxicity, the key concern is Be. It is not yet possible to definitively compare Be toxicity with activation hazards. Chemical toxicity does not decay; however, possible accidental public hazards due to Be per unit amount released are less than many of the metals mentioned above.

One aspect of the vulnerability of the high heat flux component radioactivity is the component's response to inadequate cooling. By definition, the heat load on such components will be quite high. Therefore, the time allowed for corrective action following a loss of coolant or flow blockage/stoppage before damage can occur will tend to be short. Component design and material choice should consider how the component can be made more tolerant to such faults.

In the area of tritium permeation, the component structure's permeability is key. In this regard, W is most preferred. Then come, in order, Mo, Cu and steels. V and Nb are least desired. If a two-metal structure is used, it is far more important that the low permeability material be on the coolant side so that implantation on the plasma side cannot bypass the intended tritium barrier.

The relative importance of the tritium permeation problem is dependent on the difficulty of removing tritium from the high heat flux component. Water coolant is probably the worst choice since it is quite expensive to remove HTO from water. Use of lithium or ^{17}Li - ^{83}Pb as coolants would minimize the issue of the permeability of the structure--more tritium would likely be produced in the lithium than enters from the plasma. Tritium inventory in the component structure should also be minimized.

II.3.1.2 Interaction with Surrounding Reactor Components

HHF component design should strive to minimize the possibility of serious interaction. During off-normal operation or accidents, and to mitigate any consequences of interaction.

Plasma disruptions, or similar events, may cause their most serious damage to high heat flux components, e.g., a limiter. The component should be designed to withstand these transients, both to decrease direct damage to the component and to eliminate indirect damage to other reactor components from propagating failures.

Leakage of component coolant into the vacuum chamber could also lead to overpressure of the chamber or oxidation of its surfaces. For example, release of a substantial amount of water from a limiter into the chamber could rupture the vacuum barrier and damage vacuum pumps. Exposure of a reactive metal, like V, at high temperature to steam from a water-cooled limiter failure could lead to severe wall damage.

Similarly, a double failure of the high heat flux component and blanket would allow component and blanket fluids to interact. A single initiator event, e.g., disruptions or seismic events, could, in some cases, cause such a double failure. The consequences of such reactions would be severe enough in the case of a water-cooled HHFC and a liquid lithium blanket that such combinations may be ruled out for use in power reactors, particularly if V alloy is used as the FW/S structure. The reaction between ^{17}Li - ^{83}Pb and water is less energetic, and the use of water-cooled HHFC components with such blankets may be acceptably safe. Water coolant is acceptable for use in HHFC's with all solid breeder blankets.

II.3.2 Activation Products

Accurate evaluation of neutron-induced component activation is crucial in establishing shielding requirements, design concepts, maintenance procedures, and personnel access to experimental areas. The evaluation is also vital in reactors.

There are three areas that must be investigated in terms of the impact of component activation: (1) radioactive waste management; (2) reactor maintenance; and (3) reactor safety characteristics. The primary incentives for consideration of reactor safety in the case of a severe accident, and of long term waste management, relate to environmental impacts. Incentives influencing maintenance considerations, in particular "hands-on" or "contact" maintenance considerations relate primarily to economics.

Table II-3 summarizes some of the activation analyses published to date, using two radioactivity measures, Ci/cc and rem/h. All the analyses have been done for values of integral neutron wall load, I_w , typical of power reactors. Materials for high heat flux components are grouped into three categories: (1) coating/surface materials; (2) heat sink materials; and (3) structural materials mostly for the first-wall application. The analyses have been carried out without including typical impurities/trace elements except for a few structural materials (e.g., Al-6063, Ti6Al4V, V15Cr5Ti, and PCA) and SiC, where these are more important.

Several important observations can be made from the results shown in the table. First, the high-purity coating/surface materials shown do not seem to pose any long-term radwaste problems. Particularly, the low-Z coating/surface materials exhibit a very rapid decay of activation shortly after shutdown. A careful maintenance scenario must be established for use of SiC as a high heat flux material because of the formation of ^{31}Si (2.6-h, 1.27-MeV gamma emission) by the $^{30}\text{Si}(n, \gamma)$ reaction. All of the long-term activation associated with the constituent elements are soft-energy beta emitters such as ^{14}C and ^{10}Be . For near-term machines the plateau values of the long-term activation will be substantially lower than those shown, for near-term machines because the production rate is approximately proportional to the integral wall load. Second, copper used as a heat sink material will be highly activated and may require careful consideration for the post-irradiation handling. Shortly after shutdown, the copper activation is dominated by the 1.35-MeV gamma emission of ^{64}Cu which appears to pose a difficult maintenance problem. Even in a near-term machine, an appreciable amount of ^{64}Cu production is expected because of the short half life (12.7 years) of ^{64}Cu . High purity vanadium alloys seem to be more acceptable for use as heat sink materials from the activation standpoint, but strict control of the impurities/trace elements is crucial.

Finally, all of the structural materials shown exhibit quite high radioactivity in Ci/cc as well as in rem/h. In addition to important waste disposal considerations, the activation of these structural materials will become very important for reactor accessibility or contact maintenance considerations, as indicated by their very high biological dose rates shortly after shutdown. Unfortunately, it is expected that quite high dose rates are likely to occur

Table II-3. Comparison of Activation for Various HFEC Materials

Radioactivity (Ci/cc)				Dose (Rem/hr)			Major Contributor to Long-Term Activation	
Material	Ref.	0	1 Yr	100 Yr	0	1 d		100 Yr
(A) COATING/SURFACE MATERIALS								
C	a	6(-10)	6(-10)	6(-10)				$^{14}\text{C}(5730\text{y})$
††* SiC	b	1(2)	7(-5)	7(-5)				$^{14}\text{C}(5730\text{y})$, $^{55}\text{Fe}(2.7\text{y})$
Be	c	2(1)	6(-6)	6(-6)				$^{10}\text{Be}(1.6\text{My})$
Ta-5W	d	4(2)	5(1)	0				$^{182}\text{Ta}(115\text{d})$
(B) HEAT SINK MATERIALS								
CU	a	4(2)	4(1)	5(-1)				$^{60}\text{Co}(5.3\text{y})$, $^{63}\text{Ni}(100\text{y})$
V-20Ti	d	7(1)	3(0)	0				$^{49}\text{V}(330\text{d})$, $^{45}\text{Ca}(163\text{d})$
(C) STRUCTURAL MATERIALS								
††* Al6063	e	1(2)	6(-2)	5(-4)	1(8)	3(7)	7(1)	$^{55}\text{Fe}(2.7\text{y})$, $^{26}\text{Al}(0.7\text{My})$
* Ti6Al4V	e	9(1)	7(-1)	1(-4)	5(7)	3(7)	9(-1)	$^{63}\text{Ni}(100\text{y})$, $^{26}\text{Al}(0.7\text{My})$
* V15Cr5Ti	e	1(2)	4(-1)	8(-5)	2(7)	9(6)	3(-1)	$^{93}\text{Mo}(3.5\text{Ky})$, $^{94}\text{Nb}(20\text{Ky})$
* PCA	e	4(2)	3(1)	1(-4)	4(7)	1(7)	8(0)	$^{55}\text{Fe}(2.7\text{y})$, $^{94}\text{Nb}(20\text{Ky})$
HT-9	f	5(1)	2(1)	7(-4)				$^{63}\text{Ni}(100\text{y})$, $^{93}\text{Nb}(13.6\text{y})$
Nb	a	2(2)	3(-5)	3(-5)				$^{92}\text{Nb}(20\text{My})$, $^{93}\text{Zr}(1.5\text{My})$
Mo	a	1(2)	9(-2)	1(-1)				$^{93}\text{Mo}(3.5\text{Ky})$, $^{93}\text{Nb}(13.6\text{y})$
Zr	a	1(2)	9(-1)	1(-4)				$^{93}\text{Zr}(1.5\text{My})$, $^{90}\text{Sr}(28.1\text{y})$
Ni	a	4(2)	7(1)	4(-2)				$^{60}\text{Co}(5.3\text{y})$, $^{63}\text{Ni}(100\text{y})$

Ref. (a) "Panel Report on Low Activation Materials for Fusion Application," Draft, Appendix II (1982); $I_W = 3 \text{ MW}/\text{m}^2$.

Ref. (b) "Background Information and Technical Basis for Assessment of Environmental Implication of Magnetic Fusion Energy," Draft, U.S. DOE (1983); $I_W = 18 \text{ MW-yr}/\text{m}^2$.

Ref. (c) J. Jung, Nucl. Technology/Fusion, 4 144 (1983); $I_W = 9 \text{ MW-yr}/\text{m}^2$ (Cat-D).

Ref. (d) C. Baker, ANL/FPP-80-2 (1980); $I_W = 18 \text{ MW-yr}/\text{m}^2$.

Ref. (e) J. J. Jung, Nucl. Technology/Fusion, 4 566 (1983); $I_W = 18 \text{ MW-yr}/\text{m}^2$.

Ref. (f) J. Jung, unpublished work for STARFIRE (1979); $I_W = 5 \text{ MW-year}/\text{m}^2$.

†: Read as 6×10^{-10} ; ††: (*) denotes impurities included.

even in near-term machines due to a variety of short-lived gamma emitters formed in the constituent elements Al, Ti, V and Fe.

The long-term activation in V15Cr5Ti is generated only by its impurities, most importantly by niobium and molybdenum. These two elements predominantly contribute to the high radioactivity in Ci/cc as well as to the high dose rate in rem/h via the generation of ^{93}Nb , ^{93}Mo , and ^{94}Nb . It is therefore very desirable from the standpoint of radwaste management to use vanadium-base alloys with extremely low levels of Nb and Mo impurities.

II.3.3 Economics

II.3.3.1 Reliability

Highly reliable HHF components are needed in near-term fusion devices in order to maximize the operating time available for experiments. In power reactors high levels of reliability in HHFC are crucial, to keep unscheduled maintenance for changeouts of failed components to a minimum so as to maximize plant availability.

An example of the reliability levels considered necessary for typical HHF components is shown in Table II-4. For first wall coolant panels and pumped limiters in tokamaks, the estimated reliability--as expressed in mean time between failures, MTBF--is given for the two systems (each composed of n units as indicated). These are reliability levels considered necessary to be achieved in each device in order to reach the designated overall availability goal for that device. (Similar numbers for TMRs and alternate concepts are not available.)

It should be noted that, in order to assure with a high probability that the desired MTBF for any HHF component will actually be achieved when operating in the device, the total test time that has to be accumulated beforehand on near-identical components is higher than the desired MTBF by several multiples.⁽¹⁾ For example, if testing in ETR is used to assure with 95% confidence that the MTBF of the same limiter in DEMO will be 8000 hours, then a total of at least 28,000 hours of limiter operation in ETR (all limiter units combined) would be necessary. Such relationships of accumulated testing time to desired reliability levels are very important when planning development programs for HHF or any other critical component for fusion devices.

Table II-4. Estimated Requirements for Reliability
of HHF Components for Tokamaks

Component (No. of Units)	Machine	MTBF (Mean Time Between Failures)	MTTR (Mean Time to Replace)
First Wall Panels (24)	ETR ^a	2,200 hrs	420 hr
	DEMO ^a	4,400	210
	STARFIRE ^b	~ 9,000	~ 288
Pumped Limiter (12)	FTR ^a	4,000	336
	DEMO ^a	8,000	168
	STARFIRE ^b	~72,000	~ 288

^a Ref. 1

^b Ref. 3

II.3.3.2 Use of Power from HHF Components

The efficiency with which the power removed from HHFC can be used in the power generation system of a fusion reactor can strongly affect the cost of electricity (COE) generated by the plant. It is important to design the HHFC, if possible, to permit the use of coolants with high outlet temperatures so that power conversion efficiency can be maximized. From the standpoint of power production, the choice of HHFC coolant and its operating parameters will be influenced by:

- o Power conversion system cost and complexity
- o Potential for tritium contamination of/loss into steam generator secondary side (or turbine, etc.)
- o Thermal efficiency obtainable with various system configurations

In recent tokamak reactor studies,^(2,3) water coolant at relatively low pressures and temperatures has been selected for both the FW/B and the impurity control device. The MARS study⁽⁷⁾ used high-pressure, high-temperature water coolant from the TMR's halo scraper and direct convertor for feedwater

heating, in conjunction with the ^{17}Li - ^{83}Pb liquid metal breeder/coolant used for the FW/B.

The design lifetime or reliability of the HHFC is likely to be reduced as coolant outlet temperatures are increased. For example, surface erosion lifetimes for pumped limiters are a function of allowable erosion thickness, which may decrease with increasing coolant outlet temperature if thickness is limited by the difference between maximum allowable surface temperature and coolant/structure interface temperature. However, reduced life and reliability must be traded off against economic penalties. An extreme case would be to reject all power absorbed by the HHFC as waste heat. As examples, for Starfire,⁽³⁾ eliminating the 200 MWth from the limiter (145°C coolant outlet temperature) which was used for feedwater heating would decrease net power output by ~ 5%. For MARS,⁽¹⁴⁾ rejecting the ~ 280 MWth from the halo scrapers and direct converters (320°C coolant outlet temperature) would decrease net power output by ~ 8%. COE would rise approximately in inverse proportion to the power reduction in both cases.

II.3.3.3 Design Lifetime

The design lifetime of an HHFC component, in terms of impact on reactor availability and economics, can be defined as the time elapsed between its installation and its removal as part of scheduled maintenance to prevent it from exceeding allowable limits based on some parameter (e.g., minimum ductility limit based on total fluence accumulated).

The impact of shorter design lifetimes on plant economics for power reactors comes principally through (1) increased costs for spares, and (2) decreased availability in the event that total time required for component changeout exceeds the downtime allotment for other scheduled maintenance. For near-term experimental devices, the impact of reduced availability is on the testing schedule rather than on economics.

As with the decisions on the utilization of power from HHFC, the determination of design lifetime for these components must also involve tradeoffs to optimize reactor economics. As an example of the importance of HHFC design lifetime, the Starfire⁽³⁾ pumped limiter design lifetime was 6 years (based on surface erosion) and the limiter was assumed to be changed out with the FW/B sectors on a rotating schedule during the annual shutdown for maintenance.

Thus limiter life had no particular effect on availability. However, if erosion rates were much higher so that all limiters had to be replaced every six months, an additional scheduled shutdown of at least 30 days would be needed for this operation. Assuming no other changes to the maintenance scenario, plant availability would decrease from 75% to less than 67%, and COE would increase by ~ 12%.

For those compact HPD systems that can operate efficiently at high neutron first wall loading ($10\text{--}20\text{ MW/m}^2$) with exo-blanket resistive coils, fusion power core (FPC) power densities in the range of $10\text{--}15\text{ MWt/m}^3$ are possible. For a radiation life fluence of 15 MW-yr/m^2 the 20 MW/m^2 CRFPR system would operate for 0.75 full-power years (FPY), or approximately one chronological year (75% plant factor) before changeout of the 45-tonne first wall and blanket system. If the 20 MW/m^2 HPD design can be technically achieved, a few single-piece "batch" maintenance schemes may be possible, wherein a completely assembled and pre-tested first-wall/blanket/shield/toroidal-field-coil unit would be installed in the reactor hall after off-site fabrication and quality assurance. The potential for a reduced mean-time-to-repair can be traded off with the possibly decreased mean-time-to-failure. High plant availability and reduced costs may thus be possible, which makes the higher wall loading designs attractive, since cost of electricity is a weakly diminishing function of wall loading above $\sim 10\text{ MW/m}^2$.

II.3.3.4 Maintainability

The maintenance of HHFC in present experimental devices is characterized by fully manual operations for removal and replacement, or in-situ repair when feasible. Full access to workers is possible because tritium is absent, material activation levels are very low, and the building atmosphere is air. The maintenance of HHFC for all future devices which burn D-T fuel will have to be by fully remote methods. In-situ repairs will probably be restricted to re-coating operations only. If activation levels permit workers to enter the reactor hall at all, the workers will wear "bubble suits" because of tritium contamination of the building atmosphere, which would probably be a gas rather than air except for near-term devices.

As a result, for future devices there will be strong emphasis on making HHFC easily maintainable, with changeouts accomplished remotely using simple

operations and total downtime minimized. Coolant line connections and vacuum boundary seals in particular will have to be easily broken and remade. Any refurbishment of the removed components (e.g., replacement of armor tiles) will have to be accomplished in hot cells after a new replacement component has been installed and device operation has resumed; most in-situ refurbishment (with the device not operating) would simply take much too long to be either economically feasible (for power reactors) or reasonable in terms of test schedule impact (for near-term devices), as indicated by Fig. II-18.

As the requirements and problems of HHFC have become better defined, reactor design studies have placed increasing emphasis on improving the maintainability of those components. An example is provided by pumped limiters for tokamaks. The limiter for Starfire⁽³⁾ was integrated structurally with the FW/B sector, and was removed and replaced together with the sector (Fig. II-19a). For the later DEMO study,⁽²⁾ the limiter design was modified so that the entire unit could be withdrawn from the blanket sector as a separate module (Fig. II-19b), without requiring sector removal. Estimated removal/replacement time was cut from ~ 12 days to ~ 6 days (unscheduled maintenance for a failed component).

II.3.3.5 Tritium Breeding Effects

The incorporation of relatively large HHF components, such as divertors or limiters in tokamaks, into the reactor requires that the associated penetrations be accommodated by the tritium breeding blanket system. Such HHFC will require openings in the first wall having direct visibility to neutrons from the plasma.

The tritium breeding ratio (TBR) will be affected by (1) the changes due to the penetrations themselves⁽¹⁵⁾ (influenced by neutron spectrum variation, reduction in breeding zone volume, enhanced neutron loss), and by (2) the absorption or multiplication of neutrons by the materials in the HHFC. Radiation streaming through penetrations enables high energy neutrons to penetrate deeper into the blanket, increasing the Li-7 reaction rate. Volume reductions in the breeding zone depend strongly on the type of penetration. The relative TBR reduction depends on the exact characteristics of the penetration. For typical neutral beam injectors the reduction can be ~ 2%.⁽¹⁶⁾ The use of a

Fig. II-18. Impact of availability in experimental fusion devices.

(a) STARFIRE — Limiter Removed With Blanket Sector

(b) DEMO — Limiter Module Separately Removable

Fig. II-19. Comparison of limiter maintenance approaches for Starfire and Demo.

pumped limiter can reduce TBR by ~ 5%,⁽¹⁷⁾ and a divertor system can result in even greater reductions.

Recent analyses⁽¹⁸⁾ to determine the effects of limiter location and coolant choice on TBRs for various blanket concepts indicate the effects of both can be important, particularly for blankets concepts marginal in breeding. Table II-5 summarizes those preliminary results. The effect of H₂O in depressing TBR for a He-cooled Li₂O blanket is evident, and would have to be accounted for in selection of a limiter coolant. Likewise, moving the limiter from the midplane to a bottom location appears to improve TBR in all three cases, but by differing degrees.

Table II-5. Effect of Limiter Location and Coolant Choice on Tritium Breeding Ratio^b

<u>Blanket Concept</u>	<u>1-D TBR</u>	<u>Limiter Position^a</u>	
		<u>Midplane</u>	<u>Bottom</u>
(A) Li ₂ O Breeder H ₂ O Coolant	1.11	1.05 (H ₂ O)	~1.06 (H ₂ O)
(B) Li ₂ O Breeder He Coolant	1.19	1.14 (He) 1.09 (H ₂ O)	1.14 (He) 1.11 (H ₂ O)
(C) Li Breeder Li Coolant	1.24	1.19 (Li) 1.13 (H ₂ O)	---- 1.16 (H ₂ O)

^a Limiter coolant indicated in parentheses.

^b Starfire limiter assumed; 1 cm Be coating on outside of structure.

II.3.4 Effects on Other Reactor Subsystems

Magnetics - The choice of heat removal material for a toroidally continuous first wall or limiter impacts the magnetic design of tokamak. If the materials are magnetic, they alter the flux pattern. The application of such materials, therefore, must be carefully controlled. A first wall or limiter

constructed of a highly conductive material, such as copper, provides a good toroidal eddy current path near the plasma. This feature assists in providing plasma stability and prevents arc erosion during transient electromagnetic events, such as disruptions. On the other hand, the startup and control coil powers and voltages are adversely affected by this highly conducting toroidal path. The FED-A design study⁽¹⁹⁾, which incorporates a highly conducting copper limiter and first wall, demonstrated that a self-consistent and practical design with respect to plasma stability, startup, and control is feasible.

Plasma Interaction - All components exposed to the plasma are subject to erosion by sputtering, evaporation, etc. and constitute a potential source of impurities to the plasma. Furthermore, recycling and permeation of plasma particles, including tritium permeation through walls and into cooling channels, depends on the chosen materials. Consequently, the materials choice for plasma side components has to be compatible with plasma operation. The effects of plasma material interaction are discussed in detail in the Technical Assessment report of the PMI task group.

II.4 Key Issues and R&D Needs

The four prioritized issues and related R&D needs discussed below were developed based on the systems integration aspects of HHFC as discussed in the preceding sections. Two points should be mentioned. First, these issues focus principally on the needs of the two mainline reactor concepts, tokamak and tandem mirror. Successful resolution of these issues will also resolve similar issues for many of the low-power-density alternative concepts. The HHFC-related issues for most of the high-power-density alternative concepts do not appear to be well enough defined at this time to warrant large-scale R&D expenditures. Second, the issues relate principally to the needs of power reactors rather than near-term or present experimental devices, because the most pressing problems for such devices appear to be related to materials and thermal-hydraulics issues rather than to systems integration.

II.4.1 Lifetime of Impurity Control Devices

For limiters and divertor target plates in tokamaks, and for halo scrapers in TMR's, useful life appears to be limited principally by high erosion rates of the low-Z armor. Lifetimes of less than 1-2 years will likely result

in extensive downtimes for scheduled removal and replacement, resulting in reduced availability which raises the cost of electricity.

Operating at cold plasma edge conditions would permit use of high-Z materials with low sputtering rates which would reduce the problem's severity.

R&D needs:

- Investigation of conditions necessary for operation at cold plasma edge conditions.
- Development of armor-to-structure attachment methods for maximum thermal conductivity to maximize allowable armor thickness.
- Development of rapid I-C device changeout techniques and maintenance equipment.

II.4.2 Tritium Contamination of Coolant in HHFC's

Permeation rates of tritium from the plasma through armor (if present) and structural materials into coolant are largely unknown for the fusion environment. For water coolant, high levels of contamination will result in high costs for tritium removal from the water and the unacceptability of the heated water for use in power conversion systems, a significant economic penalty in the case of impurity control components.

R&D needs:

- Basic research in tritium barrier effectiveness: oxide films, coatings

II.4.3 Safety of Liquid-Metal-Cooled FW/B with H₂O-Cooled HHFC's

The use of liquid-metal-cooled FW/B concepts in conjunction with H₂O-cooled HHFC's such as limiters or choke coils is presently perceived to entail significant risk of the occurrence of large-scale energy releases for some projected accident scenarios. Very little work--analytical or experimental--has been done to better quantify this risk.

If H₂O-cooled HHFC's are judged unacceptable for use with liquid metal FW/B concepts, the cooling problems for such HHFC's will become much more difficult since other coolants are significantly poorer from the thermal-hydraulics standpoint.

R&D needs:

- Testing to determine the probabilities of the individual failures and events necessary to bring about large-scale energy releases in realistic configurations for LM/H₂O contact, using candidate liquid metals.
- Probabilistic FMEA, using results from the above tests, to provide quantitative data sufficiently accurate to permit judgements of the acceptability of the risk.

II.4.4 Effect of Impurity Control Devices on TBR for Tokamaks

Impurity control options for tokamaks such as bottom pumped limiters and single- or double-null divertors take up significant breeding blanket volume. Unless realistic designs can be developed that incorporate breeding zones into these impurity control devices, their use may ultimately preclude the use of blanket concepts with lower potential TBR's (e.g., Li₂O/H₂O).

R&D needs:

- Engineering development of leading candidate impurity control options which include breeding zones with TBR values $\approx$ unity.

