# HEAT TRANSFER AND MECHANICAL INTERACTIONS IN FUSION NUCLEAR SYSTEMS*


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## ABSTRACT

This general review of design issues in heat transfer and mechanical interactions of the first wall, blanket and shield systems of tokamak and mirror fusion reactors begins with a brief introduction to fusion nuclear systems. The design issues are summarized in tables and the following examples are described to illustrate these concerns: the surface heating of limiters, heat transfer from solid breeders, MHD effects in liquid metal blankets, mechanical loads from electromagnetic transients and remote maintenance.


## INTRODUCTION

Current fusion devices advance our understanding of plasma physics but we have not yet progressed sufficiently to confine a D-T fusion plasma for the long burn times anticipated in commercial fusion reactors. The sustained heat and particle fluxes from a burning D-T plasma, intense neutron radiation, need for self-sufficient breeding of tritium (fuel) as well as the production of electrical power are features of advanced fusion reactors that present new and challenging design requirements.

Major design studies of fusion reactors have provided much of our collective insight into the progress in supporting technologies that will be needed to realize the goal of commercial fusion power. The examples of components and engineering issues in this paper draw extensively upon the following design studies: STARFIRE (1,2), MARS (3-5), DEMO (6,7), FED/INTOR (8,9) and BCSS (10,11) (Blanket Comparison and Selection Study).

For readers seeking more detail, two excellent general review articles (12,13) are recommended. The evolution of design concepts, engineering issues and research programs related to first wall, blanket and shield components have also been recently reviewed. (14) More design specific information such as design analyses for specific components may be found in topical conferences on fusion technology. (15-17)

## BASIC ELEMENTS IN FUSION TECHNOLOGY

Together the first wall, blanket and shield provide a nuclear envelope that isolates the fusion plasma and the energetic particles produced by the plasma from the rest of the fusion reactor. Figure 1 illustrates these components in a magnetically-confined fusion reactor based on the reaction of deuterium and tritium (D-T).

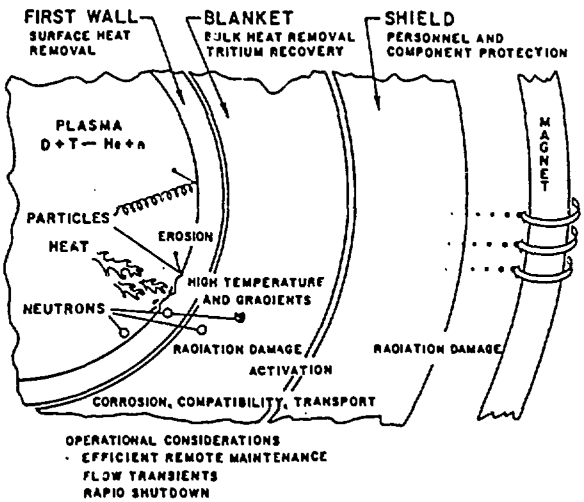


Figure 1. Functions and environment of the first wall, blanket and shield of a fusion reactor.

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$$D + T \rightarrow {}^4He\ (3.5\ MeV) + n\ (14.1\ MeV) \qquad (1)$$

First wall components will directly face the plasma and be subjected to a severe environment including intense electromagnetic radiation and bombardment by high energy neutrons and by energetic particles from the plasma. The primary function of the first wall is to maintain the physical boundary that defines the plasma chamber without adversely affecting the plasma by introducing impurities.

Rejection of the intense surface heat loads is an important requirement in tokamaks, particularly for specialized first wall components, such as limiters, armor and divertor plates, designed to protect the first wall by preferentially intercepting plasma particles. (Mirror devices have much lower first wall heating.) In a fusion reactor that produces 3000 MW of power from the DT fusion reaction, the first wall components will receive about 600 MW as surface heat.

In most conceptual designs for power-producing fusion reactors, process heat is extracted from both the first wall and the blanket. Therefore, the coolant(s) for these components must reach temperatures useful for power production. About 320°C is generally held to be a lower bound for supply to steam cycles.

The blanket has two functions, collecting heat and producing tritium to fuel the reactor, and must also provide for their extraction from the blanket. Heat generation in the blanket is dominated by the nuclear reaction through which $^6Li$ transmutes into tritium and helium.

$${}^6Li + n \rightarrow T + {}^4He \qquad (+4.8\ MeV) \quad (2)$$

$${}^7Li + n \rightarrow T + {}^4He + n \qquad (-2.5\ MeV) \quad (3)$$

Both solid and liquid lithium-bearing materials have been utilized in reactor design. Solid breeders are ceramics such as $Li_2O$, $LiAlO_2$ and $Li_8AlO_6$. The liquid metals Li and 17Li-83Pb are prime candidates for lithium-metal blankets. Because of neutron multiplication by the lead, the seemingly small fraction of lithium in 17Li-83Pb produces sufficient tritium when enriched in $^6Li$, which has a larger cross section than $^7Li$.

The shield provides biological protection for personnel and reduces radiation and nuclear heating to levels acceptable for the operation of sensitive components such as superconducting magnets. Some of the primary shielding in most reactor designs is readily moveable to permit remote maintenance, assembly and repair.

Our projections of "commercial fusion technology" are embodied in designs such as STARFIRE and MARS (Mirror Advanced Reactor Study); these are the most comprehensive design studies to date respectively for tokamak and tandem mirror reactors. Figures 2 and 3 show representative configurations for the "donut-shaped" tokamak and the linear-type tandem mirror reactor. Table 1 gives some parameters from these designs and from DEMO, a "next generation" tokamak that would precede a commercial reactor.