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CHAPTER II

SYSTEMS INTEGRATION

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III. MATERIALS AND PROCESSES

III.1 Operating Conditions

The establishment of the plasma edge and scrape-off parameters represents the starting point in determining the operating environment for impurity control components. For tokamaks, the critical parameters in the scrape-off layer are plasma density n_e , electron- and ion-temperature T_e and T_i and characteristic decay lengths λ_T and λ_q of particle and power flux respectively. For reactor relevant plasma operation the plasma density in the scrape-off layer can vary between 10^{12} and 10^{13} cm^{-3} with electron- and ion-temperatures on the order of 100 eV, ion fluxes parallel to the magnetic field of $\Gamma_i = 3 \times 10^{19} \text{ cm}^{-2} \text{ s}^{-1}$ are possible at the plasma boundary. Along with the corresponding electron flow, this can result in heat fluxes of several kW/cm^2 perpendicular to the magnetic field. Particle and heat fluxes of this order are measured in present devices. In order to accommodate these heat fluxes, limiters and divertor collector plates are shaped to spread out the heat fluxes over a large area. Several actively cooled systems for future machines are designed for heat fluxes of 3-5 MW/m^2 . The corresponding particle fluxes are of the order of $\sim 2 \times 10^{18}/\text{cm}^2\text{-s}$. In addition to particle and power fluxes, the corresponding decay lengths in the scrape-off layer have to be known for the design of limiters, divertor plates and RF-antennae. Since the characteristic decay lengths λ_q and λ_T depend on the limiter configuration itself as well as on perpendicular and parallel transport, they cannot be predicted accurately and have to be parameterized. A reasonable range for λ_q is 0.5 to 5 cm, where 1-3 cm seems to be most likely. The corresponding length for the particle flux, which is important for the design of pump limiters, is nominally $(5/3) \cdot \lambda_q$ at $T_e = T_i$.

The operating environment experienced by impurity control components in tokamaks will vary with the time frame of the device being considered, as shown in Table III-1. In general, burn lengths, availability, neutron wall loading, and neutron fluence will increase in going from near term to long term devices. The surface heat flux to these components is not expected to vary greatly from the heat loads seen in present devices, but the requirements for heat removal will change as the burn length increases. The total number

Table III-1. Impurity Control Parameters - Tokamaks

Parameter	Present Machines (0 - 3 y)	Near Term (3 - 8 y)	ETR's (1990s)	Long Term (> 2000)
Heat Flux	$4 \text{ MW/m}^2 - 1.5 \text{ s (TFTR)}$ $70 \text{ MW/m}^2 - .25 \text{ s (TFTR)}$	$\leq 5 \text{ MW/m}^2$	$\leq 5 \text{ MW/m}^2$	$\leq 10 \text{ MW/m}^2$
Neutron Flux	0	$1 - 2 \text{ MW/m}^2$	$1 - 2 \text{ MW/m}^2$	$2 - 5 \text{ MW/m}^2$
Neutron Fluence	0	$< 1 \text{ dpa}$	$\sim 20 \text{ dpa}$	$\sim 50 \text{ dpa}$
Burn Time	0.2 - 5 s	20 - 300 s TFCX	200 s INTOR	Cont.
Structural Materials	Nickel Base Alloys Stainless Steel	Copper Alloys	Copper Alloys Refractory Metals	Copper Alloys Refractory Metals
Coolant	None	Water	Water	Water Liquid Metals
Edge Temperature	10 - 100 eV	$\sim 100 \text{ eV} - \text{TFCX}$	$\sim 20 - 100 \text{ eV INTOR}$	20 - 100 eV - Tokamaks
Configuration	Tiles Mechanically Attached to Heat Sink	Coating/Cladding Bonded to Heat Sink	Coating/Cladding Bonded to Heat Sink	Coating/Cladding Bonded to Heat Sink
Operating Temperature				
Surface	$800^\circ\text{C} - 1800^\circ\text{C}$	$< 1000^\circ\text{C}$	$< 1000^\circ\text{C}$	$< 1000^\circ\text{C}$
Heat Sink	$< 200^\circ\text{C}$	$\leq 400^\circ\text{C}$	$\leq 400^\circ\text{C}$	$\leq 750^\circ\text{C}$
Number of Cycles	10^5	$\sim 10^5 \text{ TFCX}$	$\sim 10^6 \text{ (INTOR)}$	$< 10^3$
Surface Materials	Graphite TiC, SiC Coated Graphite Be	Be SiC BeO C	Be W Ta	Be W Ta V

of burn cycles will decrease for long term devices as the burn lengths move towards steady state operation.

The operating parameters of impurity control components in mirrors will be similar to those in tokamaks in terms of operating temperatures and heat fluxes. Therefore, the candidate materials considered for mirrors are the same as for tokamaks. There are some differences expected in the operating conditions, however, which could influence the final choice of materials. Mirrors are expected to operate in a continuous mode, and the total number of operating cycles will be greatly reduced compared with near term tokamaks. The plasma edge (or halo) temperature may also be different in mirrors. Present analyses indicate that the halo temperature will be ~ 10 eV compared with ~ 100 eV for the plasma edge in tokamaks. Finally, the total amount of radiation damage experienced by impurity control materials in mirrors may be reduced compared with tokamaks, since these components are located in the mirror end cells where the neutron flux is much lower than in the central cell.

Auxiliary heating components including neutral beam dumps, RF antennas, and waveguide domes, are likely to experience high heat fluxes. Beam dumps for present systems are designed to accommodate very high heat fluxes, and they are all composed of water cooled, copper alloy heat sinks. Future beam dump conditions and designs are likely to be similar to those in present systems. RF antennas and waveguide domes will be directly exposed to the plasma and neutron flux, and therefore they will experience roughly the same environment as the limiter blades and divertor collector plates. In addition, however, there are additional electromagnetic requirements placed on the materials used for these components. Copper alloys will be used for both their high thermal and electrical conductivity, while ceramics will be used primarily for their electrical insulating properties. BeO , for example, is used as the waveguide dome material in most designs because of its high transparency to the RF waves.

III.1.1 Requirements and Candidate Materials

There are a number of requirements for the high heat flux materials. The ability to withstand high heat fluxes depends to a great degree on the thermo-physical properties of the material. In particular, the materials should have a high thermal conductivity, high specific heat, a low coefficient of thermal

expansion, and a low elastic modulus to minimize thermal gradients and stresses. The materials should exhibit mechanical properties which guarantee structural integrity for the desired operating lifetime of the component. The mechanical properties of importance are tensile, creep, fatigue, and crack growth. The materials used in devices that produce high neutron fluences should be resistant to radiation damage. Both the thermophysical properties and mechanical properties should not be significantly degraded by radiation. In addition, the materials should be reasonably resistant to dimensional changes caused by radiation induced swelling and creep. The materials should also be capable of being fabricated into the complicated geometry of high heat flux components. Besides the usual fabrication considerations, it is important that the materials are capable of being successfully joined together. The property requirements for the joints are the same as the bulk materials. Finally, the materials should not be a significant source of plasma contamination. The properties of importance relate to the interactions of the plasma with the material's surface, which are described in detail in the PMI task group technical assessment. The selection of the material which is exposed to the plasma will depend on the plasma edge conditions as described below.

In most cases, no single material can be identified that satisfies all of the requirements for high heat flux components. For example, the special requirements needed for plasma-materials interactions lead to choices that are not generally considered to be structural materials. Therefore, at least in the area of impurity control, the materials are divided into the classes of plasma side materials and heat sink materials. In the case of beam dumps and other components that do not interact directly with the plasma edge, only the heat sink materials usually need to be considered. When both a plasma side material and a heat sink material are required, they are usually designed to be metallurgically bonded together. Thus, the behavior of bonds is also an important consideration for high heat flux components.

The choice of a plasma side material is strongly influenced by the plasma edge conditions. The major concerns are the potential for plasma contamination as a result of physical sputtering from the surface and the possibility of rapid erosion leading to short lifetimes for these components. The sputtering rate is dependent on the incident particle energy, mass, and flux. It is expected that sputtered material will eventually return to the surface to

cause additional sputtering. The most important parameter influencing material selection is the plasma edge temperature. At low plasma edge temperatures (≤ 50 eV), the high-Z materials such as tungsten or tantalum are preferred because the net erosion rates are predicted to be essentially zero. At higher edge temperatures, it is predicted that sputtered particles from high-Z surfaces will return to the surface at energies sufficient to yield self sputtering coefficients greater than unity resulting in an unacceptable sputtering cascade. The only materials that could be used under these conditions are those whose self-sputtering coefficients never exceed unity. The elements which meet this criteria are those with atomic numbers below that of silicon. The low-Z materials that have been examined most closely are Be, C, BeO, and SiC. These materials meet both the sputtering requirement and the requirement to be able to accommodate high heat fluxes.

The primary requirements for the heat sink materials are the ability to accommodate high heat loads and the ability to maintain structural integrity over the lifetime of the component. The first requirement leads to the selection of high thermal conductivity alloys as candidate materials. High conductivity alloys that are presently available in large quantities are aluminum and copper alloys. Copper alloys have received greater attention primarily because of their ability to operate at higher temperatures compared with aluminum alloys. Other alloys that also offer potential are transition metals such as molybdenum, vanadium, niobium, and tantalum. These metals are capable of operating at very high temperatures and are compatible with liquid metal coolants. The second requirement is more difficult to assess because of the lack of data for many properties that are relevant to long term operation.

III.2 Plasma Side Materials

The properties relevant for plasma side materials include the surface, physical, mechanical, and irradiation properties. The surface properties that are important in plasma-materials interactions are covered in the technical assessment report of the PMI Task Group, and only a brief summary is presented here. The other properties are discussed in greater depth.

III.2.1 Surface Properties - Interface With PMI

The choice of plasma side materials is strongly restricted by plasma operation. Processes like sputtering, arcing, and evaporation contaminate the plasma with impurities the radiation of which constitutes an energy loss channel. To keep the radiation losses on a tolerable level, either the atomic number Z of the material has to be low or the impurity production process has to be minimized. At present, it is not known how to cool the plasma edge sufficiently to prevent sputtering; hence, low- Z materials, such as graphite are selected. If the plasma edge can be cooled down below the sputtering threshold, which goes up with increasing Z of the material, it might be possible to utilize higher- Z materials like molybdenum, or tungsten.

The processes that contaminate the plasma with wall impurities represent at the same time erosion mechanisms for the first wall components. Present machines operate in a short pulse, low duty cycle mode and the impurity aspect is the dominant concern. In contrast, surface erosion is likely to be the dominant issue in future devices which are being designed now for long pulse and high duty cycle operation. Extrapolating plasma parameters of present machines, erosion rates of several cm per year have to be accommodated. Consequently, unless the plasma edge temperatures can be reduced below the sputtering threshold, thin wall designs for removal of high heat fluxes cannot be utilized.

The major materials processes of interest are physical sputtering, chemical sputtering, and hydrogen (helium) implantation and release. The integration of these effects with the fusion environment is predicted to result in continuous erosion and redeposition of the plasma side surfaces during normal operation. Although there is a significant data base for sputtering and hydrogen implantation and release, there are essentially no materials data on the combined erosion and redeposition expected in fusion devices.

III.2.2 Assessment of Plasma Side Materials

The candidate plasma side materials include the low- Z candidates Be, C, SiC and BeO, and two high- Z candidates W and Ta. A detailed data base for these materials is presented in Ref. 1. The physical and mechanical properties of these candidate low- Z materials, and presumably also other candidates such as B, B₄C, TiC, etc., are strongly dependent on the fabrication method

and on the product form. This results in a range of property values encountered in the literature, and requires a careful selection of properties to be used in design with these materials. The properties of all of these materials can be affected by irradiation, although data are sparse for irradiation conditions that adequately simulate fusion reactor service. Beryllium metal is unique in the low-Z candidates in that its thermal conductivity is not affected by irradiation; for the other low-Z candidates the thermal conductivity is rapidly degraded by neutron irradiation.

The physical properties of the high-Z candidate materials are well understood, and are relatively insensitive to product form and fabrication technique. Relatively few data are available on irradiation effects on W and Ta but little change is expected in their physical properties during irradiation, and loss of ductility can be expected if service temperatures are relatively low.

The properties and prospects of the plasma side materials were summarized by the FED/INTOR project. The following material is extracted from that summary. Representative property values for low-Z and high-Z candidate plasma side materials are presented in Tables III-2 and III-3, respectively (from Ref. 1).

Beryllium has a very low atomic number, good thermal conductivity and heat capacity and relatively high heats of vaporization and fusion. Graphite has many attractive properties including a much higher melting temperature; it normally tends to vaporize before melting. The vaporization rate becomes excessive at temperatures above ~ 2000 K. Although graphite has a relatively high thermal conductivity at low temperatures, the conductivity is rapidly reduced at relatively low radiation levels (< 1 dpa). Silicon carbide also tends to decompose and vaporize before melting, and therefore, avoids the melt layer problem. However, the maximum operating temperature (~ 1700 K) and the thermal shock resistance of SiC are generally lower than those for graphite. As in the case of graphite, the low-temperature thermal conductivity of SiC decreases rapidly with radiation fluence. The BeO has a high melting temperature, high thermal conductivity at low temperatures, and is not believed to be susceptible to chemical sputtering. Although data for the liquid phase are sparse, calculations indicate that a thin melt layer will form on BeO during a disruption. This compound is also susceptible to rapid deterioration of the low temperature thermal conductivity at low neutron fluences.

Table III-2. Property Values at 800 K for Low-Z Materials

Property (Units)	Property Value			
	Be	BeO	C	SiC
Melting Temperature (K)	1557	2823	-	-
Sublimation Temperature @ 10^{-2} Pa(K)			2300	1800
Density ^a (Mg/m ³)	1.85	3.01	1.8 ^b	3.20
Thermal Expansion, 300-800K (10^{-6} K ⁻¹)	15.9	8.2	4.2	4.9
Thermal Conductivity (W/mK)				
Unirradiated	103	66	103	37
Irradiated ^c	103	36	25	9
Specific Heat (J/kgK)	2250	1840	1620	1150
Young Modulus (GPa)				
Unirradiated	190	36	8	400
Irradiated	-	-	18	400
Yield Strength (MPa)				
Unirradiated	200	76	14 ^d	360
Irradiated ^c	-	-	21 ^d	360
Elongation, Irradiated ^c (%)	5	<1	<1	<1

^a Density at 300 K.^b Density of Graphite Ranges up to 2.25 Mg/m³ Depending on Product Form.^c High Fluence Irradiation Properties.^d Tensile Strength for Graph-N3M is about 470 MPa (Unirradiated), Estimated 230 MPa After High Fluence Irradiation.

Table III-3. Properties of High-Z Materials at 500 K

Property	Tungsten	Tantalum
Atomic Weight	183.85	180.95
Density, g/cm ³	19.25	16.6
Melting Point, °K	3683	3269
Thermal Expansion, $\times 10^{-6}$ K ⁻¹	4.2	6.58
Thermal Conductivity, W/m·K	145	61.8
Heat Capacity, J/kg·K	138	148
Modulus of Elasticity, GPa	398	176

The non-metals C, SiC and BeO all have relatively high thermal conductivities at low temperature. However, radiation reduces the conductivity of these types of materials. Although considerable effort has been expended to develop high conductivity SiC, it is not clear that the high conductivity can be maintained in the radiation environment. The thermal conductivities of these types of materials generally decrease significantly as the temperature is increased above room temperature. For example, the thermal conductivity of the CVD β SiC is about a factor of 2 higher ($67 \text{ W/m} \cdot \text{K}$) at 25°C than at 525°C . Under irradiation the conductivity tends to become insensitive to temperature at values slightly below the high temperature values. Other compounds such as B_4C and Al_2O_3 show similar behavior. Limited data also indicate that the low temperature thermal conductivity of BeO decreases rapidly with radiation fluence.

Tungsten and tantalum are of interest for the plasma side materials because of their low light-ion sputtering yields at low plasma edge temperatures, their relatively high threshold energy for DT sputtering, and their relatively high energy ($\sim 700 \text{ eV}$) at which self-sputtering exceeds unity. Values for the physical properties of tantalum and tungsten at temperatures in the range of interest are summarized in Table III-3. Only limited data exist on the effects of radiation on the properties of tungsten and tantalum. The ductile-brittle transition temperature (DBTT) of tungsten is increased from 65 to 230°C after fast reactor irradiation at 385°C to fluences of $4\text{--}9 \times 10^{25} \text{ n/m}^2$. At higher fluences ($1.5\text{--}4.4 \times 10^{26} \text{ n/m}^2$) and similar temperatures the uniform elongation of tantalum was reduced to $\sim 0.1\%$ with a total elongation of 8-10%. Limited data indicate excellent low-cycle fatigue resistance of tantalum at temperatures below 732°C .

The available data on the physical and mechanical properties of plasma side materials are adequate for conceptual design with these materials, provided the data are used with the awareness of the product-form dependence for the low-Z candidates. However, detailed design will require qualification testing and measurement of critical properties on prototype component materials. The effects of neutron irradiation on the critical properties are not known well enough for prediction of service behavior. The strong dependence of property response on irradiation temperature will require the irradiation of appropriate product form of the low-Z materials under conditions judged to

adequately simulate service conditions. Important property measurements must include conductivity and strength properties, using tests specified to demonstrate the ability of the material to withstand service stresses. Changes in physical properties of high-Z candidates during irradiation are not a concern, but verification of adequate ductility after irradiation may require additional experimental work.

III.3 Heat Sink Materials

The heat sink must provide structural support for the plasma side material (if used) and provide for heat removal, normally by flowing a coolant through the heat sink. The material used for the heat sink should exhibit good strength and ductility, high thermal conductivity, good fabricability, and reasonable cost. In the long term, the heat sink material should also exhibit good radiation damage resistance. For the next generation of devices, the heat sink material is almost certain to be a copper alloy, and the coolant will be low temperature water. For DEMO and beyond, refractory metal alloys of V, Nb, Ta, and Mo are also being considered. The coolant could be either water or a liquid metal, such as lithium.

III.3.1 Copper Alloys

III.3.1.1 Physical and Mechanical Properties

Copper and copper alloys are leading candidates as the heat sink material. They are characterized by high thermal and electrical conductivity and, in general, corrosion resistance to flowing water. There is a large industry built up to produce and fabricate copper forms and products. Unfortunately many copper alloys have not had their physical and mechanical properties adequately measured, particularly at temperatures of interest: room temperature to 500°C.

Table III-4 gives a listing of candidate heat sink copper alloys including composition, room temperature yield strength and thermal conductivity. Copper alloys, as with most materials, can have their physical and mechanical properties changed widely by thermal-mechanical treatments (TMTs). Copper alloys tend to have their cold work and precipitation hardening effects annealed out by temperatures in the 200-550°C range. Thus, for some

applications the use of cold worked and aged copper alloys may not be possible without overaging.

Of the nine copper alloys listed, only OFHC Cu and CuAg can be considered to have an adequate unirradiated physical property data base for the temperatures of interest.

The physical properties of the other seven alloys have, in general, been poorly measured or only estimated from values measured for similar alloys or extrapolated to higher temperatures from low temperature measurements taken for the same alloy. The physical properties: thermal conductivity (k_T), electrical conductivity (k_e), specific heat (C_p), coefficient of thermal expansion (α), Poisson's Ratio (ν) and elastic modulus (E) all need to be measured from room temperature to 500°C for the copper alloys (except for OFHC Cu and CuAg) shown in Table III-4. In some cases the alloys need to be measured in several TMT conditions. These would include such conditions as annealed; solutionized and aged; solutionized, cold worked and aged; and solutionized, aged and cold worked.

The mechanical properties of copper and copper alloys have not been adequately characterized. The alloys OFHC Cu and CuAg have been studied a great deal more than any of the other candidates. As with the physical properties, a great deal of the available data is for room temperature properties. Because of the relatively low temperature nature of copper alloys, creep and creep-fatigue behavior is important even at temperatures as low as 150°C.

Present-day devices operate under low thermal loadings. Devices being designed and built now will experience higher loadings, especially in heat sinks for limiters, beam dumps and collectors for divertors. These greater thermal loadings will lead to higher thermal stresses, which combined with the electromagnetic and coolant stresses will force the selection of higher strength copper alloys such as CuBe alloys, dispersion strengthened alloys, or CuCrZrMg alloys.

The mechanical properties that most need to be measured for copper alloys include:

- o tensile properties as $f(T)$ from room temperature to 500°C
- o fatigue behavior (low cycle temperature to 500°C)
- o creep behavior
- o creep-fatigue interaction

Table III-4. Candidate Heat Sink Copper Alloys

Alloy	Designation	Nominal Composition Wt. %	0.2% YS ^a MPa	(ksi)	k _T ^a W/m-°K	°C	MP (°F)
OFHC Copper	C10100	Cu	70 ^b 310 ^c	(10) (45)	398	1083	(1981)
CuAg	C10700	Cu-0.085Ag	70 ^b 310 ^c	(10) (45)	388	1083	(1981)
CuZr	C15000	Cu-0.15Zr	360 ^d	(52)	349	980	(1796)
CuAgPMg	C15500	Cu-0.06Ag-0.06P-0.11Mg	475 ^d	(69)	348	1078	(1972)
CuZrCrMg	C18100	Cu-0.4Cr-0.15Zr-0.06Mg	510 ^d	(74)	320	1075	(1967)
CuAl ₂ O ₃	C15720	Cu-0.2Al ₂ O ₃	490 ^c	(71)	353	1082	(1980)
CuCr	C18200	Cu-0.9Cr	450 ^d	(65)	323	1070	(1958)
CuBeNi	C17510	Cu-0.4Be-2.0Ni	620 ^d 760 ^d	(90) (110)	249	1004	(1840)
CuBe	C17200	Cu-1.9Be	1105 ^e 1255 ^d	(160) (182)	94	865	(1590)

^a Room Temperature.^b Annealed.^c Cold Worked 37% (Hard).^d Cold Worked and Aged.^e Solutionized and Aged.

These mechanical properties, as with the physical properties, need to be measured for the alloys in a variety of TMT conditions. In some cases these TMT conditions will be defined by the fabrication steps necessary to form and join the heat sink materials.

III.3.1.2 Irradiation Effects

The irradiation data base for Cu and Cu alloys is sparse. Both neutron and ion irradiations have been utilized in the study of copper, but the fluence levels have been generally limited to less than 1 dpa. Swelling in pure copper has been measured at a fluence level of 0.3 - 0.5 dpa. The swelling is characterized by a peak at $\sim 325^{\circ}\text{C}$ and a swelling rate of $\sim 1\%/dpa$. Alloying with small amounts of aluminum, germanium, silicon, or nickel can result in reductions in swelling, whereas alloying with small amounts of silver or cadmium can increase swelling.

The effect of radiation on the mechanical properties of pure copper is to increase the tensile strength and reduce the ductility. Again, however, data exists only for low fluences, so that it is not possible to predict the changes that would occur at levels of 10-50 dpa.

The information needed to design high heat flux components in a neutron field is low-moderate temperature ($< 400^{\circ}\text{C}$) and low-moderate fluence (≤ 50 dpa) data on a variety of copper alloys. The irradiation properties of greatest interest are swelling, radiation creep, tensile properties, and fatigue/crack growth. A major obstacle to obtaining this information is the lack of low temperature irradiation test facilities in the United States. The two candidate reactors are ORR and HFIR located at ORNL. However, ORR is capable of producing only ~ 7 dpa/y. HFIR is capable of producing ~ 30 dpa/y, but only in high flux regions where the temperatures cannot be easily controlled. Considerable work is required to develop instrumented assemblies for temperature control in test reactors.

III.3.1.3 Coolant Compatibility

While copper structure would not be compatible with a liquid metal coolant, copper and its alloys should be corrosion resistant to pure water in the absence of oxidizing agents. However, this resistance would be severely reduced if small concentrations of hydrogen peroxide and oxygen are present in

the water from radiolysis. The magnitude of radiation-induced effects needs to be established by analysis of existing data and by experiments with varying concentrations of oxygen and hydrogen peroxide. The use of a hydrogen overpressure to minimize the concentration of oxidizing may lead to embrittlement of the copper or copper alloy. Another effect that has to be clearly established for the appropriate alloys is that of water velocity. At high flow rates, corrosion-erosion may occur. While no great effect of composition variation on the general corrosion resistance of copper alloys has been observed, alloying can improve erosion resistance at high flow velocities. However, while alloy development for far-term applications may be of help in producing a more corrosion- and erosion-resistant copper alloy, the development of appropriate methods for the control of water chemistry would probably have the greatest effect on limiting the corrosion of copper and copper alloys.

III.3.2 Refractory Metals

III.3.2.1 Physical and Mechanical Properties

The refractory metals form a class of materials that fall into groups VB and VIB of the periodic table. They are all body centered cubic (b.c.c.) in atomic structure. The group VIB metals consist of chromium, molybdenum, and tungsten, and group VB metals consist of vanadium, niobium, and tantalum.

Refractory metals are characterized by good thermal conductivity, low thermal expansion, high melting point, and generally high reactivity. The combination of good thermal conductivity and low thermal expansion gives this class of metals a capability to accommodate high heat fluxes. The high melting points mean that these metals can be used at much higher temperatures than are possible with copper alloys. Refractory metals are considered to be specialty metals, and, as such, there is a limited production capability. Table III-5 gives a listing of the candidate alloys along with their composition, yield strength, elongation, thermal conductivity, and melting points.

The physical properties of refractory metal alloys are generally not well characterized. In many cases, property values are estimated from data taken using similar alloys or at different temperatures. The physical properties of interest are the thermal conductivity, electrical conductivity, specific heat,

Table III-5. Candidate Refractory Metal Alloys

Alloy	Composition	0.2% YS MPa	ksi	Elongation ^a %	k_T^a W/mK	MP °C	$\alpha \times 10^{-6}/K$
V-15Cr-5Ti	15-Cr 5-Ti Bal.-V	500	72.5	27	24	1900	9
Nb-1Zr	1-Zr Bal.-Nb	200	29 ^b	35 ^b	45	2468	6
FS-85	27-Ta 10-W Bal.-Nb	475	69	24	45	2468	6
TZM	0.5-Ti 0.08-Zr Bal.-Mo	760	110 ^c	16 ^c	115	2610	5
Ta-10W	10-W Bal.-Ta	1100	160 ^d	10 ^d	53	2996	7
T-111	8-W 2-Hf Bal.-Ta	895	130 ^e	15 ^e	53	2996	7

^a Room Temperature.

^b 20% Cold Worked.

^c Fully Annealed.

^d As Wrought.

^e Stress Relieved 1 h at 1100°C.

coefficient of thermal expansion, Poisson's ratio, and elastic modulus. The mechanical properties of these materials are also poorly characterized. The tensile properties (ultimate tensile strength), yield strength, elongation have been measured for most alloys, but other properties, such as fatigue, creep-fatigue, and crack growth, have not been measured in many cases. In addition, the mechanical properties can vary substantially with the thermo-mechanical treatment, so that the properties should be measured for a variety of treatments in order to optimize the alloy structure.

All refractory metals experience the phenomenon referred to as the ductile-brittle transition temperature (DBTT). The DBTT is essentially a range of temperatures below which a ductile material behaves in a brittle manner and fails at very low strains. The group VIB metals which consist of chromium, molybdenum, and tungsten have a characteristic DBTT well above room temperature, while the group VB metals have a DBTT well below room temperature $\sim -200^{\circ}\text{C}$. This temperature range is not fixed but can be either raised or lowered depending upon chemistry or microstructure. For example, the interstitial elements such as C, O, N, and H can have a significant effect on the DBTT, in that it can be raised as these impurities are increased. Therefore, it is important to keep these impurities as low as possible not only during manufacturing and assembly, but also during the service life of these materials. Another way of altering the DBTT is through the use of cold working. Cold working creates a heavy dislocation microstructure which tends to increase the metals' tolerance for interstitial elements and results in a lower DBTT. It is for this reason that the metals molybdenum and tungsten are supplied in the cold worked and stress relieved condition. The large amount of cold work makes subsequent working with these metals extremely difficult and essentially makes welding impractical because of the increases in DBTT. For tungsten, cold working has been found to decrease the transition temperature from 420°C for sintered (recrystallized) tungsten to about 100°C with 95% cold working. Even though cold working significantly reduces the transition temperature of tungsten, it will still be above room temperature. This means that forming will have to be done hot. Alloying of tungsten and molybdenum with rhenium has also been found effective in reducing the DBTT. At low concentrations of rhenium, cold working is usually used to reduce the DBTT to lower levels than achieved in pure metals. At higher concentrations, the

material is usually supplied in the recrystallized conditions. These alloys can be welded. These compositions are W-25Re and Mo-50Re.

Tantalum, niobium, and vanadium have DBTT well below room temperature and, as a result, are alloyed for strength improvements rather than to alter the . Alloys of these metals are supplied in the recrystallized condition and are all weldable. Typical alloys which are commercially available and have a reasonable data base are Ta (Ta-10W and Ta-8W-2.5W [T-111], Nb (Nb-27Ta-10W-0.8Zr [FS-85] and Nb-10W-10HF-.01Y [C129Y] and V (V-20Ti and V-15Cr-5Ti).

III.3.2.2 Irradiation Effects

The irradiation effects data base for refractory metals is limited, although it is somewhat more extensive than the data base for copper alloys. The irradiation properties of greatest interest are swelling, embrittlement, and creep. The swelling in pure refractory metals has been measured, but in most cases, swelling in the candidate alloys has not. Pure niobium, vanadium, and tantalum exhibit swelling peaks at 600, 550, and 650°C respectively. Swelling in pure molybdenum appears to be relatively independent of temperature in the range from 430-700°C. There is some evidence which indicates that refractory metals (b.c.c. structure) are inherently more swelling resistant than alloys with an f.c.c. structure. Alloying of refractory metals, such as the addition of titanium to vanadium, can reduce or eliminate swelling. The major concerns with embrittlement are the upward shift which may occur in the ductile-brittle transition temperature (DBTT), and the possible occurrence of plastic instability. In the case of molybdenum, the DBTT can be raised to ~ 400°C by neutron irradiation, while in the case of the vanadium alloy, VANSTAR-7, plastic instability was observed at room temperature in samples irradiated to only .0095 dpa. Radiation creep has not been measured for any of the refractory metals.

III.3.2.3 Coolant Compatibility

Refractory metals such as niobium, tantalum, vanadium, and molybdenum and their alloys, are highly resistant to dissolution by liquid metals. In certain cases, severe localized attack of niobium and tantalum by alkali metals can occur when the oxygen concentration of these refractory metals exceeds several hundred parts per million by weight. However, this type of attack can

readily be prevented by careful handling, alloying (Zr in Nb and Hf in Ta), and proper heat treatment. Vanadium has not shown this type of "oxygen disease" except for abnormally high oxygen concentrations which produce sub-oxides in vanadium. Apart from this corrosion reaction, the major general compatibility concern for refractory metals and alloys exposed to alkali metals is interstitial transfer (carbon, nitrogen, oxygen) and the subsequent effect on materials properties. Examples of such reactions include oxidation of refractory metals in sodium or potassium of high oxygen activity, thermal gradient-induced transfer of nitrogen or carbon from the hot section of a refractory metal loop to a cooler part, and the carburization or nitridation of the refractory metal (alloy) when a liquid metal circuit includes both the refractory metal and a ferrous alloy (in the lower temperature region). While such reactions have not been studied for every refractory metal alloy-liquid metal couple, sufficient work has been done (particularly for lithium, to understand the most important compatibility reactions and to know how to minimize their effects. Needed work in this area would simply be testing an optimized system under prototypic conditions, paying particular attention to the effects of the projected higher flow velocities.

Relatively little is known regarding the corrosion of refractory metals and their alloys by water. Recent short-term tests of vanadium alloys in water at 260, 273, and 300°C have yielded encouraging results in that only small weight gains were measured. Presumably, the corrosion of such alloys was reduced by the presence of a thin oxide film. However, data for longer exposures under varying chemical conditions will be required before a particular alloy is deemed suitable for long-term service in water.

While vanadium and other refractory metals can readily absorb hydrogen (which can be present from the reduction reaction or from intentional additions to decrease the oxygen concentration of the water), analyses of vanadium alloys exposed for 100 h to 300°C water with a hydrogen overpressure did not reveal any significant increases in the hydrogen concentration of the specimens. It was thought that the oxide layer on these alloys also played a beneficial role in this regard by reducing hydrogen absorption. Again, longer-term tests with the appropriate alloys are required to determine hydrogen effects over extended exposure periods.

III.4 Interface/Attachment

III.4.1 Characteristics and Properties

Many high heat flux components will require a bond between the plasma side material and the heat sink. These materials may have very dissimilar physical and mechanical properties, and therefore they may be difficult to successfully bond together. There are many types of bonds, including brazes, diffusion bonds, explosive bonds, and inorganic adhesives, which may be considered for application to high heat flux components. Fabrication techniques for several of these bonds are discussed in Sec. III.5.

There has been extensive work performed on the bonding of different materials, but in general the bonds are developed for specific materials combinations and operating conditions. Therefore, the use of previous bond development results to predict the bond properties of fusion high heat flux components is of limited value where the materials combinations and environments are unique. Specification of all the processing variables which characterize the joining process will determine the compatibility of each materials combination. Given these complexities, it is clear that attempts to model such systems from the established data base cannot be performed with any degree of confidence. Furthermore, the residual stress distributions that characterize a particular joint after fabrication are extremely complex since they will depend upon the detailed knowledge of the materials properties and time-temperature history during the fabrication process. The thermal stresses that develop during fusion reactor operations will superimpose on the residual fabrication stresses and change the stress state in ways that are difficult to predict. Thus, experimental confirmation of component performance under a realistic simulation of the operating environment will be needed to evaluate bonds for high heat flux components.

III.4.2 Irradiation Effects

III.4.2.1 Results of Previous Work

The largest body of information found in the literature on the irradiation effects to the interface/attachment region between a coating and a substrate material was written by Johnson, et al.,^(2,3) for the development of

low friction coatings for the breeder reactor. From this rather extensive program, valuable insights can be gained from the results of irradiation tests on over one hundred coating/substrate combinations as well as practical experience regarding the length of time and the amount of effort required to successfully qualify a coating/substrate system for in-reactor application. While many of the design goals for developing materials for breeder reactor applications were unique (e.g., sodium compatible, low friction, close dimension tolerance), others such as mechanical integrity, irradiation resistance and thermal cycling are very similar to those of high heat flux component designers.

The following observations have been made:

- (1) Irradiation testing of coated materials eliminated more coatings than any other qualification test (e.g., thermal cycling under stress, epoxy lift off, environmental). This was in spite of the fact that the thermal cycling tests, for example, involved a larger number of cycles under more severe thermal conditions than those experienced in the irradiation tests. Irradiation effects, therefore, must be a major consideration in the testing of composite materials.
- (2) No clear correlations between temperature, fluence and coating damage were apparent.
- (3) The ability of the mechanically bonded coatings, such as plasma-sprayed and detonation gun coatings, to survive irradiation appeared to be related to coating bond strength, interface stresses, and internal coating stresses. The high strength detonation gun coatings with minimum thermal expansion differences maintained their integrity while most plasma sprayed coatings of the same composition but having lower bond strengths, failed by spalling. "Graded" plasma sprayed coatings were more resistant to irradiation-induced failures than other plasma coatings, and offered potential for further development.
- (4) Metallurgically bonded coatings, such as those produced by diffusion coating, electro-spark deposition, explosive bonding, or weld deposition showed no observable deleterious effects of irradiation (cracking, spalling or unbonding).

The mechanisms of irradiation damage in the coated materials are not well understood. When postirradiation tests are positive, assurance is gained that the materials will perform adequately; however, when specimens fail the irradiation test the cause of failure is often open to speculation. The damage mechanisms that operate in metallic materials should also produce damage in coatings. For example, swelling or differential swelling in composite coatings may produce coating failure. Some probable mechanisms (all of which require study) that could result in coating failure are listed below:

- (1) Point defect or transmutation induced stress at coating substrate interface, or within the coating.
- (2) Irradiation induced changes in mechanical and thermo-physical properties of both coating and substrate. This is particularly important if the duplex system must withstand deformation.
- (3) Impurity and transmutation product migration to the interface region.
- (4) Gas migration when interface is under stress or maintains a thermal gradient.
- (5) Intra-coating stresses as a function of coating thickness.
- (6) Differential swelling and thermal expansion.

III.4.2.2 Recommendations

Because of the sensitivity to process variables and the complexity of duplex or triplex systems, theoretical predictions of irradiation performance are of limited value. In order to adequately evaluate the integrity of the interface between the coating and substrate, appropriate test specimens must be subjected to a series of qualification tests. Previous experience in developing and qualifying coatings for breeder reactor applications shows that lead times of 6 to 8 years may be required to screen, test and qualify coating substrate combinations for irradiation performance to a level of 20-30 dpa.

The time required for irradiation testing of HNF materials is a function of the test temperature and fluence, specimen configuration and size, and the irradiation facility. Currently, relatively large HNF specimens (1.5 cm dia. x .1-1 cm thick) are being irradiated in the low flux zones of the FFTF and EBR-II reactors. The minimum bulk coolant temperature in these reactors is approximately 370°C which defines the lowest achievable temperature. It is anticipated that HNF testing will require irradiations in the 100-300°C range. Therefore, water cooled mixed spectrum reactors, the thermally controlled closed test loops in the FFTF, or eventually the FMIT, must be utilized.

While currently operating fission reactor facilities will provide significant defect damage (dpa) and irradiation volume, they will not produce the correct solid and gaseous transmutation products which are expected to strongly effect the mechanical and physical properties of the interface region. The FMIT, projected for operation in early 1990, will provide the correct transmutation to dpa ratios but within a very limited testing volume. All of the current and proposed test facilities have advantages and limitations which must be carefully assessed when planning our HNF irradiation strategy.

III.5 Fabrication

III.5.1 Plasma Side Materials

There are two alternative approaches to fabricate 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.

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 claddings are related to the attachment of the cladding to a heat sink. These attachment issues and associated research requirements are discussed in a subsequent section of this document.

Coatings offer distinct advantages over claddings 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 damage or eroded surfaces. At present, the primary technology for coating high heat flux components is chemical vapor deposition (CVD) of thin ($< 20 \mu\text{m}$) 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 low deposition rates, 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 and 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.

In summary, 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 may be important for advanced fusion machines. However, it is too early to clearly evaluate the prospects for successful development of this technology.

III.5.2 Heat Sink Materials

III.5.2.1 Copper Fabrication

The candidate copper alloys listed in Table III-4 are all capable of being fabricated to various degrees. Most copper alloys are readily available, although they may have to be special ordered as many forms are not available directly from stock. That is, operations such as machining, forging, shaping, welding, brazing and other joining methods are compatible with them. All of the alloys, if in the cold worked condition, will be annealed by welding or high-temperature brazing. Thus, the selection of a specific alloy is highly dependent upon the fabrication methods to be used.

The most likely fabrication problems for copper heat sink materials are how to provide internal coolant channels in the heat sink shapes, attach them to substrates and attach low-Z tiles or coatings to them. The last two subjects will be covered in the next section. If the plate is thick enough and simple in geometry, internal coolant channels can be formed by simply drilling coolant passages and manifolding them. Devices with more complex shapes or limited thickness may be forced to use techniques such as brazing, diffusion bonding, electroforming or explosive bonding to close out the coolant channels. All of these joining techniques have been used with success on copper. High temperature brazing ($> 600^{\circ}\text{C}$) will result in significant strength reduction in all alloys except alloys already annealed or dispersion strengthened alloys (such as CuAl_2O_3). The CuBe alloys are unique in that they do not need cold work in order to develop high strength. Thus, an over-aged material can be resolutionized and aged to recover its strength.

Since all the candidate alloys have high thermal conductivity, welding is difficult because of the heat sink effect of the surrounding material. This is especially true for thicker sections. Electron beam welding is very well-suited for welding copper as its high power density allows the welding to be done without annealing the bulk of the material. Laser welding is only possible in the thinnest of sections because of the reflectivity of copper alloys. More conventional welding methods (TIG, arc, etc.) are possible but result in severely annealing the area around the weld. Thus, they are suitable only for annealed copper or alloys capable of being hardened without cold work, such as CuBe alloys.

Programs are needed in the following areas in order to fabricate heat sinks for use in the next generation of devices:

- (1) Investigation of techniques to allow welding of copper alloys with retention of a high percentage of original strength and/or to localize softening region to immediate area around weld.
- (2) Development of brazing techniques for high-strength copper alloys. Investigate newer low-temperature braze alloys for use with copper alloys. Develop techniques and guidelines to minimize overaging of cold worked structures. Examine effect of operating temperature on braze strengths.
- (3) Electrodeposition of copper provides an alternate fabrication method for heat sinks. Methods of producing high-strength electrodeposited copper that can operate at temperatures above 250°C need to be investigated.

When conducting any program to investigate fabrication of heat sink materials, close consideration must be placed on methods to be used in attaching the primary thermal loading component to the heat sink. The selected joining method must be compatible with the heat sink fabrication method.

III.5.2.2 Refractory Metals

Primary Fabrication

The combination of a high ductile-to-brittle transition temperature and resistance to deformation makes the processing of the group VIB metals entirely different from that of the VB metals. Initially with the advent of advanced casting techniques, it was thought that arc casting could be used to consolidate tungsten, thereby substantially increasing the sizes available in tungsten sheet or plate over what could be produced by the powder metallurgy route. Unfortunately, the large grain size coupled with tungsten's resistance to deformation make initial breakdown or blooming of the cast ingot extremely difficult because of the very high temperature needed to roll tungsten and the large mill separation forces to prevent alligating of the slab. For this

reason, arc casting was abandoned for the production of sheet and plate in favor of powder metallurgical techniques. Today, virtually almost all tungsten is manufactured via the powder metallurgy process. In this process fine grain tungsten powder is either hydraulically (mechanical) or isostatically pressed into a billet which is subsequently sintered in hydrogen at around 2500-2700°C.

Initial breakdown is usually accomplished at 1500°C with the temperature slowly being reduced with increasing cold work to prevent recrystallization which would raise the DBTT. For fusion, the transition temperature should be 250°C or lower to reduce the potential for fracture during a plasma disruption. For the transition temperature to be this low will require about 80% cold work. To put this much cold work into tungsten would require a combination of forging and rolling at fairly high temperatures which will limit the weight and size of the part that can be produced. Currently the maximum size that can be produced without substantial scale-up is 50 x 50 cm for a 1 cm thick plate and 35 x 35 cm for a 2 cm thick plate. While it is technically feasible to produce larger parts, the increased cost and risk in increasing the transition temperature does not appear to warrant this scale-up for parts much larger than currently available.

Molybdenum is processed in much the same way as tungsten, however, molybdenum and its alloys can be wrought from arc cast ingots and, consequently, can be made into sheet and plate roughly twice the size of tungsten sheet and plate. In obtaining sheet and plate in these two materials, there are a number of factors that limit these size rangings from supplier furnace temperature and size limits to the weight of the part itself. Since tungsten is almost twice as dense as molybdenum, handling of large components is difficult and this also limits size.

The group VB refractory metals are extremely ductile and can be readily processed from either arc cast or electron beam melted ingots. The "as cast" ingots are then forged into slabs or sheet bars for subsequent rolling at temperatures around 400°C. These slabs which range in size from 5 to 10 cm in thickness are then rolled on conventional rolling mills to the designed thickness. The final product is then fully annealed in a vacuum. Because of these materials' high ductility (30-50% elongation), the large reductions per pass needed to prevent alligatoring in tungsten and molybdenum are not required.

The net result is that larger sheets can be produced on the same piece of equipment because of lower separation forces on the rolling mill. It also means that tantalum and niobium can be processed on any rolling mill in the country capable of rolling large stainless steel plates. Currently the maximum size of tantalum plate available is roughly 60 x 120 cm for 1 cm thick plate and 60 x 60 for the 2 cm thick plate. Each of these plates weigh approximately 125 kg. The primary limitations in the size of tantalum available is in the pickling operation. Because of tantalum's resistance to corrosion (it's resistant to most acids including aqua regia), only extremely concentrated acid solutions containing large concentration of hydro-fluoric acid can be used. These concentrated acid solutions complicate the processing of large, heavy plates with conventional handling equipment. Since ingots weighing up to 2000 kg can be cast and forged into slabs, the processing of larger plates is technically feasible except it will require capital expenditure for additional equipment to process the larger plates. However, in producing the larger plates, there will be a cost increase over current plate costs. The same limitations that apply to tantalum alloys are also valid for niobium alloys.

Secondary Fabrication

The refractory metals have been successfully fabricated in a number of complex shapes. However, the bulk of the fabrication has been on relatively thin structures (< 2 mm); very little work has been done on thicker structures. The most extensive fabrication of heavy gage (> 1 cm) has been done on Nb-12r in the early 1960's. In this processing, the same equipment and techniques used for high strength steels were applied to the Nb-12r. In fabricating tantalum and niobium alloys today, the same techniques would be used. For tungsten and molybdenum, the processing will be at temperatures approaching those used in primary fabrication.

Satisfactory braze joints have been obtained in both molybdenum and tungsten using vacuum and inert atmosphere furnaces. The bulk of the brazes developed have been for high temperature applications (> 1000°C) and little information is available for low temperature brazes. In high temperature brazing, care has to be exercised not to recrystallize the tungsten or molybdenum which will result in an embrittled structure. There appears to be a

uniform lack of property data on braze joints made with refractory metals, and those that do exist are for short term tests. The bulk of brazing work to date has been for refractory metals joined to refractory metals, however, there is some information available regarding brazing of molybdenum.

Investigators have tried for a number of years to develop techniques to weld tungsten and molybdenum. In all cases even though successful welds have been made, they usually crack because of residual stresses. The only successful welds have been those made with a tungsten-rhenium or molybdenum-rhenium alloy. Because of the difficulty in welding tungsten and molybdenum and the brittle nature of their welds, this approach is not feasible.

Tantalum and niobium are weldable and a number of large structures have been fabricated using a variety of techniques. The primary concern is the contamination by interstitials. To prevent contamination, welding is usually performed either in an inert atmosphere or vacuum chamber. In instances where the structure is too large to be fitted into a chamber or bubble, ductile welds have been obtained in air by flooding the weld zone with an inert gas and using a trailing shield to protect the weld zone during cool down. For welding sheet thicknesses up to 1.5 mm, post weld stress relieving is usually needed to minimize distortion. While tantalum can be readily welded to itself, welding it to other metals, with the exception of niobium and vanadium alloys, has been largely unsuccessful. There are a variety of reasons for this but the two primary ones are that the melting point of tantalum is usually above the boiling point of the metal it's to be welded to or that the resultant joint contains brittle intermetallic compounds or forms low melting eutectics.

III.5.3 Attachment

The most promising techniques for joining dissimilar materials for high heat flux components are brazing, diffusion bonding, coating, inorganic bonding agents and mechanical attachments. The following discussion summarizes the state of knowledge with respect to the various attachment techniques mentioned.

Brazing techniques have been widely used to form both metal-metal and ceramic-metal joints. Brazing offers a great deal of promise for the fabrication of high flux components since a wide variety of commercial brazes are

available and techniques for their use have been established. However, very little experience has been gained in the fabrication of the particular materials combinations needed for high heat flux components. Thus, a considerable effort will be required to establish the optimum material combinations, geometry, braze alloy selection, joint design and brazing cycle that will perform successfully under fusion reactor conditions. Almost no data base is available on testing of brazed high heat components to permit any predictions of interface performance under fusion reactor conditions. Since brazing offers a great deal of flexibility in the choice of different materials combinations, it should be given a high priority for further evaluation. However, brazing is not a preferred method for materials combinations where one of the materials is much thinner than the other, as in a thin coating. Brazing may also be difficult where complex contours are required. Scale-up may also be a major problem. Brazing approaches should be considered for both near term and far term machines.

Diffusion bonding like brazing has been used for both metal-metal and ceramic-metal joining. Diffusion bonding requires simultaneous application of both pressure and heat during the bonding process. Thus, diffusion bonding is often more expensive than brazing and there are practical limitations on the size and geometry of the surfaces to be joined. Diffusion bonding can produce very high quality joints, and maximum service temperatures are often higher for diffusion bonds than for comparable brazed joints. In some cases, process temperatures for diffusion bonding are lower than brazing temperatures for a comparable joint. This is particularly attractive for attachment to copper alloy heat sink material that may develop undesirable microstructures at typical brazing temperatures. At present little is known about diffusion bonding of potentially interesting ceramic-metal joints. A major problem for both diffusion bonding and brazing of ceramics to metals is the typically large mismatch in the thermal expansion. The database for metal-metal diffusion bonding is somewhat better, but it is still very limited compared to the data available for metal-metal brazing.

Although mechanical attachments for high heat flux components are attractive from the standpoint of servicability and ease of replacement, serious questions remain about their ability to handle the high heat loads anticipated in a fusion reactor; thus, it is not anticipated that mechanically attached

schemes will find widespread application for high heat flux components. However, mechanical attachment schemes may be considered for near term machines or for those components which receive the lowest heat loads.

A lower level effort should be expended to identify and evaluate some promising inorganic adhesives for interface attachments such as metal or glassy based bonding agents, which cure at relatively low temperatures. Such adhesives would offer the advantage of ease of application over large and complex surface areas and low fabrication temperatures. They may suffer the disadvantage of not having adequate bond strengths for this application. Inorganic adhesives could be developed for near term machines and then evaluated further for advanced machines.

III.6 Other Issues - Tritium Permeation

Tritium permeation through first walls, limiters, or divertors subjected to energetic tritium charge exchange neutral bombardment is a potentially serious problem area for advanced D-T reactors operating at elevated temperatures. High concentrations of tritium in the near surface region can be reached by implantation of the charge exchange neutral flux combined with a relatively slow recombination of these atoms into molecules at the plasma/first wall interface. Because of this large concentration of mobile tritium near the inner (plasma) wall surface, a concentration gradient is established, causing tritium to diffuse into the bulk and eventually to the outer wall surface where it can enter the first wall coolant. Calculations have shown that the choice of inappropriate materials (e.g. V for the MARS direct convertor) could lead to tritium permeation as large as 10^8 Ci/day.

Current modeling of tritium permeation is performed by numerical solution of Fick's Law including a tritium source term and recombination boundary conditions. The model has been tested extensively and gives good agreement with laboratory experiments. Central to the model is the recombination constant. A simple formula is used to estimate this constant from a combination of bulk properties (diffusivity and solubility) and surface condition. This formula has been verified experimentally for austenitic stainless steels, iron, inconel, nickel, vanadium, and titanium. Variation of surface conditions has been shown to affect permeation in a predictable manner.

Two types of experimental data for high heat flux materials is needed. First, for many materials such as TiC, graphite, etc., the hydrogen solubility and diffusivity is not well known. Knowledge of these properties makes it possible to obtain a preliminary estimate of tritium permeation. Second, for those materials of serious interest, a direct measurement of recombination constant is needed. Since this constant depends strongly on both temperature and surface conditions, a measurement with conditions appropriate for machine operation must be obtained. These measurements are currently being made by measuring recycling in operating machines. Unfortunately this procedure has the disadvantage of having to build the machine before the tritium permeation problem has been addressed. Alternatively, the measurements can be taken in a plasma-driven permeation apparatus. This technique has the disadvantage of trying to simulate machine-relevant surface conditions.

The above discussion applies to materials in which hydrogen transport is by atoms. For materials such as glasses, oxides, etc. in which the transport process occurs through molecules, the above arguments are significantly modified. No modeling (or experiments) for such materials is currently available. If such materials are proposed for high heat flux application, an extensive experimental effort would be required to determine the underlying processes that control plasma-driven permeation.

III.7 Materials and Processes Technology Needs

The materials and processes data base assessment indicates that information is needed in several areas. The development areas can be divided as shown below.

Baseline Property Data

- Thermophysical
- Mechanical
- Corrosion
- Surface

Fabrication Technology

- Primary Fabrication
- Secondary Fabrication
- Bonding

Irradiation Damage Effects
Thermophysical Properties
Mechanical Properties
Bond Integrity

Each major area applies to the plasma side materials, heat sink materials, and bonds.

Baseline experimental data provides important information concerning the potential operating limits of different materials. This information is required as input to design studies and analysis which will in part determine the optimum configuration for high heat flux components. The amount of information needed to completely characterize a material is extensive, and the materials described in this section are all lacking in some important data. The most important properties for high heat flux components are thermophysical, mechanical, corrosion, and surface properties. Thermophysical properties include thermal conductivity, thermal expansion, elastic modulus, and specific heat. The primary variables are temperature and fabrication procedure. The mechanical properties include tensile properties (ultimate strength, yield strength, total elongation, uniform elongation), fatigue/crack growth, thermal creep, and creep-fatigue interactions. The primary variables are temperature, stress, strain rate, and fabrication procedure. The corrosion areas include coolant corrosion, stress corrosion cracking, and compatibility between different materials. The important variables are temperature, stress, coolant chemistry, material composition, and fabrication procedure. The surface properties include sputtering (D, T, He, self), particle implantation effects (D, T, He), surface structural properties following sputtering, surface chemistry (stoichiometry changes, surface segregation), and properties of redeposited materials. The important variables are particle bombardment energy, particle flux, and material temperature. The surface data base assessment is covered in detail by the PMI task group data base assessment and will not be considered any further in this section. It is clear that the large number of properties to be studied combined with the list of candidate materials results in a potentially extensive test matrix.

Fabrication technology data provides information to the designer about the practical limits for the manufacture of high heat flux components. The major areas of fabrication technology are primary fabrication, secondary fabrication, and bonding. Primary fabrication consists of the processes involved in transforming raw materials into consolidated forms, such as ingots, billets, plates or sheets. Secondary fabrication consists of transforming the products of primary fabrication into finished products. These processes include machining and forming. Bonding processes include welding, brazing, diffusion bonding, etc. Often the materials properties are highly dependent upon the fabrication processes. Therefore, it is important to test materials in the form in which they are likely to be used in the fusion reactor.

Irradiation damage will degrade the baseline material properties and will be an important consideration for high fluence devices. It is generally desirable to study the effects of radiation on all the properties mentioned above for a range of fluences and irradiation temperatures. In addition, microstructural development during irradiation (swelling, dislocation structure, precipitation), will need to be investigated. The potentially large number of tests along with the high costs and long lead times of irradiation experiments means that careful planning is a necessity.

The research and development needs for materials and processes vary with the time frame of the fusion device. In the remainder of this section the development needs and critical issues will be identified for the different fusion concepts.

In present devices, the material development needs are minimal. Recently work has begun on the design and fabrication of beryllium limiters for ISX, and potentially, beryllium limiters will be installed in JET. Thus, the new development required for present machines is for the fabrication and use of beryllium. There is considerable experience in the fabrication of beryllium components, so that no major development programs are anticipated.

In near term devices and ignition devices, there are major development needs. Both tokamaks and mirrors will employ large, actively cooled, high heat flux components that are exposed to the plasma. The primary goals of the development are the selection of the reference plasma side and heat sink materials and the identification of the most appropriate fabrication procedures. In order to reach these goals, major programs are needed to obtain baseline property data and fabrication data.

The choice of a reference plasma side material will depend on the surface and bulk properties along with the device operating conditions. At the present time, low-Z materials are preferred for use in tokamaks. The candidate materials are Be, C, BeO, and SiC, and there are critical issues associated with each material. Graphite has many desirable properties, but it is expected to interact chemically with the hydrogen plasma. The critical issue for graphite is whether sputtering erosion enhanced by the chemical interactions can be tolerated by the tokamak plasma. A related issue is the redeposition of sputtered material. It is not clear that redeposited graphite will be adherent or whether it will exhibit satisfactory bulk properties. Both of these issues need to be examined before graphite can be considered to be a viable plasma side material. It should also be mentioned that small additions of other materials, like SiC, to graphite may alleviate the chemical sputtering problem, and such graphite "alloy" should also be tested. Beryllium is not expected to exhibit chemical sputtering, and thus it appears to be a satisfactory material for normal operation. However, there is concern about the use of beryllium during disruptions. Beryllium, being a relatively low melting point and high thermal conductivity material, may form a thick melt layer during disruptions. The major issue for beryllium is the potential rapid loss of material due to the formation and loss of the melt layer. SiC and BeO are refractory compounds. A major issue for compounds is whether sputtering occurs preferentially for the individual elements. Preferential sputtering would lead to a change in the surface composition and possibly to a change in the surface properties. All of these surface related issues are covered in greater detail in the PMI technical assessment,⁽⁴⁾ but they are mentioned here because they are critically important to the selection of a plasma side material. Baseline data is also needed to describe the bulk properties of plasma side material. The non-metals, C, BeO, and SiC, require special attention because their properties can vary significantly depending upon the fabrication procedure. Both the thermophysical properties and the mechanical properties need to be measured over a range of temperatures and stresses.

The heat sink material in near term devices is most likely to be a high thermal conductivity copper alloy. However, a reference alloy has not yet been selected, because in many cases, the baseline data base is inadequate. Thermophysical and mechanical properties should be measured for a variety of

alloys over a range of temperature stress, and strain rate conditions. If no commercial alloy provides all of the required properties, then development of a new alloy may be needed.

An integral part of the development for near term devices is the fabrication of high heat flux components. The fabrication of separate materials is generally well developed, but the fabrication of structure incorporating both plasma side and heat sink materials requires development. The key issue related to fabrication is whether metallurgical bonds between plasma side and heat sink materials can be successfully and consistently produced. A related issue is whether the bond can be fabricated without adversely affecting the bulk properties of the heat sink. Several copper alloys gain much of the strength by cold working, and the temperatures needed for bonding may anneal the structure and reduce the strength. Because bonds are developed for specific materials and applications, a different bonding technique may be required for every plasma side material.

In general, materials and processes needed for mirrors will be the same as those used in tokamaks. Besides the materials already mentioned, high-Z materials such as molybdenum and tantalum are being considered as both plasma side and structural materials. Work will be needed in the areas of mechanical property evaluation and fabrication development for these materials.

High heat flux components in reactor prototypical devices will be subjected to moderate to high neutron fluences. The critical issue for this class of devices is whether these components can maintain acceptable properties following irradiation. The additional requirement of radiation damage resistance may have a significant impact on the choice of materials. For example, non-metal candidates for plasma side materials, such as C, SiC, and BeO, may be eliminated because of rapid degradation of the thermal conductivity with radiation damage. The selection of a heat sink material may also be affected if radiation embrittlement and swelling are significant. A major research program will be needed to examine radiation damage issues for the candidate materials and bonds. Thermophysical properties, mechanical properties, and radiation swelling and creep should be measured over a range of temperatures and fluences.

Reactor prototypical devices may also utilize refractory metals, such as vanadium, niobium, or tantalum, in high heat flux components. The data base

for these metals, at the expected operating conditions, is small, and thus a broad based program is needed to develop them for use in fusion reactors. Baseline property data is needed in several areas including mechanical properties and coolant corrosion. Fabrication development is required to show that these materials can be manufactured in the full size components. Finally, irradiation damage studies are needed to determine if these materials can survive the neutron environment.

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CHAPTER III

MATERIALS AND PROCESSES

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CHAPTER IV

THERMAL HYDRAULICS

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IV. TECHNICAL ASSESSMENT OF THERMAL-HYDRAULIC TECHNIQUES FOR HIGH HEAT FLUX FUSION APPLICATIONS

IV.1 Introduction

This chapter is a technical assessment of three possible thermal-hydraulic, high heat flux removal techniques which will result in adequate heat removal with minimum penalty from components in fusion reactors. The heat removal alternatives discussed are: (1) subcooled flow boiling (SFB) with water, (2) high velocity helium gas convection, and (3) liquid metal heat transfer in the presence of a magnetic field. The assessment emphasizes advantages, disadvantages, available data base, and applicability of fusion component requirements. The reader is referred to Ref. 1, which is an expanded version of this technical assessment, for additional details.

Because of the relatively high heat flux levels and long pulse duration in the next generation of fusion reactors, in-vessel components must be actively cooled. Table I of Ref. 1 contains a partial list of fusion devices (planned or under construction), specific fusion components to be cooled, and anticipated mean and peak steady-state heat-flux loads. All fusion components will be heated nonuniformly over their surface and the surface area of most components vary from 0.1 m^2 to 100.0 m^2 . Most components will be subjected to steady-state heat fluxes ranging from 0.005 kW/cm^2 up to 1.0 kW/cm^2 . Only the heat flux of the first wall for compact fusion reactors, beam dumps, and armor plates will exceed these levels.

IV.2 Three High Heat Flux Heat Removal Techniques

In order to provide a common basis for comparison, computations are made for each high heat flux (HHF) removal technique for a specific case study of cooling a uniformly heated coolant channel, 1.5 m long with an inside diameter of 1.0 cm, subjected to an external uniform heat flux (q) of 0.5 kW/cm^2 . Although in practice the heat flux will be nonuniform, a uniform flux is useful for comparative purposes. The selected heat flux level is relevant to many of the existing and future fusion machines (e.g., TFTR, TEXTOR, D-III, TFEX, etc.).

It should be noted that the geometry chosen may not be optimized for each heat removal technique. There are many coupled design constraints (e.g., material selection, thermal stresses, coolant leakage, safety, tritium control, and plasma edge heat flux profile) which must be incorporated before a specific design can be completed.

IV.2.1 Subcooled Flow Boiling (SFB) with Water

Subcooled flow boiling is one of the most efficient techniques of transferring high heat fluxes. As an increasing thermal heat flux is applied to the walls of a coolant channel containing a flowing subcooled fluid (i.e., the fluid's bulk temperature, T_f , is always below the saturation temperature, T_{sat}), the heat transfer capability increases with an accompanying rise in the wall temperature. The cooling fluid changes from a single-phase condition to a two-phase liquid/vapor regime up to a point in the SFB regime where a maximum or "critical" heat transfer rate is achieved. This maximum heat flux is referred to as the critical heat flux (CHF) and, if exceeded by the applied heat flux, will result in a significant drop in the rate of heat transfer to the flowing fluid, a precipitous rise in the heated wall temperature, and possible destruction of the coolant channel. Since the CHF forms a practical upper limit for the imposed heat flux in flow boiling, emphasis is placed on CHF, its characterization, prediction, enhancement, and available data base.

Substantial progress has been made in understanding some mechanisms and the multiple-parameter dependence of SFB CHF(1). However, in considering the requirements for fusion reactor components, the SFB data base is incomplete. CHF data exists over portions of the following range of parameters (see Table III of Ref. 1): (1) CHF ranging from 0.013-32.0 kW/cm², (2) coolant channel heated length to diameter ratio, L/D, ranging from 4.8 to 790., (3) mass velocity, G, ranging from 0.068-90 Mg/m²s, (4) water velocity ranging from 0.07-95.0 m/s, (5) pressure, p, ranging from 0.1-78.5 MPa, (6) flow channel hydraulic diameter, D, ranging from 0.4-19.0 mm, (7) binary mixtures, and (8) many different geometries and thermal loading configurations. However, there are large gaps in the above parameter ranges and the data accuracy can vary from about $\pm 5\%$ to as high as $\pm 40\%$. In addition to the above experimental data parameter ranges, a large number of flow-boiling CHF correlation parameter ranges can be found in Ref. 1.

Fundamental questions which must be addressed with respect to future fusion components include: What are the limits of L/D for which the SFB regime can be maintained over the entire length of the coolant channels; and, in addition to increasing the exit pressure, mass velocity, and subcooling, how else can these limits of L/D be increased?

The SFB case study was conducted for an inlet temperature (T_{in}) of 20°C and an exit pressure (p_{ex}) of 1.55 MPa. The results are shown in Fig. IV-1. Based on the coolant's exit bulk fluid temperature, ($T_{f_{exit}}$), subcooled nucleate boiling occurs within the channel when mass velocity (G) < 8.2 Mg/m²s. The onset of nucleate boiling (ONB) occurs at an exit wall temperature (T_w) of 207°C and fully developed subcooled boiling occurs at a wall temperature of 236°C. Therefore, partial nucleate boiling occurs between these two temperatures and begins at the exit for mass velocities between 8.2 and 8.8 Mg/m²s. Fully developed boiling (FDB) heat transfer is approximately five times the single-phase heat transfer; this will result in approximately a factor of two increase in the heat transfer coefficient computed using single-phase equations (i.e., $h_{FDB} = 118 \text{ kW/m}^2\text{K}$ for $G \leq 7.0 \text{ Mg/m}^2\text{s}$)(1). The bubble boundary layer is assumed to detach at the exit where the bulk fluid temperature curve intersects the curve for fluid temperature required for bubble detachment (T_{fd}).

Four CHF correlations by Gambill, Bowring, Katto, and Macbeth were used to determine the variation of CHF with G (see Fig. 3 of Ref. 1). Based on Bowring's correlation, the mass velocity must be greater than 6.6 Mg/m²s to avoid CHF. For a ratio of CHF/ q of 1.5,* the required value of G is about 12.5 Mg/m²s. Smaller values of G are possible, depending on the heat flux distribution, the heat transfer enhancements used, flow channel geometry, channel materials, and thermal stress limitations. The pumping power (P) per channel is about 0.16 kW and 0.6 kW for $G = 8.0 \text{ Mg/m}^2\text{s}$ and 12.0 Mg/m²s, respectively. In these cases, the pressure-drop (Δp_{SCB}) is 0.23 MPa and 0.62 MPa, respectively. If precise values of the wall superheat are needed in a design, the local pressure drop must be used to compute the local variation in the saturation temperature.

*The ratio, CHF/ q , is system-dependent and is coupled to thermal-hydraulic and other design constraints.

Fig. IV-1: Predictions of the Exit Conditions (Wall and Bulk Fluid Temperatures, Nusselt Number and Heat Transfer Coefficient), Pressure Drop (Single-Phase, Δp_{SP} , and Subcooled Boiling, Δp_{SCB} and Pumping Power, P , as a Function of Mass Velocity (G) for a 1.5 m Long, 1.0 cm Diameter (Inside) Tube Subjected to a Uniform Heat Flux of 0.5 kW/cm^2 and Cooled by Subcooled Water with an Exit Pressure of 1.55 MPa, and inlet subcooling of 180K. T_{WFDB} and T_{WONB} are Wall Temperatures at Which Fully Developed and the Onset of Nucleate Boiling Begins.

The advantages and disadvantages of SFB heat removal from HMF fusion components are:

Advantages:

- There is an established SFB literature and substantial data base⁽¹⁾.
- Relatively low pressure losses and pumping power result with SFB compared to high velocity helium gas and liquid metal cooling.
- SFB results in low operating surface temperature of fusion components.
- SFB results in high heat transfer.
- SFB heat transfer is increased by electric fields produced by the plasma.
- There are many SFB heat transfer enhancement options⁽¹⁾.
- There are no effects of magnetic field on the SFB pressure drop.

Disadvantages

- Possible occurrence of CHF. However, a significant SFB data base exists⁽¹⁾.
- Cleanup, in case of leakage, is a problem for any liquid (water or liquid metal). However, water clean-up is more straight-forward than, e.g., liquid metal or molten salts.
- Difficulty in tritium cleanup compared to helium and liquid metal coolants.
- SFB with water in HMF components may not be compatible with liquid metals used in the blanket or first wall regions. This raises a safety issue (see Chapter II) which must be addressed.

IV.2.2 High Velocity Helium Gas Convection

High velocity helium gas has been used extensively in the cooling of nuclear reactors. Since the first commercial gas-cooled reactor began operating in 1956 in the United Kingdom, 40 gas-cooled reactors operating in 8 countries have generated about a third of the world's nuclear power. However, most of the experience has been with annular geometries and additional work is needed for tubular-type channels. Helium is a chemically inert, single-phase gaseous fluid. The thermodynamic and transport properties of helium coolant are known for a wide range of temperature and pressure⁽¹⁾.

Helium is one of the best heat-transfer media among the gaseous materials. It has been used extensively as the coolant in reactors due to its low neutron absorption and low chemical activity. In commercial reactors, helium's heat transfer capability is improved by means of extended heat transfer surfaces (Mgnox reactors) or by roughening the surface of the fuel element.

The performance of a smooth and a roughened tube was evaluated for the helium case study. The conditions used include an inlet helium temperature of 40°C and an inlet pressure of 5.0 MPa. The results are given in Fig. IV-2. As indicated, at a mass flow rate of 150 gm/s and without roughening, the maximum tube wall temperature is 320°C and the coolant Mach number is 0.32. The corresponding pressure drop is 0.13 MPa. With 3-D roughening, the wall temperature can be reduced to 230°C, but the pressure drop becomes very high (> 3.4 MPa). Although the 3-D roughening can be very effective in increasing heat transfer, it is also very costly due to increases in pumping power. It should be noted that the increase in heat transfer results in an increase in the helium gas temperature. Since at constant pressure the Mach number is directly related to the product of the mass flow rate and the square root of the gas temperature, this increase in the temperature results in substantial differences in the values of Mach number for the smooth and roughened channels. The tube size, coolant pressure, and wall roughness should be optimized together for given fusion component requirements.

The advantages and disadvantages of high velocity helium gas heat removal from HMF fusion components are:

Advantages

- Helium is chemically inert.
- In case of leakage, vacuum chamber cleanup is easy.
- Tritium can be extracted from helium by standard, developed helium purification systems.
- Helium is non-reactive and can be used with any blanket coolant.
- Helium is nonmagnetic and not electrically conductive.

Disadvantages

- To reduce pressure drop, the coolant has to be operated at high pressures, typically from 5.0 to 10.0 MPa.

Fig. IV-2: Predictions of the Exit Conditions (Wall Temperature, Heat Transfer Coefficient and Total Pressure Drop), as a Function of Mass Flow Rate, for a 1.5 m Long, 1.0 cm Diameter (Inside) Tube Subjected to a Uniform Heat Flux of 0.5 kW/cm^2 and Cooled by High Velocity Helium Gas at 5. MPa (Solid Line for Smooth Wall and Dotted Line for Roughened Wall).

- The higher operating pressure may result in thicker channel walls and possible thermal stress problems.
- Pumping power of helium gas is much higher than that of SFB.

IV.2.3 Liquid Metal Heat Transfer (in the Presence of a Transverse Magnetic Field)

It is known, in general, that liquid metals are good heat transfer fluids. However, in a fusion reactor environment, a large magnetic field is always present and interacts with the liquid-metal (electrically conducting) flow. The thermal and hydraulic behavior of liquid metal flows in the presence of a magnetic field are quite different from that of a liquid-metal flow in the absence of a magnetic field. The relatively large amount of data and literature which deal with liquid metal heat transfer in the absence of a transverse magnetic field are not directly relevant for fusion components such as the first wall, the blanket, and the limiter/divertor. The discussion here will be focused on the effect of a transverse magnetic field on liquid-metal heat transfer and related subjects. It should also be pointed out that compatibility (corrosion) between liquid metal and structural material as well as the structural temperature limit in a radiation environment usually restricts the maximum operating temperature of the liquid metal to well below its boiling point. Thus, fusion components are not likely to operate in the boiling regime under normal conditions.

Three aspects of liquid metal flow in a transverse magnetic field will be considered briefly before the case study is summarized. First, the pressure-drop of a liquid metal in the presence of a magnetic field is primarily due to the Lorentz force and results in a pressure drop more than an order of magnitude larger than that in the absence of the transverse magnetic field. Hence, large inlet pressures (~ 75 MPa @5 m/s) are required compared to helium gas convection or SFB. Therefore, as a result of the large pressure-drop, large pumping power is required. Finally, there are competing heat transfer effects. For example, at high Hartmann numbers, the heat transfer is increased, as is manifested by a thinner boundary layer. However, a magnetic field suppresses turbulence and thus reduces the heat transfer. In addition, nonuniformities in the magnetic flux density, coolant channel wall thickness or channel shape would induce similar nonuniformities in the velocity profile. Other effects which complicate the heat transfer are summarized in Ref. 1.

Fig. IV-3: Predictions of the Pumping Power and the Exit Wall Temperature, as a Function of Velocity, for a 1.5 m Long, 1.0 cm Diameter (Inside) Tube (Stainless Steel with 2.0 mm Wall Thickness) Subjected to a Uniform Heat Flux of 0.5 kW/cm^2 and Cooled by Liquid Lithium, in the Presence of a 5.0 Tesla Transverse Magnetic Field.

Conditions used for the liquid metal case study are: the transverse magnetic flux density is 5.0 Tesla, the stainless steel channel wall thickness is 2.0 mm, the Hartmann number is approximately 2300, and the liquid lithium inlet temperature is 230°C .

Fig. IV-3 indicates that at 0.5 kW/cm^2 and a velocity of 5.0 m/s, both the pumping power (30 kW per coolant channel) and the wall temperature (590°C) are very large compared to the results for SFB. Details of this analysis can be found in the appendix of Ref. 1.

The advantages and disadvantages of liquid metal cooling of HHF fusion components in the presence of a transverse magnetic field are:

Advantages

- Liquid metals have a high thermal conductivity.
- There is considerable experience in handling liquid metal systems.
- Liquid lithium or lithium-lead can be used both as a breeder and coolant in the blanket.

Disadvantages

- Some liquid metals are corrosive and may not be compatible with certain structural materials.
- The pressure drop and hence the pumping power are very large compared to SFB. If feasible, the designer should arrange the flow to be parallel to the magnetic field to minimize the pressure drop.
- Liquid metals are more chemically reactive than either helium or water.

IV.3 Comparison of Three High Heat Flux Heat Removal Techniques

The three high heat flux removal techniques are compared by examining the heat transfer coefficients, maximum wall temperature (at the exit of channel), pumping power, and pressure drop. The conditions for which the comparisons are made are applicable to many of the fusion components such as limiters and divertors (in e.g. TFTR and TFCX). For these comparisons, it is assumed that the coolant channel is subjected to a uniform heat flux of 0.5 kW/cm^2 (5.0 MW/m^2), the channel is 1.5 m long and has an inside diameter of 1.0 cm. In the case of liquid metals the fluid is liquid lithium, the transverse magnetic field density is assumed to be 5.0 Tesla, and the coolant channel is assumed to be stainless steel with a wall thickness of 2.0 mm.

Since the design for cooling a fusion component is coupled with many other design constraints (e.g., materials, and thermal stresses), such a design is machine- and component-dependent. In addition, the thermal-hydraulic performance of each heat removal technique has not been optimized. Therefore, the results in Figs. IV-1 through IV-3 are not optimized and should not be used for the design of specific fusion components.

One limitation placed on the design of a fusion component is the maximum allowable wall temperature. For an imposed heat flux of 0.5 kW/cm^2 , the

wall temperature variations are: (1) 147.°C to 236.°C for SFB, (2) 230°C (roughened tube) to 320°C (smooth tube) for high velocity helium gas cooling, and (3) 444.°C to 590.°C for liquid lithium. At these channel wall temperatures, additional comparisons are given in Table IV-1. The pumping power requirements increase by more than an order of magnitude for helium gas when compared to SFB; that for liquid metals is more than four times that for helium gas. When the channels are roughened the pumping power for helium gas cooling can be almost two orders of magnitude above that for SFB. The roughened channel heat transfer coefficient for helium gas is about a factor of three below that for fully developed SFB. Because of the high thermal conductivity of liquid metals, the heat transfer coefficient is between 58 and 90% higher than that for helium gas cooling and between 42 and 30% below that for SFB.

IV.4 Conclusions and Recommendations

This assessment deals primarily with the heat transfer aspects of fusion component heat removal using three different techniques. In selecting a coolant for a specific fusion component, one must include all the design constraints. This is crucial since some of the advantages and disadvantages of each method can only be brought out by considering all the design constraints for specific components.

Based on the present technological limits of heat removal, the estimated capability of each alternative examined is shown in the last line of Table IV-1, which shows that SFB has the greatest heat removal capability followed by helium gas and liquid metals. In addition, for the same imposed heat flux, SFB is a more efficient heat removal technique because:

- (1) Higher heat transfer coefficients result,
- (2) Lower pumping power, and hence cost, is required, and
- (3) Lower coolant channel wall temperatures result.

Since SFB is the most efficient heat removal technique, the critical issues and engineering research and development (R&D) needs to develop this cooling method for HNF fusion components are summarized next. The prioritized critical issues for HNF SFB heat removal from fusion components are:

TABLE IV-1: Comparison of Three High Heat Flux Heat Removal Alternatives*

	Subcooled Flow Boiling (SFB)	High Velocity Helium Gas Cooling		Liquid Metal† Cooling (Liquid Lithium)
		Smooth Channel	Roughened Channel	
Investigated interval for wall temperature:				
(i) Lower exit wall temperature (LEWT) (°C)	147	250***	230***	435
Flow condition	16 Mg/m ² s	300 gm/s	110 gm/s	10.0 m/s
(ii) Upper exit wall temperature (UEWT)(°C)	236	365	365	590
Flow condition	7.3 Mg/m ² s	140 gm/s	65 gm/s	5.0 m/s
Pumping power (kW) per coolant channel at:				
(i) LEWT	1.4	45.0	20.0	118.
(ii) UEWT	0.12	10.0	7.5	30
Heat transfer coefficient (h) at LEWT (kW/m ² K)	118.**	20.	43.	68.@5m/s 82.@10m/s

Present technological limits on imposed heat flux for fusion components (kW/cm ²)	0.5 (L/D < 200) 8.0 (L/D < 50)	1.0#†	2.0#	1.0
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*Caution: These results should not be used for design purposes since there are other coupled constraints unique to specific fusion components which have not been included. The comparisons are made for the following conditions: 0.5kW/cm² Uniform Heating; Circular Coolant Channel; L=1.5m, D=1.0cm.

**If enhancement techniques are included h will be higher. The SFB predictions are applicable for an inlet subcooling of 180K and exit pressure of 1.55 MPa.

***Limited by the coolant maximum Mach number < 0.5.

†The transverse magnetic field was assumed to be 5.0 Tesla; the coolant channel was assumed to be stainless steel with a 2.0 mm wall thickness.

#Limited by maximum external tube wall temperature < 550°C for Prime Candidate Alloy (PCA: 316 stainless steel doped with titanium) with a wall thickness of 1.0 mm.

#*For the high velocity helium gas cooling technological limits of 1.0 and 2.0 kW/cm², the pumping power is very high, i.e., approximately 190 and 275 kW per channel, respectively.

- (1.) Evaluation of SFB CHF between 0.1 and 2.0 kW/cm² for large surface area (0.1 to 100.m²) fusion components with large L/D ratio (50 to 600) coolant channels. The CHF data base in this range is sparse or nonexistent and is essential to the design of near-term fusion components.
- (2.) Establishing the SFB thermal-hydraulic stability criteria of multiple, interconnected, and subchannel coolant conduits is essential to near-term component design. However, this becomes less important if the slope of the pressure-drop/flow-rate curve of each coolant channel is required to be positive or greater than the pump's performance curve.
- (3.) Increase the understanding of the physical mechanisms leading to SFB CHF in order to develop engineering thermal-hydraulic models which can be used to characterize the data base.

The prioritized engineering R&D needs for RHF SFB heat removal from fusion components are:

- (1.) Develop unconditionally stable SFB CHF and pressure loss data for single and multiple coolant channels with heat fluxes between 0.1 and 2.0 kW/cm² and determine the quantitative effects of surfactant additives, binary mixtures, swirl flow, and helical subchannels relative to stable axial-flow data.
- (2.) Perform linear (and later nonlinear) thermal-hydraulic stability analyses to examine, e.g., the Ledinegg, flow pattern transition, parallel channel, and condensation-induced instabilities. The fusion component flow channel design will not be complete without such an analysis.
- (3.) The mechanistic understanding of conditions leading to SFB CHF must be increased by:
 - (a.) Developing engineering models, for SFB CHF characterization, based on first principles which relate the applied heat flux to both the bulk SFB parameters (velocity, heated length, diameter, inlet temperature, exit pressure, and thermophysical properties) and local SFB parameters (e.g., void fraction distribution; minimum, mean and maximum bubble size; bubble density; and boundary layer thicknesses).

- (b.) Performing flow visualizations to observe both hydrodynamic and thermal characteristics which will lead to an overall understanding of SFB CHF and provide guidance in engineering model development.

The prioritized engineering R&D needs for high velocity helium gas convection for HHF removal from fusion components are:

- (1) Extend the existing heat transfer and friction data from annular geometries to tubular-type geometries.
- (2) Determine the appropriate manufacturing techniques for two- and three-dimensional roughened surfaces.

Finally, the prioritized R&D needs for liquid metal heat transfer in the presence of a transverse magnetic field for HHF heat removal for fusion components are:

- (1) Determine the effect of high Hartmann number, nonuniform magnetic flux density, non-circular cross-section and nonuniform coolant channel wall thickness on the velocity distribution and the heat transfer, and
- (2) Determine the significance of natural convection (in large conduits, e.g., in blanket and first wall applications), and the thermoelectric (or Seebeck) effect.

REFERENCES FOR CHAPTER IV

1. Boyd, R. D., Wong, C. P. C., Cha, Y. S., "Technical Assessment of Thermal-Hydraulics for High Heat Flux Fusion Applications," SAND84-0159, Sandia National Laboratories, Albuquerque, NM, 1984.

CHAPTER V

THERMOMECHANICAL RESPONSE

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V. THERMOMECHANICAL RESPONSE

V.1 Thermal Response

High heat flux components must be adequately cooled to properly perform their functions. The method of cooling, whether passive or active, depends strongly on the applied heat flux, pulse length, thermal properties, and the maximum allowable temperature.

V.1.1 Passive Cooling

Passive cooling is limited by the heat capacity, melting point, and surface emissivity of the material. This limit can be determined by estimating the heat flux required to melt the surface of a "thermally thick" object (1):

$$q_{\text{melt}} = \frac{1}{2} \sqrt{\frac{\rho C_p k}{t_p}} (T_m - T_o) \quad \text{W/cm}^2 \quad (1)$$

q_{melt}	= heat flux	W/cm ²
	= density	g/cm ³
C_p	= specific heat	J/g-K
k	= thermal conductivity	W/cm-K
t_p	= pulse length	s
T_m	= melting point	K
T_o	= initial temperature	K

This equation is plotted in Fig. V-1 for nine selected materials. For heat fluxes in the range of 10 to 100 W/cm² (first walls), passive cooling is sufficient for both existing and near-term devices but not for long-term machines with pulse lengths greater than 100 s. For heat fluxes in the range of 100 to 1000 W/cm² (limiters, divertor collector plates, halo scrapers, etc.), passive cooling becomes marginal for near-term machines and will not work for long-term machines. For heat fluxes from 1000 to 10000 W/cm² (neutral beam dumps), active cooling is required for any fusion device with pulse lengths greater than 1 to 5 s. Finally, for heat fluxes greater than 10000 W/cm² and pulse lengths on the order of 10 ms (plasma disruptions), active cooling will not be able to prevent surface melting or vaporization

SURFACE HEAT FLUX TO MELT SURFACE

Figure V-1. Combinations of surface heat flux and pulse length required to melt the surface of an uncooled, "thermally thick" material.

since the thermal diffusion length is typically much shorter than the cooling channels' wall thickness.

The best materials for absorbing a short pulse of heat without melting are those that have a high value of $(T_m - T_o) \sqrt{\rho C_p k}$ [see Eq. (1)]. Figure V-2 shows that Mo, WC, and W have the highest heat absorption capacity among the high-Z materials ($Z > 24$). VC, TiC, and TiN rank the highest among medium-Z materials ($16 < Z < 24$), while graphite, BN, and B_4C rank the best among low-Z materials ($Z < 16$). It is interesting to note that although 316 stainless steel has the lowest heat absorption capacity of all the materials listed in Fig. V-2 (except for Be_2C and Si_3N_4), it is currently used in existing fusion devices as a limiter material due to its high ductility and excellent fracture toughness.

V.1.2 Active Cooling

Since active cooling will be needed for most high heat flux components in future devices, this implies the use of a relatively thin wall separating the plasma side material from a high velocity coolant. Steady-state conditions will exist and temperature profiles through the thickness will be approximately linear. The maximum steady-state heat flux needed to melt the surface is given by:

$$q_{\text{melt}} = \frac{k (T_m - T_b)}{h} \quad \text{W/cm}^2 \quad (2)$$

q_{melt}	= heat flux	W/cm ²
k	= thermal conductivity	W/cm-K
T_m	= melting point	K
T_b	= backside temperature	K
h	= wall thickness	cm

Figure V-3 shows that W, Cu, and Mo have the best heat conduction capacity among the high-Z materials. For medium-Z material, TiC and VC are good, while among the low-Z materials, SiC, BeO, graphite, and Be rank the best. In practice, however, the maximum allowable temperatures will be significantly lower than the melting point due to radiation damage, loss of mechanical properties, corrosion, vaporization losses, etc.

HEAT CAPACITY FIGURE-OF-MERIT $(T_m - T_0) \sqrt{\rho C_p k}$ KJ/cm²/√sec

Figure V-2. Heat capacity figure-of-merit. Larger numbers imply that more heat can be absorbed in a given amount of time without melting.

Figure V-3. Maximum heat flux needed to melt the surface of a 1 cm thick plate cooled on its backside to 100°C during steady-state conditions.

V.1.3 Thermal Response to Plasma Disruptions

In existing tokamaks, plasma disruptions can occur frequently, depositing energy densities from 100 to 1000 J/cm² over time periods from 10 μ s to 10 ms. The best materials for resisting plasma disruptions have high melting points, low vapor pressures, and high values of ρ , C_p , and k . Figures V-4 and V-5 show theoretical predictions of melting and vaporization. Figure V-4 shows that W, TiC, and BeO have smaller melt layers than Be, Ta, 316 SS, or Cu. Figure V-5 shows that W, Ta, and Cu are more resistant to vaporization than BeO, Be, and 316 SS. Ti and SiC fall in the middle of these two extremes.

Due to the lack of experimental data, lifetime predictions have relied heavily on computer predictions. These codes have progressed significantly in recent years but need improved models for vapor shielding. These codes need to be compared with electron and ion beam simulations. Two-dimensional effects of the e-beam spot size and power distribution need to be accounted for. Also, the question of melt layer stability in a pulsed electromagnetic field is a serious issue that needs to be addressed both theoretically and experimentally.

V.1.4 Irradiation Effects

Lattice defects produced by neutron irradiation will decrease the thermal conductivity due to the increase in resistance to phonon transport. The effect is strongest for non-metals. For example, the thermal conductivity of SiC is known to be reduced by a factor of four after exposure to a neutron fluence of 10^{22} n/cm² at 800 K⁽²⁾. Graphite, BeO, and TiC would experience similar behavior. This implies that thinner armor tiles would have to be used in order to minimize thermal stresses then the irradiated properties are used. In the case of SiC, this results in a factor of four decrease in the lifetime when surface erosion is the life-limiting factor.

V.1.5 Thermal Analysis Computer Codes

There are currently available a wide variety of sophisticated finite element and finite difference computer codes that are suitable for thermal analysis of high heat flux components. Many capabilities exist: transient,

Figure V-4. Amount of material melted during a 20 ms plasma disruption.

Figure V-5. Amount of material vaporized during a 20 ms plasma disruption.

3-D solutions; thermal radiation; non-linear, anisotropic thermal properties; gap conductance models; phase change capability; coupled heat transfer/stress analysis solutions; etc.. The capabilities of these codes are adequate for conceptual design studies and analysis of experiments.

The accuracy of these predictions will be improved by a better knowledge of the boundary conditions, in particular, the applied heat flux on the plasma side and the heat transfer coefficient on the coolant side. There is a need for computer programs that can calculate the detailed heat flux profile on an arbitrarily shaped surface as a function of (1) the plasma scrape-off distance; (2) power sharing and connection lengths between limiters, antennas, etc.; and (3) the motion of the plasma during disruptions. More accurate correlations are needed for heat transfer with high velocity, sub-cooled nucleate boiling water.

V.2 Structural Response

Stresses and deformations in high heat flux components are caused by five types of loads: (1) gravity, (2) coolant pressure, (3) electromagnetic forces, (4) thermal expansion, and (5) irradiation creep and swelling. Dead weight loads due to gravity are typically small and easily managed. Likewise, for actively cooled designs with coolant pressures on the order of 10 MPa, hoop stresses are not large because the ratio of channel radius to wall thickness is typically less than 10. Electromagnetic forces caused by eddy currents induced by plasma disruptions can cause large transient stresses. This is discussed in the Chapter VI, Electromagnetic Response. Thermal stresses and radiation effects are discussed in this chapter.

V.2.1 Thermal Stresses

Thermal stresses are caused when temperature changes induce thermal expansion in a constrained material. The degree of constraint strongly affects the stress level and depends on both the method of thermomechanical attachment and on the non-linearity of temperature gradients within the structure. As an example, graphite limiter tiles used in existing fusion devices are mechanically bolted to support plates because high thermal conductance to the support plate is not required to actively remove heat during the short pulses.

Thermal stresses are consequently lower because the tile is relatively free to expand and rotate. However, for longer pulse lengths, the tiles need to be rigidly bonded to an actively cooled heat sink. This increases thermal stresses because of the stiffer mechanical constraint. Thermal stresses caused by plasma disruptions will be very large at the heated surfaces because the non-linear temperature gradient induces a high degree of self-constraint within the tile, independent of the type of mechanical attachment.

Thermal stresses are proportional to $E\alpha\Delta T (1 - \nu)$, where E = Young's Modulus (MPa), α = thermal expansion coefficients (K^{-1}), ν = Poisson's Ratio, and ΔT = temperature difference (K). If the structure is passively cooled, then $\Delta T \sim (\rho C_p K)^{-1/2}$. Therefore, an appropriate figure-of-merit for transient thermal stresses is given by the dimensionless ratio of yield strength to thermal stress:

$$\begin{array}{l} \text{Transient Thermal Stress} \\ \text{Figure-of-Merit} \end{array} = \frac{\sigma_y (1 - \nu) \sqrt{\rho C_p K}}{E\alpha} \quad (3)$$

Figure V-6 shows that graphite, Cu, and Mo alloys can sustain the largest heat pulses before yielding.

For actively cooled components, steady-state conditions are reached and the temperature differences are simply proportional to the thermal conductivity, $\Delta T \sim k$. In this case, the appropriate dimensionless figure-of-merit is given by:

$$\begin{array}{l} \text{Steady-State Thermal Stress} \\ \text{Figure-of-Merit} \end{array} = \frac{\sigma_y (1 - \nu) k}{E\alpha} \quad (4)$$

Figure V-7 shows that graphite, Cu, and Mo alloys also have the lowest dimensionless thermal stresses during steady-state conditions.

Practical experience confirms these rankings. Passively cooled limiters on TFTR, D-III, ISX-B, etc. that are made of TiC-coated graphite have experienced no structural failure after many thousands of cycles. Actively cooled neutral beam dumps are currently made of either bare copper or molybdenum alloys and have demonstrated successful performance at heat flux levels up to 8000 W/cm^2 steady-state⁽³⁾.

Figure V-6. Thermal stress figure-of-merit for transient heating. Higher values of the figure-of-merit imply that more heat can be absorbed during a given amount of time without yielding the material.

Figure V-7. Thermal stress figure-of-merit for steady-state heating. Larger values imply that greater heat fluxes can be removed steady-state without yielding the material.

There is little experience in the fusion community with actively cooled duplex or triplex components constructed of thick (~ 1 cm) armor tiles bonded to a metallic heat sink, as shown in Fig. V-8. The armor, cover plate, and base plate materials can be chosen independently in order to optimize performance. Since the heat sink is protected from disruptions and surface erosion, then thin-walled cover plates (1 to 5 mm) can be used to minimize thermal stresses. Likewise, the base plate will have low thermal stresses if the cooling channels are placed near the plasma side. The armor material will have the largest thermal stresses since the thickness required for erosion protection results in a large temperature drop across the tile. Also, the interface between the armor and heat sink will be subject to large shear stresses due to differential thermal expansion. Fortunately, cracking of armor is less serious than a coolant leak or bond detachment.

V.2.2 Thermal Stresses from Plasma Disruptions

Plasma disruptions will induce large compressive thermal stresses at the surface. At these high temperatures, the material can easily plastically deform and creep, resulting in residual tensile stresses at the surface after cooldown. These stresses may be sufficiently large to cause reverse yielding and surface microcracking if the material's ductility is low. This mechanism of cracking has been observed on TiC-coated graphite limiters in ISX-B, D-III, TFTR, and on targets of 316 SS and Be that have been tested in ESURF and the Sandia electron-beam test facility. In principle, these microcracks can grow and coalesce into a large surface flaw that may continue to grow due to normal operating stresses. This is a concern primarily for long-term machines.

V.2.3 Irradiation Effects on Structural Response

In future fusion devices such as ETR, INTOR, DEMO, and commercial reactors, neutron irradiation damage will significantly affect the distribution of stresses in high heat flux components. For example, the decrease in thermal conductivity of non-metallic materials will increase stresses over time. Fortunately, irradiation creep will prevent the unlimited buildup of these stresses. Predictions by inelastic computer codes have shown that swelling stresses can reach the same magnitude as thermal stresses⁽⁴⁾. Unlike thermal stresses which fatigue cycle with each pulse, swelling stresses are permanently locked into the structure.

Figure V-8. Triplex 3-layer construction of a high heat flux component.

Differential swelling between different layers of a duplex structure will produce a complex stress state and may eventually cause debonding at the interface. In addition, the permanent deformations caused by swelling and creep may prevent easy removal of components for maintenance. The accuracy of the predictions is, of course, limited by the sparse data base on creep and swelling of copper, molybdenum, tantalum, and tungsten alloys as well as the non-metallic plasma-side materials.

V.2.4 Stress Analysis Computer Codes

There are currently available a wide variety of sophisticated finite element computer codes that can accurately predict the stresses and displacements of high heat flux components. The capabilities of these codes include: 3-D, transient solutions; large deformations; elastic-plastic, non-linear, viscoelastic, anisotropic, temperature dependent materials models; sliding boundaries; creep and swelling models; coupled heat transfer/stress analysis solutions; etc. Efficient interactive pre- and post-processor programs are also available. Their accuracy is primarily limited by constitutive models for high temperature material behavior with combined cyclic loading.

V.3 Failure Analysis

The successful operation of high heat flux components depends on a good understanding of the potential modes of failure. This requires that predictions of the thermomechanical response be integrated with models for damage accumulation. For existing and near-term fusion devices, the ASME Boiler and Pressure Vessel Code used by the fission industry has provided an excellent set of standards and design criteria for protecting against structural failures. However, future machines will subject these components to more fatigue cycles, higher temperatures, harder disruptions, and neutron radiation damage. Accurate failure analysis will become more difficult because of (1) complex synergistic effects, (2) a sparse data base, and (3) incomplete damage models. This section discusses the critical issues associated with two areas: time-independent and time-dependent failure modes.

V.3.1 Time-Independent Failure Modes

Time-independent structural failures are caused by the single application of an excessive load. These are listed in Table V-1. In general, the models for predicting these types of failures are not well understood. For a one-time load, synergistic effects are not important, except for situations where one failure leads to a second, more severe failure. For example, the debonding of an armor tile from the heat sink would cause excessive temperatures, a loss in strength, and finally fracture or melting.

Table V-1. Time-Independent Failure Modes

1. Tensile plastic instability (ductile rupture)2. Compressive instability (buckling)3. Brittle fracture4. Burnout (critical heat flux)5. Interface debonding6. Excessive deformation

It is important to generate materials data on (1) the short-term mechanical properties and stress-strain data at elevated temperatures and (2) interface bond strengths for braze bonds, diffusion bonds, and plasma sprayed coatings at elevated temperatures. Existing test facilities are considered adequate for this purpose.

V.3.2 Time-Dependent Failure Modes

The progressive accumulation of damage over an extended period of time can eventually cause component failure. The damage can either be localized to areas of high stress concentration or can extend globally over the entire structure. Table V-2 lists the most important failure modes.

Global time-dependent failures are caused by excessive deformation due to creep, swelling, cyclic plasticity, or loss of material. Creep buckling is not considered to be a problem. Ratchetting due to cyclic plasticity may cause problems at high heat flux levels and needs to be studied in more detail. Excessive deformations due to thermal or irradiation creep are not a

Table V-2. Time-Dependent Failure Modes

Global

1. Creep buckling
2. Ratchetting (incremental growth)
3. Excessive creep deformation
4. Excessive swelling deformation
5. Wall thinning (corrosion/erosion)

Local

6. Creep instability (necking, thinning)
7. Creep rupture
8. Fast fracture
9. Crack leak-through
10. Interface debonding

concern because the design allowable creep strain limits are typically small so as to prevent localized creep rupture. Swelling is a more serious problem since the removal of components for repair may be prevented by excessive deformations. Wall thinning caused by erosion of plasma-side materials or corrosion/erosion of internal coolant channels is also a serious concern. The PMI Task Group is addressing the erosion problem. However, the use of high velocity water cooling may cause significant internal erosion of copper heat sinks.

Localized time-dependent failures occur in regions of high stress concentration and/or degraded mechanical properties. Table V-3 lists ten different damage mechanisms that may contribute to localized failures. Existing models for fatigue crack initiation and growth are well developed, especially when linear elastic fracture mechanics assumptions can be used. These models also include reductions in the fracture toughness due to embrittlement processes. However, improvements are needed for predicting crack propagation through highly non-linear stress fields and interfaces between bonded layers.

Table V-3. Damage Mechanisms

1. Cyclic plastic deformation/crack initiation
2. Creep deformation
3. Creep cavitation
4. Fatigue crack growth
5. Creep crack growth
6. Hydrogen embrittlement
7. Helium embrittlement
8. Neutron radiation damage
9. Impurity migration to bond interfaces
10. Corrosion

Apart from radiation damage concerns, the maximum allowable operating temperatures for high heat flux components will be determined by creep damage processes. The nucleation and growth of creep cavities at grain boundaries and near crack tips caused localized creep rupture and creep crack growth in regions of high stress and temperature. The models for creep damage are not well developed and need improvement. Due to the pulsed nature of tokamaks, creep-fatigue interactions are expected to accelerate the accumulation of damage, but models are also lacking in this area. Another damage mechanism that is difficult to model is the migration of impurities, gas atoms, or transmutation products to interfaces between bonded layers as the result of stress and temperature gradients. Finally, localized corrosion is a concern for some alloys and coolants.

This section has discussed 16 types of potential failure modes for high heat flux components. For nearly all of these, the data base is inadequate for making accurate lifetime assessments. The appropriate use of safety factors (on stress levels, etc.) can relax the need for a large data base, but this usually results in an overly conservative design with degraded performance. Nevertheless, there are a number of critical issues with regard to failure analysis and these are summarized in Table V-4.

Table V-4. Critical Issues for Failure Analysis

FAILURE MODE	TIME FRAME			
	Present Day	TFCX	INTOR	Commercial
1. Ductile Rupture	3	3	3	3
2. Buckling	3	3	3	3
3. Brittle Fracture	3	2	2	2
4. Burnout	4	1	1	2
5. Interface Debonding	4	1	1	2
6. Excessive Deformation	4	3	3	3
7. Creep Buckling	4	4	4	4
8. Ratchetting	4	2	2	2
9. Creep Deformation	4	4	2	2
10. Swelling Deformation	4	4	2	1
11. Wall Thinning	4	2	1	1
12. Creep Instability	4	2	2	2
13. Creep Rupture	4	2	2	1
14. Fast Fracture	4	2	1	1
15. Leak-Through Crack	4	1	1	1
16. Interface Debonding	3	1	1	1
	Passive Cooling	Active Cooling	Neutron Damage	Long Lifetime

NOTES:

- 1 = critical issue
- 2 = needed for improved performance
- 3 = well understood
- 4 = not important

V.4 Design Criteria

A common set of standards and design criteria is needed for structural components in fusion reactors. However, for present-day machines, the ASME Boiler and Pressure Vessel Code, Sections III and VIII for Class I nuclear components provides adequate guidelines, data, and design criteria. In the next three to eight years, for devices such as TFCX, the ASME Code Case N47 for Elevated Temperature Service would also be appropriate to use. However, the safety factors and stress limits are too conservative to be applied to protective tiles or coatings because they do not constitute a pressure boundary. These limits would not be appropriate to use for the highly localized stresses caused by plasma disruptions.

For INTOR and commercial fusion reactors, a new set of standards should be developed that specifically include the effects of irradiation creep, swelling, embrittlement, disruptions, etc. This new code also should require a fracture mechanics analysis to be performed, including both fatigue and creep crack growth effects. Since it will take many years to rewrite the ASME B&PV Codes, it is recommended that the "Fast Breeder Reactor Core Component Design Criteria. Draft Standards RDT F9-7,8,9" be used in the interim. Also, simplified methods of inelastic stress analysis should be developed. These computer codes can be easily used in conceptual studies to evaluate the impact on lifetime of newly developed correlations for materials behavior, in particular, irradiation creep and swelling, erosion, and crack growth.

V.5 Thermomechanical Testing

The successful operation of high heat flux components in future devices will depend strongly on a dedicated program of thermomechanical testing of prototypes in simulation facilities. Tests on simple geometries allow useful comparisons with model predictions, while prototype tests simulate synergistic effects that are difficult to predict. Existing MFE facilities include the ESURF/ASURF electron beam facilities at Westinghouse and the electron beam test facility (EBTF) at Sandia-Albuquerque (SNLA). Ion beam test facilities include beam dumps for the Neutral Beam Engineering Test Facility (NBETF) at Lawrence Livermore National Laboratory, the Medium Energy Test Facility (METF) at Oak Ridge National Laboratory, the Intense Neutron Source (INS) at SNLA, and the Plasma Materials Test Facility (PMTF) at SNLA, currently under construction.

Table V-5 summarizes the results of tests of actively cooled components completed within the last five years. With the exception of the beryllium clad stainless steel duplex structures, all of the others consisted of a water-cooled heat sink made of a single material, either copper, molybdenum, or stainless steel. For heat flux levels under 200 W/cm^2 , none of the stainless steel heat sinks experienced any structural failures. From 300 to 1000 W/cm^2 , there was no testing done. The molybdenum tubes were tested at 1000 to 2000 W/cm^2 without failure up to 600 cycles. From 2000 to 7500 W/cm^2 , only copper heat sinks were tested. Wall thicknesses ranged from 1 to 4 mm. The METF neutral beam dump has survived 26,000 cycles to date at 5500 W/cm^2 and the INS target ran for a total exposure of 140 hours at 4000 to 6000 W/cm^2 before it failed due to a pinhole water leak. These tests are very encouraging and have demonstrated that actively cooled heat sinks can be successfully operated at high heat flux levels ($\sim 1000 \text{ W/cm}^2$) for a significant number of cycles ($\sim 10,000$ cycles).

The tests of 2-mm thick beryllium tiles brazed to a 316 SS heat sink were not successful. The large Be tile cracked after 35 cycles at only 50 W/cm^2 due to shear stresses, while the smaller tiles ($< 2''$ width) did not crack. The testing of duplex and triplex structures is a high priority task since it appears that the performance of duplex structures may not be as good as the single material heat sinks.

V.6 R&D Needs

This section discusses the R&D activities that are needed for improving the thermomechanical performance of high heat flux components.

V.6.1 Existing Machines

1. 3-D Heat Flux Profile Code. A computer program is needed that can calculate the detailed heat flux profile on an arbitrarily shaped surface as a function of (1) the plasma scrape-off distance, (2) power sharing and connection lengths between limiters, antennas, etc., and (3) the motion of the plasma during disruptions. This will improve the analysis of thermocouple and infrared thermography data, and simplify the task of generating heat flux boundary conditions for 3-D finite element thermal analysis.

Table V-5. Summary of Testing of Actively Cooled Components

FACILITY	INS	EBTF	EBTF	ESURF	ECURF
TARGET	19-channel cooling panel	19-channel cooling panel	4-tube array, internal fins	5-channel cooling panel	single tube
MATERIAL	OFHC copper	OFHC copper	OFHC copper	316 S.S.	316 S.S.
HEAT FLUX (W/cm^2)	4000 - 6000	4000	7500	300	300
WALL THICKNESS (mm)	1	4	1	4	4
CHANNEL I.D. (mm)	2	2	10		20
PULSE LENGTH (s)	steady-state	10	10	60	50 - 60
COOLANT	water	water	water	water	water
PRESSURE (psi)	300	300	300	100 - 1000	100 - 1000
VELOCITY (m/s)	33	35	10		0.8 - 4
BOILING	yes	yes	yes	yes	yes
TOTAL CYCLES	200	10,000	500	200	200
TOTAL TIME (hrs)	140	28	15	3	3
FAILURE MODE	pinhole leak, internal erosion	no leak, small surface cracks	pinhole leak, burnout	no failure, plastic deformation	no failures

Table V-5. Summary of Testing of Actively Cooled Components (continued)

FACILITY	ESURF	ESURF	ESURF	ESURF
TARGET	single tube	tube, with dis- ruptio melting	single tube	single tube
MATERIAL	316 S.S.	316 S.S.	molybdenum	molybdenum
HEAT FLUX (W/cm^2)	100	100	1100	2100
WALL THICKNESS (mm)	8	8	3	3
CHANNEL I.D. (mm)	20	20	6	6
PULSE LENGTH (s)	60	60	2	10
COOLANT	water	water	water	water
PRESSURE (psi)	100 - 1000	100 - 1000	300	300
VELOCITY (m/s)	0.8 - 4	0.8 - 4	16 - 24	16 - 24
BOILING	no	no	no	no
TOTAL CYCLES	500	100	600	30
TOTAL TIME (hrs)	8.3	1.7	0.3	0.1
FAILURE MODE	no failure	no failure	no failure	no failure

Table V-5. Summary of Testing of Actively Cooled Components (continued)

FACILITY	ASURF	ASURF	ASURF	METF	NBETF
TARGET	MFTF-B end cell cooling panel	MFTF-B end cell cooling panel	Duplex cooling panel	swirl tube neu- tral beam dump	heat absorb- tion panel
MATERIAL	copper	304 S.S.	Be brazed to 316 S.S.	OPHC copper	Amzirc copper
HEAT FLUX (W/cm ²)	150	150	50	5:00	1100 - 1600
WALL THICKNESS (mm)	1.3	1.3	2 mm Be 3 mm 316 S.S.	1.6	2
CHANNEL I.D. (mm)	19 x 1	13 x 2	19 x 13	6.3	2 x 1
PULSE LENGTH (s)	15	25	60	0.5 - 30	30
COOLANT	water	water	water	water	water
PRESSURE (psi)	125	125	1000	150	200
VELOCITY (m/s)			0.2 x 1.0	12	17
BOILING	no	no	no	yes	yes
TOTAL CYCLES	3400	10,000	35	26,000	~ 10,000
TOTAL TIME (hrs)	14	69	0.6	22	80
FAILURE MODE	no failures	no failures	fracture of large Be tile, no failure of small Be tiles	no failures	no failures

V.6.2 TFCX (3-8 Years)

2. Plasma Disruptions. Improved models for vapor shielding need to be included in the computer codes used for predicting melting and vaporization effects caused by plasma disruptions. These predictions need to be compared with electron beam experimental measurements to improve their accuracy. The location, duration, and intensity of the thermal quench portion of the disruption needs to be defined more precisely.

3. Boiling Heat Transfer. Improved correlations are needed for predicting the heat transfer coefficient during forced convection subcooled nucleate boiling. The data should include multiple channel effects, swirl flow, and internal fin effects for heat fluxes in the range of 100 to 2000 W/cm².

4. Inelastic Behavior. Constitutive laws for the inelastic behavior of candidate heat sink materials at elevated temperatures need to be developed to allow more accurate predictions of time-dependent stress changes due to plastic deformation and thermal creep. Highest priority are the high strength/high thermal conductivity copper and molybdenum alloys. These constitutive laws should include cyclic effects (e.g., cyclic hardening) so that more accurate fatigue damage calculations can be performed.

V.6.3 INTOR

5. Irradiation Effects to Thermal Properties. The decrease in thermal conductivity due to neutron irradiation needs to be measured for non-metallic high heat flux armor materials, in particular, SiC, TiC, BeO, and graphite.

6. Irradiation Effects to Mechanical Properties. The response of copper and molybdenum heat sink alloys to neutron damage needs to be measured; in particular, the change in yield strength, ultimate strength, and swelling and creep behavior are most important.

7. Simplified Inelastic Stress Codes. Due to the complexity of performing detailed finite element structural analyses, there is a need for simplified inelastic stress analysis programs that can efficiently simulate the long-term response of simple geometries and boundary conditions to swelling and creep deformation. These codes would be useful for doing parametric design studies and for evaluating the impact of newly developed correlations for radiation damage effects. When integrated with models for fatigue,

erosion, crack growth, etc., the package becomes a powerful tool for lifetime predictions.

V.6.4 DEMO/Commercial

8. Creep Crack Growth. The rate of crack growth for materials at constant load and high temperature (i.e., creep crack growth) needs to be measured for candidate heat sink materials, since this may be the life-limiting damage mechanism for long-pulse and steady-state fusion reactors.

9. Fusion Design Code. A new structural design code, similar to the ASME Boiler and Pressure Vessel Code, needs to be written specifically for fusion power reactors.

10. Corrosion. The long-term corrosive effects of boiling water and liquid lithium coolants on copper and molybdenum alloy heat sink materials needs to be addressed.

V.7 Critical Issues

V.7.1 Existing Machines

No critical issues were identified for existing machines with respect to thermomechanical response.

V.7.2 Near Term (TFCX)

1. Bond Integrity. The structural integrity of the attachment interface between the plasma side armor material and the actively cooled heat sink must be demonstrated. This requires thermal fatigue testing of prototype components on ASURF and PMTF, generic component testing on ESURF and EBTF, bond strength and thermal conductance data, and 2-D finite element thermal and stress analysis.

2. Melt Layer Stability. Loss of the melt layer due to eddy current forces during a plasma disruption must be avoided to prevent excessive surface erosion. Laboratory experiments and computer simulations must be performed to assess the severity of this problem and to facilitate design changes that may eliminate the effect.

3. Burnout. Catastrophic failure of an actively cooled heat sink caused by burnout of the coolant channel must be prevented. More accurate correlations for critical heat flux must be developed for relevant component geometries to provide an adequate margin of safety without being overly conservative.

4. Fatigue Crack Growth. Crack growth leading to a coolant leak or failure of the plasma-side material must be prevented. Basic data on fatigue crack growth rates in candidate materials and across interfaces is required for computer simulations. Thermal fatigue testing of intentionally flawed prototype components should be performed to verify the fracture mechanics analysis.

V.7.3 ETR/INTOR

5. Bond Integrity. Neutron radiation damage will degrade the bond between armor and heat sink due to (1) differential swelling stress buildup, (2) impurity transport to the interface, and (3) embrittlement. Long-term irradiation must be performed in FFTF, EBR-II, and FMIT on bonded specimens to develop appropriate damage models that can be used in computer simulations of bond failures.

6. Erosion. Surface erosion is a critical lifetime issue that directly affects the thermomechanical response.

7. Embrittlement. Neutron-induced embrittlement of candidate high heat flux materials must be understood to avoid catastrophic fractures. Fracture toughness measurements on irradiated specimens should be integrated with a fracture mechanics analysis to provide an adequate safety margin.

V.7.4 DEMO/Commercial

8. Creep/Fatigue. The interactions between creep and fatigue damage during neutron irradiation may accelerate crack growth rates and lead to premature failure. More experimental data and modeling are required.

9. Creep Rupture. For long-pulse or steady-state fusion reactors, creep rupture may be the life-limiting failure mode, especially under the influence of radiation damage. More experimental data and modeling are required.

10. Swelling. Low swelling in-vessel materials must be developed for long-term applications because excessive deformations may prevent easy removal for maintenance and repair.

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CHAPTER VI

ELECTROMAGNETIC RESPONSE

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VI. ELECTROMAGNETIC RESPONSE

VI.1 Introduction to the Issues

VI.1.1 Eddy Currents during Normal and Disruptive Plasma Behavior

Eddy currents of significant magnitude are induced in the vacuum vessel and all in-vessel components made of metals when the magnetic fields change, which are generated by surrounding coils and the plasma current. In tokamaks, such changes occur during plasma startup and shutdown, during discharge cleaning procedures, and during plasma disruptions. The induced eddy currents during plasma disruptions are by far the greatest, and since unintentional disruptions take place with a significant probability, they determine to a large degree the design requirements for the vacuum vessel, for in-vessel components, and even for external coils.

The electromagnetic response of high heat flux components to plasma disruptions cannot be considered in isolation from the response of the entire vacuum vessel, other in-vessel components, and the disrupting plasma itself. The reason is that the eddy currents induced during a disruption in the various parts of the vessel are coupled by the mutual inductances and to the decaying plasma current itself.

It is, therefore, inevitable that a global, and somewhat simplistic, eddy-current analysis encompassing the entire vacuum vessel and its internal components must precede any detailed analysis of the electromagnetic response of a high heat flux component such as a limiter. One of the central issues is then how to combine self-consistently the global with the detailed analysis. An equally important issue is the coupling between the induced eddy currents in the vacuum vessel and the decaying plasma current. This coupling can greatly affect the motion of the plasma and the distribution of the energy deposition on high heat flux components during the disruption.

A plasma disruption is characterized, from the point of view of its potential damage to the vessel components, by two phases: the thermal quench and the current decay phase. These phases may partially or fully overlap, and the start and extent of each may vary according to the design of the fusion device.

It has been assumed in recent design studies of future fusion devices that the thermal quench period lasts somewhere between a few milliseconds and 100 ms, whereas the plasma current decay time is significantly longer⁽¹⁻³⁾. As a result, the heating and the thermomechanical response of a limiter is caused mainly by the thermal quench, and the eddy current heating is of minor importance. However, the melt layer that may form on the limiter surface may be rendered unstable when the eddy current forces reach their maximum after the thermal quench.

The lack of a complete understanding of plasma disruptions as well as the anticipated strong coupling between induced eddy currents and the plasma current decay makes it necessary to carefully examine in future development of high heat flux components the full consequences of electromagnetic effects. It is premature to rule out any of the effects to be discussed in the following based solely on present conceptions of tokamak reactors.

VI.1.2 Joule Heating by Eddy Currents

One of the consequences of plasma disruptions and induced eddy currents is the volumetric heating of the vacuum vessel and other metallic in-vessel components. The rate of heating is given by j^2/σ per unit volume and per unit time, where σ is the electrical conductivity and j is the eddy current flux in the component under consideration. The Joule heating by the eddy currents is accompanied by the heat flux from the thermal decay of the plasma during the disruption. Although the latter source of heating is believed to be greater on limiters, Joule heating is not necessarily negligible even though it is deposited more uniformly into the vacuum vessel. The reason is to be found in the relatively small skin depth for rapid current decay times, τ_c . If

$$\kappa = 1/\mu\sigma$$

is the magnetic diffusivity of the metal used for the high heat flux component, σ the electrical conductivity, and μ the permeability, then the skin depth is defined as

$$\delta = 2 \sqrt{\kappa \tau_c}.$$

Table VI.1 lists both the magnetic diffusivity and the skin depth, δ , for disruptions with three different current decay times. For short decay times and good conductors, the penetration depth of both the magnetic field perturbation and the eddy currents is less than the thickness of the component. Accordingly, the eddy current heating can be concentrated in a rather thin surface layer and may, therefore, contribute significantly to the heating, melting, and evaporation of the limiter surface.

VI.1.3 Arcing

Arcing is commonly observed in conjunction with plasma disruptions, and two types must be distinguished.

Unipolar arcs occur between the plasma (acting as an anode) and the vacuum vessel as a result of the sheath potential. The causes for this type of arcing are not well understood.

The second type of arcing may take place between electrically isolated sections of the vacuum vessel or of limiters. In order to increase the resistance of an in-vessel component and reduce the overall eddy currents, the components and the vacuum vessel are segmented. Interrupting the path for eddy currents, however, creates large voltage gaps with the potential for arc formation during disruptions. Preventing this type of arcing requires a detailed analysis of the magnitude and distribution of eddy currents following a disruption.

VI.1.4 Induced Forces

The generation of eddy currents leads to very large forces in the vacuum vessel and in-vessel components. These induced forces can be divided into two groups: one is created by the interaction of eddy currents with the external static magnetic fields, and one by the interaction of eddy currents themselves. The latter group is of minor importance for the solid structures, but it may play a significant role in the stability of melt layers formed during a disruption. The former group determines to a large extent the structural support for the vacuum vessel and all in-vessel components, and the electromagnetic forces represent, therefore, the major consequence of plasma disruption to the design of high heat flux components. The particular needs for the eddy current and force evaluation will be discussed at length in Section VI.3.

Table VI.1

Magnetic Diffusivity, κ , and Skin Depths, δ , for a Few Metallic Materials

Metal	κ (cm ² /s)*	δ (cm)		
		100 μ s	1 ms	10 ms
Cu	126	0.22	0.71	2.2
Al	204	0.29	0.90	2.9
AISI 304	5900	1.54	4.86	15.4
W	390	0.39	1.25	3.9

*Based on electrical conductivities at room temperature

VI.2 Eddy Current Codes

VI.2.1 Methods for Evaluating Eddy Current

In principle, eddy currents can be evaluated by solving the Maxwell equations

$$\nabla \times \underline{H} = \underline{J}, \quad \nabla \cdot \underline{B} = 0, \quad \nabla \times \underline{E} = -\frac{\partial \underline{B}}{\partial t} \quad (1)$$

subject to the constitutive laws

$$\underline{B} = \mu \underline{H}, \quad \underline{J} = \sigma \underline{E}. \quad (2)$$

It is customary in eddy current problems to neglect the displacement currents, i.e., $\partial D / \partial t = 0$. This assumption can be made in a conducting medium if the frequency, ω , of the electromagnetic field satisfies the condition $1/\omega \gg \epsilon/\sigma$ and if the electron or charge carrier response time, τ_e , to an electric field satisfies $\tau_e = 10^{-14}$ seconds. Hence, for all frequencies up to the infrared spectrum, the assumption $\partial D / \partial t = 0$ is justified.

For the solution of the above equations, a vector potential $\underline{A}$ and a scalar potential ϕ are introduced such that

$$\underline{B} = \nabla \times \underline{A}, \quad \underline{E} = -\frac{\partial \underline{A}}{\partial t} - \nabla \phi \quad (3)$$

thereby satisfying two of the Maxwell equations, i.e., $\nabla \cdot \underline{B} = 0$, and $\nabla \times \underline{E} = -\frac{\partial \underline{B}}{\partial t}$. The third equation together with $\underline{B} = \mu \underline{H}$ leads to

$$\nabla^2 \underline{A} - \sigma \mu \underline{A} - \nabla(\nabla \cdot \underline{A}) + (\nabla \underline{A}) \cdot (\nabla \ln \mu) - (\nabla \ln \mu) \cdot (\nabla \underline{A}) = \sigma \mu \nabla \phi \quad (4)$$

whereas the continuity equation $\nabla \cdot \underline{J} = 0$ results in

$$\nabla^2 \phi + (\nabla \ln \sigma) \cdot (\nabla \phi) = -\nabla \cdot \underline{A} - \underline{A} \cdot (\nabla \sigma). \quad (5)$$

For non-ferromagnetic materials ($\mu = \text{const.}$) and the Coulomb gauge $\nabla \cdot \underline{A} = 0$, these equations simplify to

$$\nabla^2 \underline{A} - \sigma \underline{A} = \sigma \mu \nabla \phi \quad (4a)$$

$$\nabla^2 \phi + (\nabla \ln \sigma) \cdot (\nabla \phi) = -\sigma \underline{A} (\nabla \ln \sigma). \quad (5a)$$

None of the available eddy current codes is presently capable of solving Eqs. (4a) and (5a). Instead, σ and μ must be constants in a limited number of regions for some codes, or constants through the entire component for most codes. In both cases, it is then possible to set $\phi \approx 0$. Both the strong temperature dependence of σ and the composite nature of high heat flux components makes it necessary to develop new eddy current codes for variable conductivities and 3-D geometries.

VI.2.2 2-D Codes and 3-D Codes

With regard to the electromagnetic forces exerted on the first wall and the in-vessel components, the eddy current distribution through the wall thickness is of little concern. As a result, the assumption can be made that the wall is a shell structure and the eddy currents are uniform across the wall, reducing the eddy current pattern to a two-dimensional one on the surface of a shell.

A survey of eddy current codes by Lari and Turner⁽⁴⁾ indicates that a large variety of codes exists for the 2-D analysis of eddy currents in rectangular and cylindrical geometries. The list, shown in Table VI.2, of 3-D eddy current codes is significantly shorter. The asterisk next to the code name or number indicates those codes which allow for continuous spatial variation of the electrical conductivity.

VI.3 Coupling of Eddy Currents to Thermomechanical Response

The three responses of an in-vessel component to plasma disruptions, namely heating, electromagnetic loading, and transient stresses are inter-related as indicated in Fig. VI.1. A particular intricate feedback between heating and eddy current distribution exists through the strong temperature dependence of the electrical conductivity and the Joule heating. For this reason, an accurate 3-D evaluation of the eddy current distribution with depth is considered to be important.

Table VI.2. Rectangular Geometry Eddy Current Computer Programs

Dimensions	Geometry 2 D (X-Y) Field 2 D Current 1 D					
Steel B-H Relationship	None		Linear		Nonlinear	
Changing Magnetic Field or Changing Source Currents	Steady State	Transient	Steady State	Transient	Steady State	Transient
Finite Difference Finite Elements Variational Formulation	HYBRID-5 AOSMAG-18* TRIDIF-19 COMPELL-22* FLUX2D-44	NMLMAP-7 TRIDIF-19 COMPELL-22* FINELEM-36 DIFDIRA-63	HYBRID-5 PE2D-8 NONAME-14 AOSMAG-18* TRIDIF-19 COMPELL-22* NONAME-23 SKIN1-27 FLUX2D-44 PWMS-48 MAGNET11-54 CSWEDDY-60	NMLMAP-7 PE2D-8 TRIDIF-19 COMPELL-22* NONAME-23 FINELEM-36	HYBRID-5 TRIDIF-19 COMPELL-22* NONAME-23 PWMS-48 SCEDDY-52	NMLMAP-7 PE2D-8 TRIDIF-19 COMPELL-22* NONAME-23
Integral Equation Method	HYBRID-5 NONAME-20	INTEQ-38	HYBRID-5 NONAME-20	INTEQ-38	HYBRID-5	
Other, Open Body, Elements Integrodifferential	AOSMAG-18		AOSMAG-18 SKIN1-27 INTEQA-38			

*Codes with capabilities for spatial variation of the electrical conductivity

Figure VI.1. Interrelation of the three responses (heating, electromagnetic loading, and transient stresses) of an in-vessel component to plasma disruptions.

VI.3.1 Joule Heating

The strong coupling between thermal response and eddy currents becomes obvious when considering the heat conduction equation

$$\nabla \cdot [k(T)\nabla T] - c(T)\dot{T} = -\sigma(T)(\vec{A} + V\phi)^2 \quad (6)$$

where $k(T)$, $c(T)$, and $\sigma(T)$ are the thermal conductivity, the specific heat, and the electrical conductivity, respectively. Here, the Joule heating is first expressed in terms of the electric field and then in terms of the vector and scalar potentials. For a homogeneous material, the material parameters become spatially dependent by virtue of their temperature dependence. In composite materials such as coated metals, an additional spatial dependence exists by design.

The temperature dependence of σ requires that the electromagnetic field equations (4a) and (5a) be solved simultaneously with the heat conduction problem if Joule heating contributes to the thermal response of the in-vessel components to disruptions. None of the present eddy current codes allows for such a coupling to the thermal response.

VI.3.2 Electromagnetic Force Distribution in Solid Structures

Even though the eddy currents produce a spatially distributed body force, $\vec{J} \times \vec{B}$, per unit volume, it is not always essential for the stress analysis of solid in-vessel components to have the precise electromagnetic force distribution throughout the components. For example, if attachment bolts for a limiter need to be designed, it suffices to evaluate the overall electromagnetic forces on the limiter but not the detailed force distribution. Eddy current codes for 2-D analysis are then adequate.

On the other hand, when considering eddy current forces of components plated with metallic coatings, the depth distribution of eddy currents must be evaluated. It appears then that different analysis tools are required depending on the particular application.

The eddy current forces can induce large amplitudes of vibrations in thin-walled or slender structures. The motion of the conducting structure produces, in turn, additional eddy currents of magnitude $\sigma \vec{v} \times \vec{B}$, where $\vec{v}$ is

the velocity of the material point. These additional eddy currents can produce a significant damping of the vibration and thereby reduce the amplitude of the dynamic stresses⁽⁵⁾. This beneficial effect may, however, be offset by the additional Joule heating which increases the transient thermal stresses. If this feedback into the thermal response is indeed significant, the electromagnetic, elastodynamic, and thermal analysis must all be carried out simultaneously.

VI.3.3 Electromagnetic Force Distribution in Melt Layers

Severe plasma disruption may lead to surface melting of in-vessel components. However, for the short duration of the melt layer, gravitational forces are usually not sufficient to remove the thin melt layer. Therefore, the eddy current forces are of major concern⁽⁶⁾. To correctly evaluate the distribution of these forces, a 3-D analysis as outlined above is required taking into account the depth- and time-dependent materials properties. For the analysis of the melt layer stability, it is also important to consider the self-interaction of the eddy currents in addition to the interaction with the external magnetic field.

VI.4 Conclusions and Recommendations

Eddy currents and induced electromagnetic forces impose severe design requirements on high heat flux components of both near-term and future fusion reactors. It is, therefore, absolutely essential to have adequate tools available to analyze and predict correctly the eddy current distribution and the associated heating and the forces on any in-vessel component. The consequences of a faulty design could be disastrous. Furthermore, optimal designs can only be achieved with the correct tools.

The major issues which arise in the analysis and evaluation of plasma disruptions and their consequences are as follows:

- a) The induced eddy currents react back on the decaying plasma current. This affects the motion of the plasma and the subsequent energy deposition on in-vessel components. Such coupling or feedback has not been adequately treated in previous design studies.

- b) The magnitude, direction, duration, and distribution of the electromagnetic forces have been evaluated for simplified geometries of in-vessel components but not for real geometries.
- c) High heat flux components in future devices and reactors will most likely be heterogeneous and have complex geometries with grooves, cuts, and convoluted coolant channels.
- d) Local arcing and erosion can occur between parts or segments of a high heat flux component which are separated by a high resistivity path.
- e) Melt layers which form in response to the thermal quench can be removed by the eddy current forces.

The adequacy of present tools to address the above issues is judged somewhat differently by various researchers⁽⁷⁾. Whereas system designers consider present analysis tools adequate, but stress the urgent need for vigorously addressing issue (a), materials analysts and component designers consider a 3-D analysis capability absolutely essential to address the issues (b) through (e).

This difference in emphasis is a reflection of two different but coupled objectives and the best strategies to achieve them. A global analysis of the eddy currents must by necessity seek an approximate model for the entire first wall and the other in-vessel components. For this reason, eddy current codes are inadequate to address the issues (a) through (e). For the design of such critical components as, for example, an actively cooled limiter covered with tiles of beryllium, it is essential to develop a 3-D analysis capability for eddy currents, Joule heating, and electromagnetic forces.

The following tasks are, therefore, recommended with highest priority:

- Develop a 3-D eddy current code for high heat flux components which incorporate in their design different materials with temperature-dependent conductivities.
- Integrate this code with a general purpose, finite element structural analysis code and incorporate magnetic damping.
- Coordinate this development with a corresponding one in plasma engineering which deals with the plasma behavior in disruptions and with the global eddy current analysis.

- Conduct benchmark tests in FELIX and/or existing fusion devices to confirm design code results.
- Conduct simulation experiments on melt layer formation and their stability under electromagnetic forces.

All of the above efforts represent essential tasks which are necessary in order to develop reliable high heat flux components. As supporting efforts, it is recommended to:

- Test prototype components in FELIX.
- Evaluate the need for a coupled thermal and eddy current analysis.
- Investigate the coupling between magnetic damping, Joule heating, and electromagnetic forces.

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CHAPTER VII

HIGH HEAT FLUX COMPONENT TEST FACILITIES

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VII. HIGH HEAT FLUX COMPONENT TEST FACILITIES

There are several facilities already in operation or under construction that are dedicated to high heat flux component testing or that can be used for this purpose. Some of these facilities are specifically part of and supported by the High Heat Flux Component Development Program. Other facilities are owned by different parts of the fusion program but can be used for component testing. There are three categories of facilities that can be considered for high heat flux component testing. The first of these are dedicated facilities or those that could be dedicated to high heat flux testing; ten such facilities have been identified. Seven of these are operational: the ESURF, ASURF and ISURF facilities at Westinghouse, the Sandia-Albuquerque e-beam facility, an ion-beam facility at TRW, the PISCES ion-beam facility at UCLA and the direct converter test stand at LLNL. The Plasma Materials Test Facility (PMTF) at Sandia-Albuquerque is under construction. An electron-beam facility at HEDL is being modified for use in a hot cell to test irradiated components. A magnetic mirror facility designated ICTF (Impurity Control Test Facility) is being considered by ANL for high heat flux component and impurity control testing. The ICTF would potentially use some components from FELIX (Fusion Electromagnetic Induction Experiment). The characteristics of these facilities are described in Tables VII-1 and VII-2. The second type of facility that can be used for high heat flux testing is the neutral beam or RF test stand. Although component testing is not the primary function of these facilities, they have extensive capabilities that can be used for the high heat flux program. The neutral beam stands have the capability for both high energy ion and neutral bombardment of surfaces. Three large test stands are operational at LBL and ORNL. The capabilities of the LBL neutral beam test stand, the Medium Energy Test Facility and the High Power Test Facility at ORNL are listed in Table VII-3. The RF Test Facility (RFTF) at ORNL is in the planning stage. While its primary purpose is testing RF component and system performance, it should be suitable for testing erosion of plasma side components. The capabilities of the RFTF are also given in Table VII-3. The third type of facility that has high heat flux test capability is a fusion device itself. These have not been included in the tabulation of facility characteristics because of limited availability for testing and because the components for a fusion device generally must be tested elsewhere before being used in the machine. However, these facilities should be used for

Table VII-1. Dedicated Electron Test Facilities

	ESURF	ASURF	Sandia	HEDL
Particle Type	electron	electron	electron	electron
Energy (keV)	150	16	30	12-25
Current (A)	.005-.33	6	1	2.5-5
Power (kW)	36	64	30	60
Peak Power (kW)	50	100	-	-
Pulse Length (s)	ms-CW	ms-CW	ms-CW	.5-CW
Target Area (cm ²)	.01-500	100-10,000	.01-100	1-200
Heat Flux (kW/cm ²)	up to 5,000	up to 2,000	.3-3,000	.3-60
Coolant	water	water	water	water
Pressure (psi)	1000	2200	300	120
Flow Rate (gpm)	20	60	50	20
Heat Removal (kW)	100	2000	30	60
Status	operational being phased out	operational	operational	e-beam operational being modified for hot cell
Under HHFCDP	no	yes	yes	yes

Table VII-2. Dedicated Ion Test Facilities

	PMTF	PISCES	TRW	ICTF	LLNL	ISURF
Particle Type	ions H	ions H,D,He	ions any gas	mixed H ⁺ , H ⁻ , He, e	ions H, He	ions H to Xe
Energy (keV)	40-80	.05-.5	.05-2	.05-.5	50-100	.5-30
Current (A)	40-20	50	10	100	.1	0-2x10 ⁻⁵
Current Density (mA/cm ²)		1000	100		1-10	
Power (kW)	1600	25	20	~ 700	10	
Pulse Length (s)	30	CW	CW	CW	CW	
Target Area (cm ²)	100-200	50	100	1000 (2x500)	10-100	1
Heat Flux (kW/cm ²)	8	.05-.25	.005-.2	.2 (.77 max)	0.1-1	6x10 ⁻⁴
Beam Uniformity (%)		10	5%			
Beam Impurities			Low, ion species mix undetermined			
Coolant	water	water	water	water	water	
Pressure (psi)	1000	200				
Flow Rate (gpm)	500					
Heat Removal (kW)	10,000					
Status	construction	operational	operational	planning	operational	operational
Under HHFCDP	yes	yes	no	no	no	no

Table VII-3. Neutral Beam and RF Test Stand Characteristics

	LBL	ORNL METF	ORNL HPTF	ORNL RPTF
Particle Type	ions H,D	ions H	ions H	mixed H ⁺ ,D ⁺ ,He ⁺ ,e
Energy (keV)	40-120	80	120	~ 0.01
Current (A)	50	60	30	
Current Density (mA/cm ²)	< 100	240	250	
Power (kW)		4000	3600	1000
Pulse Length (s)	< 30	30	10	CW
Duty Cycle (s)	300	150	50	
Target Area (cm ²)		1000	100	3000
Heat Flux (kW/cm ²)	~ 2	< 30	< 50	0.4
Coolant	water	water	water	water
Pressure (psi)		250	250	250
Flow Rate (gpm)		1000	1000	> 1200
Heat Removal (kW)		> 4000	> 6000	> 1000
Status	operational	operational	operational	planned
Under MHFCDP	no	no	no	no

confirmation testing, when possible, because they simulate or match the actual hostile environment.

All the dedicated electron beam facilities have some similar characteristics. All are scanned, either raster or line scan. Thus, very high heat fluxes may be obtained by a preprogrammed scan pattern that limits the area. ASURF is the only facility capable of testing the first wall of a water-cooled commercial reactor because of its high pressure coolant loop. However, the 250 to 300 psi that is available in most other facilities is sufficient to test limiters, divertors, beam dumps and other components that are water cooled without attempting to get high quality heat. The other facilities could be used for these low pressure components. The direct converter plates and halo scraper for the Mirror Advanced Reactor Study (MARS) power plant operate with coolant pressure above 2000 psi and could only be tested in ASURF because of their high pressure. Planned, but not yet approved, upgrades for ASURF include an increase in the total beam power to 1 MW, incorporation of liquid lithium and/or helium cooling loops, and the addition of a pulsed electron gun that would be used to simulate disruption heat loads over areas of about 10 cm^2 . ASURF has raster speed of $5 \times 10^4 \text{ cm/s}$ while that of ESURF is $4 \times 10^5 \text{ cm/s}$. ESURF has some disruption simulation capability with a beam rise time of about 20 μs and a dwell time from 1 to 110 ms. The HEDL test facility will be in a hot cell. Its unique function would be to test components that have been subjected to neutron damage in a reactor. This sequential damage is the only mechanism to test in-vessel components for an ETR. The limit on component size is determined by fission reactor test volumes at this time. It is expected that multiple components would fill the e-beam facility test volume. The principal objection to electron beam facilities is that most components are primarily subject to ion damage; thus, the true environment is not being simulated. One exception is the tandem mirror direct converter plates which are primarily subject to energetic electron bombardment because α ions are induced to leave the plasma radially by drift pumping.

The dedicated ion facilities fall into two groups. The PMTF is designed to test moderate sized components with ion-induced heat fluxes at sufficiently high levels to simulate steady-state component operation in tokamaks and mirrors. Because ions are used, the environment is close to that of a fusion

device. The ion energy is a good match for neutral beamline components, beam dumps, regions of high charge exchange flux from neutral beams, and mirror end cell components. However, the energy is significantly higher than the edge temperature of a tokamak. The first phase of the PMTF has the highest power of all the planned and existing facilities. It is planned to add a second source in the future and increase the facility power by more than a factor of two.

The other facilities, PISCES, TRW, ICTF, ISURF and LLNL are principally to perform physics, materials and engineering experiments. Specific uses are erosion studies including sputtering, blistering, and redeposition, ion recycling, gas conductances, and limiter/diverter pumping efficiencies. They are typically better controlled and diagnosed than the larger facility. ICTF, being essentially a confinement device where all other facilities are accelerators, will have a mixed spectrum of ions, neutrals and electrons. Most of the energy is deposited by the ions. Testing in this mixed spectrum, which is typical of many fusion device components, may be of significant value where synergistic effects are important. The LLNL facility has operated for many individual runs of several hours each with an accumulated run time of 80 hours on a tungsten collector. It has a rather low operating expense. The TRW facility has been run for graphite sputtering with continuous operating times in excess of 4 hours and controllable ion energies between 50 eV and 2 keV. The current density depends on oscillator power and is now limited to about 100 mA/cm² and could reach 500 mA/cm². The ISURF is an ion facility jointly operated by Westinghouse and Penn State for low energy bombardment of small samples.

Westinghouse is planning to combine the ISURF and ESURF facilities, that are located in the same laboratory, through the cooling loop, power supplies and data acquisition system and adding an electron gun to ISURF for simultaneous irradiations. A simulation of combined ion surface interactions and electron surface heating may now be accomplished by rotation of specimens in ISURF and ESURF.

The in-situ diagnostics available at these facilities are an important part of testing. PISCES has, or is planned to have by FY 1985 plasma probes to measure n_e and T_e , residual gas analyzer, calorimeter, E×B mass analyzer, H α interferometer, microwave interferometer, and an energy analyzer. The TRW

facility has an optical pyrometer, thermocouples, a residual gas analyzer, ion current monitor, ion energy measurement, and source power monitor. The HEDL facility is planned to include pyrometers, thermocouples and optical viewing in addition to electron beam monitors. Identical capabilities are available at Sandia and Westinghouse.

CHAPTER VIII

INSTRUMENTATION AND CONTROL

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VIII. INSTRUMENTATION AND CONTROL

VIII.1 Introduction

The instrumentation and control (I&C) of HMF components for magnetic fusion devices is an area in the fusion effort which has received relatively little attention to date. This state of neglect is somewhat understandable in the context of the overwhelming physics problems associated with a program struggling to demonstrate scientific feasibility. I&C efforts in the fusion program have primarily focused on generating, maintaining, and characterizing millisecond bursts of plasma, a heavily diagnostic program which, by nature of the shortness of the event, is very limited with respect to closed loop feedback control features. Also due to these short pulse lengths, there has been no need for active cooling of these components. This has eliminated the need for a major portion of the I&C which will be required for future HMF components. The lack of experience with operating devices which produce significant loads on these control features with respect to both component performance and other device systems (i.e., the data base required to develop proper control algorithms is insufficient at this time.

With the next phase of device operation (present machines and their upgrades), it will become important to begin to address I&C of HMF components. The severity of impact on device operation of the failure of any actively cooled structure within the plasma chamber dictates that the performance of these HMF components will need to be continuously monitored. Although significant control will probably not be utilized on these devices, the increased use of instrumentation will allow the data base generation for control algorithms to proceed and should provide component designers with important source parameters. The devices presently in the conceptual design phase should include the ability to test some of these control schemes so that a few of these techniques can be incorporated into the standard control logic of devices such as ETR and DEMO.

The next few sections discuss the I&C which are, and will be, required to assure proper HMF component performance and compatibility with the whole fusion device. The presently utilized and potential techniques and the environmental constraints on the use of these techniques will be discussed. Finally, conclusions and recommendations with respect to the present use and future development of HMF component I&C are presented.

VIII.2 Measurement Requirements

Measurements are required on HHF components for primarily two reasons: 1) to monitor the performance of the component from a reliability standpoint and 2) to monitor key parameters which determine whether the component is operating in a mode which is compatible with operation of the fusion device. Measurements made for either of these two reasons can provide control input to a feedback loop or device operator or can simply provide data which is used as input to the component or device development/design process. Table VIII-1 gives specific examples of measurements which will be required in the evolution to a commercial reactor. It is expected that most measurements will be required during the development stages and in off-line development/test devices. The variety of measurements required will gradually decrease as the components are implemented in the devices and the devices near commercialization, with the most critical needs surviving the development process. The next two sections describe some of the required measurements for HHF components with respect to performance/reliability monitoring and device compatibility. Techniques for performing these measurements, also listed in Table VIII-1, will be discussed in Section VIII.5.

VIII.2.1 Component Performance/Reliability

Since the main concern in the design of HHF components is the ability to withstand the high heat and particle loads in a fusion device environment, the performance and reliability measurements would tend to provide measures of the thermomechanical and thermal-hydraulic response of the HHF component. The primary categories of Table VIII-1 which include these measurements are temperature, strain/displacement, and coolant properties. During operation in the actual fusion device, measurements of surface temperature, surface erosion (including clad/coating integrity), and deformation or cracking of the component which may indicate imminent component failure will probably be important to perform. The coolant temperature (especially at inlet and outlet of the heated section), pressure drop, inlet (or outlet) pressure and flow rate will also be important variables to monitor during operation. Through these measurements, one should be able to get a reasonable picture of how the component is performing.

Table VIII.1

Measurement Requirements and Potential Techniques for HNF Components

General Parameter Category	Potential Measurement Techniques	Examples of Specific Measurement Requirements
Temperature	<ul style="list-style-type: none"> ● Thermocouples* ● Resistance temperature detectors* ● Infra-red pyrometers* ● Fluoroptics ● Acoustic/sonic (gas temp.) 	<ul style="list-style-type: none"> ● Surface temperatures ● Heat sink/coolant temperatures ● Heat transfer rates ● Disruption characterization ● Critical heat flux (burnout)
Strain/Stress/ Displacement	<ul style="list-style-type: none"> ● Foil gauges* ● Crack propagation gauges ● Accelerometers* ● LVDT's/RVDT's* ● Optical moire techniques ● Holography ● Laser specklegrams ● Acoustic emission 	<ul style="list-style-type: none"> ● Imminent component failure (i.e., surface strain/cracking) ● Warpage/buckling/deformation ● Erosion ● Vibration ● Component stress and stress distributions
Surface Conditions	<ul style="list-style-type: none"> ● IR pyrometer* ● High speed photography* ● IR/UV CCTV* ● Quartz oscillator ● Implanted tracer materials ● Implanted erosion markers ● Optical reflectometer ● Laser techniques (i.e., moire, holography, etc.) ● X-ray diffraction ● Metallography/surface analysis off-line* ● Laser absorption spectroscopy 	<ul style="list-style-type: none"> ● Surface erosion rate/distribution ● Redeposition rate/distribution ● Surface melting detection ● Surface temperature ● Bond integrity ● Coating integrity ● Metallurgical response ● Internal coolant channel erosion/corrosion
Electrical/Magnetic Parameters	<ul style="list-style-type: none"> ● Hall effect gaussmeter* ● Eddy current detectors ● Floating potential measurements* 	<ul style="list-style-type: none"> ● Plasma position ● Disruption characterization ● E-M field measurements at component

Table VIII.1 (contd.)

General Parameter Category	Potential Measurement Techniques	Examples of Specific Measurement Requirements
Coolant Properties	<ul style="list-style-type: none"> • Temperature techniques* • ΔP flowmeters* • Turbine flowmeter* • Magnetic flowmeters • Pressure gauges* • Tritium monitors • Calorimetry* • Void fraction techniques • Acoustic emission • Coolant chemistry monitors* 	<ul style="list-style-type: none"> • Coolant leaks • Critical heat flux (burnout) • Two-phase flow characterization • Heat transfer coefficient
Vacuum	<ul style="list-style-type: none"> • Vacuum gauges* • Residual gas analyzers* • Tracer gases 	<ul style="list-style-type: none"> • Coolant leaks • Pump limiter/divertor performance
Power	<ul style="list-style-type: none"> • Heat flux monitors • Calorimetry* • Radiation detector* • Bolometers* • Thermocouples* • IR detectors* 	<ul style="list-style-type: none"> • Plasma power/distribution • Diverted plasma power • Plasma position • Neutral beam absorption • Neutral beam power/distribution • Direct conversion efficiency • Beam position

*Presently being used on existing fusion devices.

Additional measurements which include internal temperatures of the component, critical heat flux measurements, two-phase flow characterization parameters, heat transfer rates, and stress distributions will be required during component development in off-line test facilities. This type of information gives the designers a data base and proof test results from which to base their final designs but will probably not require measurement in the device if sufficient development has preceded the component installation. Information on cyclic fatigue, vibration, flow instabilities, coolant channel erosion, and clad or coating survivability can also be provided by off-line test facilities for prototypically sized HNF components but should not require implementation in the fusion device.

VIII.2.2 Device Compatibility

Compatibility of a HNF component with the overall fusion device is achieved when the component successfully performs the function for which it was designed (i.e., divertor removes impurities, direct convertor slows down and collects energetic particles and converts to electrical energy), performs the function within the intended design window, and produces no unexpected and undesirable effects on the device operation. In addition, to address whether the component is performing within the intended design window, determination of the source parameters (i.e., plasma power/distribution) under which the component is operating should be made.

Measurements in all categories of Table VIII-1 might be required to determine how well the HNF component is performing its function. Since these measurements are intended to provide information on how the component interacts with the fusion device, it is expected that all these measurements would be done during device operation and not in off-line test facilities.

Measurements such as surface temperature and erosion rate and distribution are just as desirable here as they are for examining component reliability. In addition, redeposition rates and distributions, detection of surface melting, and characterization of disruptions (frequency, magnitude, duration, and location) would be desirable measurements. Performance characteristics for pump limiters and divertors can be measured to some extent through the vacuum levels in divertor chambers and pump limiter throat regions. Coolant leaks into the vacuum chamber can also be detected by

ambient vacuum levels as well as plasma parameters. Finally, power measurements may be the most critical and useful measurements to be made which detail interaction between the HHF components and the plasma. Plasma power and distribution on HHF components, diverted plasma power, relative plasma position, beam absorption in the plasma, beam power and distribution, beam positions, and direct convertor efficiency (loadings) will probably all require measurement at some time.

VIII.3 Control Requirements

The control aspects of HHF components feature a mixture of open and closed (feedback) loop systems which are responsible to maintain key component parameters within the design windows (which, in turn, should insure reliability of the components), and provide control input to other device subsystems or components. The present devices primarily operate in open loop control modes, with diagnostics providing input to an operator who provides a control function if necessary. The exception to this is mainly in the control of component cooling systems, where closed loop control through stand-alone process controllers maintain prescribed performance characteristics. Future devices will demand increasingly automated operation with closed control loops which interface directly with the main reactor control system. Another aspect of HHF component control involves the early detection or anticipation of component failures and minimization of the impact or total prevention of these failures. The anticipation and prevention of failures will fall into the category of component performance/reliability control. The early detection and impact minimization assume component failure and thus should be addressed in the context of total device control. The following sections outline some of the anticipated and present control features of HHF components with respect to component performance and overall device operation.

VIII.3.1 Component Performance/Reliability

Most requirements for control systems which deal with HHF component performance address thermomechanical and thermal-hydraulic parameters. While the surface condition of the plasma-facing portion of a HHF component is a major consideration in addressing the performance and reliability, active control is either non-existent or will be discussed in the context of control

of the entire device performance. The I&C of surface conditions of HNF components is also a grey area for which responsibilities may be assumed by this task group and/or the PMI Task Group. It is recommended that both groups define requirements in this area while maintaining a strong and continuous interaction.

Table VIII-2 lists some of the potential control loops which involve HNF components as either sensors or actuators. The controlled parameters in this table that deal with HNF component performance would include optimization of limiter positions (although it could be argued that this primarily impacts device performance), component erosion, and component integrity (which includes failure anticipation and prevention).

The control of limiter positions is presently being utilized on devices with movable limiters such as TFTR. A simple relative position sensor with feedback to a servomotor of some sort is adequate for this task. Both of these devices can be located outside of the most severe environment of the reactor. Optimization of this position can be defined by several parameters including limiter temperature, vacuum levels around or behind the limiter, species levels in the ambient vacuum, limiter power, or operator preference. Specific algorithms and optimization criteria will need to be developed for each fusion device separately.

HNF component erosion portends to be a very serious problem in all reactor designs to date. Methods to control the erosion rates and distribution through plasma tailoring would be desirable from the point of view of HNF component reliability; however, techniques which accomplish this would appear to be very difficult if not impossible. If indeed possible, they would probably involve strong trade-offs with plasma performance, in which case the erosion control methods would probably not be implemented. A much more realistic approach is to design HNF components for minimal erosion, monitor the erosion through the most convenient method, and schedule replacement or in-situ recoating of the HNF component, or reduce operating levels (possibly shutdown) of the fusion device. The sensing techniques listed for HNF component erosion monitoring in Table VIII-2 are described more fully in Section VIII.5. Each of these could provide a signal to the device operator which would indicate that action is necessary to prevent component or device failure due to excessive erosion.

Table VIII.2

Potential Control Loops Involving HHF Components

Controlled Parameter	Sensing Parameter or Device	Activating Parameter or Device
Optimization of limiter position	<ul style="list-style-type: none"> • limiter position • limiter temperature • limiter power 	<ul style="list-style-type: none"> • limiter drive
Plasma position	<ul style="list-style-type: none"> • limiter power • limiter temperature 	<ul style="list-style-type: none"> • field coils
Beam strike position	<ul style="list-style-type: none"> • dump power • dump temperature • duct temperature • scrape-off panel temperature and power 	<ul style="list-style-type: none"> • source position • grid voltages/position • arc voltage
Impurity levels	<ul style="list-style-type: none"> • RGA signals • vacuum levels • divertor power 	<ul style="list-style-type: none"> • pellet injectors • gas puffers • divertor field coils
Optimization of pump limiter position	<ul style="list-style-type: none"> • vacuum/RGA signals behind limiter • limiter power 	<ul style="list-style-type: none"> • limiter drive
Total reactor power	<ul style="list-style-type: none"> • first wall power • limiter power • divertor power 	<ul style="list-style-type: none"> • beam power/duty • RF power/duty • divertor field coils • pump limiter position • impurity injection
HHF component erosion	<ul style="list-style-type: none"> • infrared signals • tracer detection • surface physics analysis station • marker observation • out-of-plane distortion signals • quartz oscillator signals 	<ul style="list-style-type: none"> • component replacement • reactor shutdown • in-situ coating schedule
HHF component integrity	<ul style="list-style-type: none"> • infrared emission • ultrasonic emission • strain/distortion signals • thermocouple signals • visual observation 	<ul style="list-style-type: none"> • coolant system adjustments • reactor shutdown • component replacement

Table VIII.2 (contd.)

Controlled Parameter	Sensing Parameter or Device	Activating Parameter or Device
Impact minimization on plasma due to HHF compo- nent failure	<ul style="list-style-type: none"> • all of above plus: <ul style="list-style-type: none"> • RGA signals (tracer detection) • vacuum levels • coolant pressure • plasma diagnostics • hard x-ray detectors 	<ul style="list-style-type: none"> • reactor shutdown • vacuum/electric iso- lation of vacuum vessel

The control of HMF component integrity is primarily a case of anticipating failure and preventing the failure if possible, as is erosion control. The failure mechanisms in HMF components consist of thermal (critical heat flux, loss of cooling, etc.) or mechanical (cladding/coating failure, internal wall erosion, wall cracking, etc.) failures. Several techniques can be used to indicate imminent failure including excessive temperature rate of rise, abnormal temperatures, acoustic emission signals from the component material, and excessive distortion. Abnormal occurrences such as these can be sensed and programmed to trip off neutral beams, terminate a plasma pulse, or make adjustments in cooling system parameters or device power levels. Existing devices already incorporate many of these "interlock" functions; however, development of much more sophisticated failure prediction techniques should be pursued for future reactor designs since the benefits of successful techniques in this area are great due to the severe impact of HMF component failure on device availability.

VIII.3.2 Device Operation

The controlled parameters in Table VIII-2 which involve control of device or device subsystems through the use of HMF components include plasma position, beam strike position, plasma impurity levels, reactor power, and minimization of the impact of a HMF component failure on device operation.

Existing devices and designs of future devices already use limiters for plasma position control. Movable limiters define the plasma edge in TFTR, and limiters provide TFCX feedback control to field coils to help locate the plasma in some designs. Most future reactor designs will probably use some sort of feedback loop which senses limiter power or temperature and feeds back to field-shaping or positioning coils to adjust operating parameters or control long time-constant plasma drifts. Present devices can provide a data base for algorithm development for this control loop.

Most neutral beam systems presently use some indication of beam-line component loading (i.e., segmented calorimeter beam dumps) to locate the beam or provide divergence information. Automatic control of beam position has been designed for neutral beam test stands which gradually position the beam through an iterative process on the pulsed beams. Future long-pulse or steady-state beams will require active feedback to source manipulators, grids,

and/or arc supplies to provide continuous beam positioning. Once again, present devices must provide the algorithm data base.

Impurity levels might be controlled automatically by feedback signals from vacuum gauges and RGA's located in the plasma edge region, divertor chamber, or pump limiter throats. Pellet injectors, gas puffers, impurity injectors, divertor field coils, or pump limiter positioners might serve as actuators in this case.

Active power control will be necessary for eventual power reactor operation. The sensed parameters will invariably be power to the limiters, first walls, and divertors, and could be measured by several of the techniques outlined in Table VIII-1. The actual control over the plasma power level will probably be accomplished through adjustment of beam and RF powers and duty cycles, divertor efficiency (field coils), pump limiter efficiency (position), and/or impurity injection rates. Again, data base generation on near-term devices is essential for development of these control systems.

Finally, in the event of HHF component failure, one would desire to minimize the impact of the failure on the device operation. Assuming the component has failed, is essential to operation, and no backup mechanism exists, the entire device will have to be shut down (or at least idled in some mode which permits repair of the component). To minimize damage to other systems and cleanup time, the HHF coolant system should be isolated with fast acting valves, and all vacuum and electrical interfaces with the vacuum vessel should be isolated in a mode compatible with each of the device subsystems. Automatic interlocks will have to be provided which trigger on signals from fast sensing devices. These sensors could include vacuum or RGA signals, coolant pressure, or plasma diagnostics. Development of trigger signals which emanate from fusion HHF component instrumentation might be accomplished in off-line facilities during HHF component development. Other data for these interlock systems might be obtained from operating devices under fault conditions or simulated fault conditions.

VIII.4 Operating Environment and Device Interfacing

The operating environments in magnetic fusion devices become increasingly severe as the devices progress to the commercialization stage. Table VIII-3 identifies some of the major concerns for I&C developers encountered with the

environments associated with magnetic fusion devices. The present concerns for I&C developers are associated with high magnetic field intensities, time varying magnetic fields, electromagnetic radiation (microwaves, x-rays, and gamma rays), energetic particle fluxes (ions, electrons, neutral atoms), high vacuum and high temperature. Beginning with the next generation devices and upgrades (D-T and D-D operation), designers will have to deal with neutron fluxes, subsequent neutron activated structures, and tritium contamination. These additional complications compound rapidly since the I&C systems must now deal with additional shielding, tritium barriers, radiation damage, and remote maintenance. The I&C of remote maintenance systems becomes a separately, identifiable area, one which will not be addressed in this technical assessment.

Pulse length will steadily increase with each new generation of devices to the point where eventual steady-state operation is expected for a demonstration reactor. One of the benefits accompanying longer pulse lengths will be a decrease in cyclic operation. While cyclic fatigue will continue to be a concern, the associated problems will probably peak for a device such as ETR where the combination of total device lifetime and cyclic operation will be at its worst. With increased pulse length, greater scientific understanding, and evolution into an engineering feasibility mode, the requirements on instrumentation bandwidths may actually decrease, while control system requirements will greatly increase. The total number and diversity of instrumentation should hopefully level off or possibly decrease. Again, the most taxing requirements for the I&C systems may very well appear during the ETR phase.

Although the range of power densities on HHF components will probably not increase significantly (with the possible exception of compact reactor scenarios), the total usable power absorbed by these components (especially the first wall) will increase. This will introduce high-pressure, high-temperature water or other high-efficiency heat transfer media, such as liquid metals or high-temperature gases, which will force additional I&C requirements on the HHF component systems.

From Table VIII-3, it is clear that the I&C systems for HHF components share many (if not all) of the same environmental concerns of the plasma materials and plasma diagnostics disciplines. Since a great deal more effort has already gone into both of these two areas, a substantial amount of

Table VIII.3

HHF Component Environment Related Concerns for I&C Systems

Environment	Concern
High Heat Flux	High temperatures, high temperature gradients, high thermal stresses, thermocouple gradient errors
High Particle Flux (ions, neutrals, electrons)	Sputtering, charge accumulation, erosion with subsequent redeposition on instrumentation (especially optical elements), electrical interference
Neutron Flux	Radiation damage, activation of instrumentation, remote maintenance requirements, shielding, internal heat generation, decalibration
Electromagnetic Radiation (gammas, x-rays, microwaves)	Radiation damage, signal interference, false signal generation, decalibration, internal heating, shielding
High Magnetic Field (including transients)	Decalibration, $j \times B$ forces, induced EMF's, signal interference, eddy currents
Electromagnetic Induced Mechanical Loads	Positioning errors, vibration, mechanical fatigue, decalibration
Tritium	Contamination, β^- heating, T barrier design considerations
High Vacuum	Sealing considerations, pressure differential loads, material selection (outgassing)
High Pressure Water	Pressure loads, water contamination, water containment, erosion (high velocity water), conductivity (electrical)
Liquid Metals	Erosion (material compatibility), induced EMF's, conductivity (electrical), containment, flow measurement
Disruptions	High temperature, erosion, melting, redeposition on instruments, electromagnetic forces, electrical transients, thermal shock
High Voltages	Arcing, material selection, surface condition
Cyclic Operation	Fatigue life, transient measurements

information may be available for application to the I&C problems which arise as a consequence of having to perform in the fusion device environment. In particular, state-of-the-art data transmission techniques involving fiber optics, electrical isolation and signal discrimination in chaotic electromagnetic environments which are already in use for the plasma diagnostics should be adaptable for use in HHF component I&C systems. The extensive use of non-contact techniques can solve many of the problems associated with electromagnetic interference and electrical overload or probes and signal conditioning units due to device (plasma) electrical transients. Although the materials selection for instrumentation is often dictated apriori due to the functional requirements, information from the materials community could be very valuable when evaluating choices between types of a particular instrument (i.e., type K thermocouples are probably a much better choice than W-Re thermocouples in the presence of high neutron fluences) or developing new ones. Valuable information for use of instrumentation in the fusion device environment can also be gleaned from previous experience in the nuclear, space, defense, and high energy physics activities, where similar, although isolated, environments have been encountered. Proper use of these sources should minimize many of the requirements for development and in-device testing of HHF component I&C.

VIII.5 Instrumentation and Control Techniques

A variety of techniques already exist to measure most if not all of the parameters which have been defined in Section VIII.2. However, the ability of these techniques to perform in the fusion environment, provide data in the proper form or speed, and interact with control systems is the key to their applicability for HHF component use.

A summary of some of the more common measurement techniques for parameters mentioned in Section VIII.2 is given in Table VIII-2. Some of these are presently being used with success on operating fusion devices and are denoted by asterisks in Table VIII-1. Most of these will have to be upgraded or reconfigured to operate in a neutron environment. Others not yet used on fusion devices will require development to operate in the plasma environment and within the physical configuration limitations of the device. Table VIII-1 is not intended to represent a complete listing of all possible measurement

techniques or requirements. The actual techniques used will depend completely on the specific measurement required, the ultimate application of the measurement signal, and the constraints of the device on which it is to be used.

Temperature measurement in existing devices has been done primarily with thermocouples and RTD's. Although thermocouples will probably be used throughout most component and time frames, and should be, it may be difficult to operate RTD's successfully in a severe electromagnetic environment. Testing of thermocouples for calibration drift in 14 MeV neutron fluences should be carried out and probably can be done in conjunction with other neutron irradiation tests using the thermocouples as diagnostics. Infrared pyrometers are presently and probably will continue to be one of the most valuable diagnostics for HHF components. Although presently in use, upgrading to reduce or eliminate the impact of emissivity changes of the object (primarily due to surface condition changes, although wavelength and temperature dependence need to be addressed) and transmission/reflectance changes of optical components (i.e., redeposition coating of windows and radiation damage to windows/mirrors/photosensitive devices) is required. The use of fluor-optics should be explored since this technique is relatively immune to harsh electromagnetic environments, although limited in temperature range and probably difficult to locate physically. If gas coolants are used, acoustic and sonic thermometry or other methods common to HTGR technology might be employed.

Measurement of the mechanical properties of HHF components is seldom done in existing devices in-situ and may ultimately only be employed in off-line test facilities for use in the component development process. However, an increased use of optical techniques for strain/displacement measurement has occurred in the last decade. Some of those techniques such as optical moire methods⁽¹⁾, holography, and laser specklegrams may be applicable to diagnosis of HHF component mechanical response. Advantages of these techniques include the ability to observe 2-D strain patterns over the whole component and identify both in-plane and out-of-plane distortions. Being non-contact techniques, these methods have a good chance of successful operation in the harsh electromagnetic environments. Disadvantages include the significant amount of data analysis required to interpret results and the potentially extensive development effort required for application to in-situ measurement of HHF

components. Conventional strain gauges, crack propagation gauges, accelerometers, and LVDT's suffer from high susceptibility to electromagnetic effects, low survivability in high-temperature environments, and the limitation of their ability to measure only local conditions. Recently developed fiber optic strain sensors⁽²⁾ eliminate the electromagnetic noise problems but still have survivability problems and require predetermined location optimization. Most probably, these types of sensors will find application in off-line test facilities but may not be required in operating fusion devices.

The monitoring of surface conditions can be performed by some of the same methods described for the surface temperature and strain measurements. Infrared cameras can provide qualitative information on surface changes which induce temperature variations on thermally loaded components (i.e., wall thickness or coating/bond failure changes in actively cooled components). Moire and holographic techniques might be used to detect out-of-plane changes in surface due to erosion/redeposition processes. High-speed photographic techniques, time-lapse photography, and closed-circuit television monitors can be used to detect time-dependent surface conditions. Surface melting in a component might be detected through attenuation measurements on a diffracted x-ray beam or changes in the reflected intensity of light beams. Several techniques to monitor surface erosion might be explored, one of which would be the use of tracer materials⁽³⁾ or markers which would be pre-implanted in the HHF component surface and could be detected as they are released or exposed as a result of surface erosion. The change in frequency of quartz crystal oscillator circuits due to mass changes have often been used in off-line erosion/redeposition experiments, but may have limited application as an in-situ technique for fusion devices due to survivability, interpretability, and physical configuration issues. Laser absorption spectroscopy has been used successfully to measure erosion rates with simultaneous species identification but again may only be useful in off-line simulation experiments since the value of such localized measurements for interpretation of overall device performance is questionable. Finally, consideration should be given to utilizing surface analysis specimens which duplicate the HHF component surface and can be removed on a scheduled basis from the device environment for off-line surface analysis and metallography. Measurements of this type might provide a periodic monitor of how the actual component surface is performing.

The electric and magnetic parameters at or near the HHF components can be measured by techniques presently being used in other parts of the fusion devices. Source parameters can be identified through the use of gaussmeters, langmuir probes, and floating potential measurements of the electrically isolated component. Eddy current generation in armor and limiter tiles might be measured with pick-up coils mounted behind the tiles, although results may be difficult to interpret.

The coolant properties are and will continue to be measured by standard techniques used in present coolant loops such as thermocouples, flowmeters, and pressure gauges. Most of these can be located outside of the environmentally difficult regions of the device and do not appear to require significant redevelopment. For two-phase flow property measurements, a variety of techniques exist for measurement of void fractions and bubble dynamics⁽⁴⁾. Since these measurements will probably only be required during off-line development of HHF components, these probably do not merit redevelopment considerations. Coolant chemistry monitoring can assist in determination of internal wall erosion, tritium levels, and coolant quality; however, these again will probably be usable as they presently exist.

The vacuum and power measurements are also primarily standard items with little need for development for fusion device use other than the previously recommended radiation hardening and signal discrimination/isolation improvement efforts.

Control logic for HHF components will need to be generated for each device at some point in its conceptual design after the operational goals and some of the major subsystems have been reasonably well defined. At this point, it will be important to identify the reactor control systems so that they can be fully integrated into the total design. While existing devices have little or no control associated with HHF components other than the "interlock" type, this has been primarily related to a lack of need due to the short pulse lengths of these devices, as pointed out previously. However, these devices and their follow-ons are providing a very important data base for control algorithm generation. Basic control theory principles can be applied to the potential control loops identified in Table VIII-2, with algorithms and gross tuning constants drawn from this data base. Fine-tuning constants will probably have to be developed on the device itself until the

final commercialization stage where the control logic will have been sufficiently developed. Surely more control loops, other than those which appear in Table VIII-2, will have already been identified, and many of those identified will never find actual application. Most of the HHF component coolant system control techniques can be taken directly from off-line development facilities or existing fission reactor coolant control systems. If certain HHF coolant systems are not tied into the main power conversion cycle of the reactor, stand-alone process control by dedicated microprocessors may be used and developed entirely in off-line HHF component development facilities. Obviously, this would still require interfacing to the main reactor control system in an overall hierarchal control scheme.

In general, active plasma control involving fine adjustments of short time-scale plasma phenomenon (i.e., plasma instabilities) will probably be difficult with HHF components as sensors or actuators. The primary thermomechanical response sensors are limited in speed and precision by the time constants and sizes of the HHF components. As actuators, HHF components primarily rely on mechanical position and thermal properties which again infer long time constants. Much faster control is probably available for fine, real time plasma control with the plasma diagnostics as sensors, and field coils, power supplies, and other electrical systems as actuators. Longer term drift compensation of plasma parameters, gross device parameter adjustment, and interlock/alarm functions will probably constitute the major contributions of HHF components to the total reactor control scheme.

VIII.6 Conclusions and Recommendations

From the previous discussions, several conclusions can be drawn with respect to I&C development requirements for HHF components. A few general comments on the development requirements are made in the next few paragraphs, followed by a summary of the most critical issues in this area and recommendations on R&D programs which could be pursued to effect solutions to these issues.

A sufficient variety and quantity of I&C techniques are available for systems dealing with the HHF component heat sink thermomechanical and thermal-hydraulic aspects. These techniques are well proven and documented in the open literature. However, as with all I&C in a magnetic fusion device

environment, upgrading may be necessary to provide radiation hardened data acquisition and control systems and interfaces which can perform reliably in the severe electromagnetic and radiation fields of future devices. Instrumentation for the plasma-facing side of HNF components and claddings/coatings in duplex structures will be very important to component reliability diagnosis and could require significant development efforts.

In general, the feedback control of other device components or subsystems by HNF components is minimal and is not expected to change significantly in the future. This is primarily due to the relatively slow time constants (thermal and mechanical) of the HNF components with respect to other device systems (mainly electrical) or the existence of more efficient control techniques. Most of the control interfacing with other systems is and will be in the form of failure prediction or identification with the main control action being device shut down for component replacement or failure impact minimization.

Finally, the developers of the HNF components will naturally utilize a far greater number and variety of instrumentation techniques than will finally be used in device operation. While developers will require these diagnostics for performance evaluation and design iterations, sufficient "weeding out" should be performed prior to their use in the main device. Critical measurements (which are specific to the particular component design) should be identified by the HNF developers, and the final I&C should be developed with the final application environment in mind to minimize time-consuming design changes during installation or operation in the fusion device.

Three of the most critical issues which must be addressed in the development of I&C for HNF components are summarized below. Along with each critical issue are recommendations for general R&D efforts which could help provide timely solutions.

- Instrumentation of the "plasma side" of HNF components. This is an area in need of some development, primarily due to the severity of the environment in which this portion of the component must operate and be instrumented, and to the severity of the impact of any undesired performance or failure on the plasma and, hence, device operation. Non-contact, in-situ measurement techniques are desirable for instrumenting the plasma side of HNF components due to advantages

of operation in the presence of severe electromagnetic and radiation fields, and the response speeds for control loop application. Non-contact measurement techniques in general require more attention to allocation of viewports and space near the device. For this reason, these techniques will have to be developed early enough so that they can be included (or at least sufficient space and ports can be allocated) in the overall device designs. The associated R&D needs are: (1) Development of in-situ, real-time measurement techniques for erosion and redeposition rates and the spatial distribution of these rates. This development effort must address the integration of the measurement system with the operating fusion device. (2) Development of improved techniques for non-contact surface temperature measurements in the presence of a D-T plasma. This may involve upgrading of IR systems to reduce or eliminate the impact of emissivity changes of the object (primarily due to surface condition changes, although wavelength and temperature dependence need to be addressed) and transmission/reflectance changes of optical components (i.e., redeposition coating of windows and radiation damage to windows/mirrors/photosensitive devices. This development effort must also address full integration with the intended fusion device.

- Definition of control loops and algorithms involving HHF components. Control logic for HHF components needs to be generated for each device during its conceptual design. Control loops and algorithms are not presently well defined for future devices at any development stage. It is important to identify the reactor control systems so that they can be fully integrated into the total design. If engineering data for control algorithm generation is not taken on devices like TFTR, the lack of such data could become a critical problem. The associated R&D programs should provide the following: (1) The ability to test HHF component control schemes should be included in future device designs so that some of these techniques can be incorporated into the standard control logic of devices such as ETR and DEMO. This should include both control of HHF component parameters and contributions of HHF components to the total reactor control scheme, which involve long-term drift compensation of plasma parameters, gross device parameter adjustment, and interlock/alarm

functions. (2) Development of control algorithms for device subsystem control. These algorithms rely heavily on results from existing devices. Existing devices and their follow-ons should include tests which provide an engineering data base for control algorithm generation.

- Radiation hardening of I&C systems. Upgrading is necessary to provide radiation hardened data acquisition and control systems and interfaces which can perform reliably in the severe electromagnetic and radiation fields of future devices. Such items as optical windows, lenses and mirrors, fiber optics, and electronics are particularly susceptible to radiation environments, and existing devices are not generally designed for operation in high neutron fluences. To ensure successful development of such radiation-hardened systems, programs such as the DOE-funded, Radiation Hardening of Fusion Diagnostics Program⁽⁵⁾, should be continued, expanded upon, or others initiated, if necessary, so that a program exists which will: (1) Define and perform experiments which will provide a data base for radiation effects (fusion-specific radiation environment) on HNF instrumentation components. (2) Place more emphasis on device designs which address I&C radiation hardening (includes I&C design, placement, and shielding).

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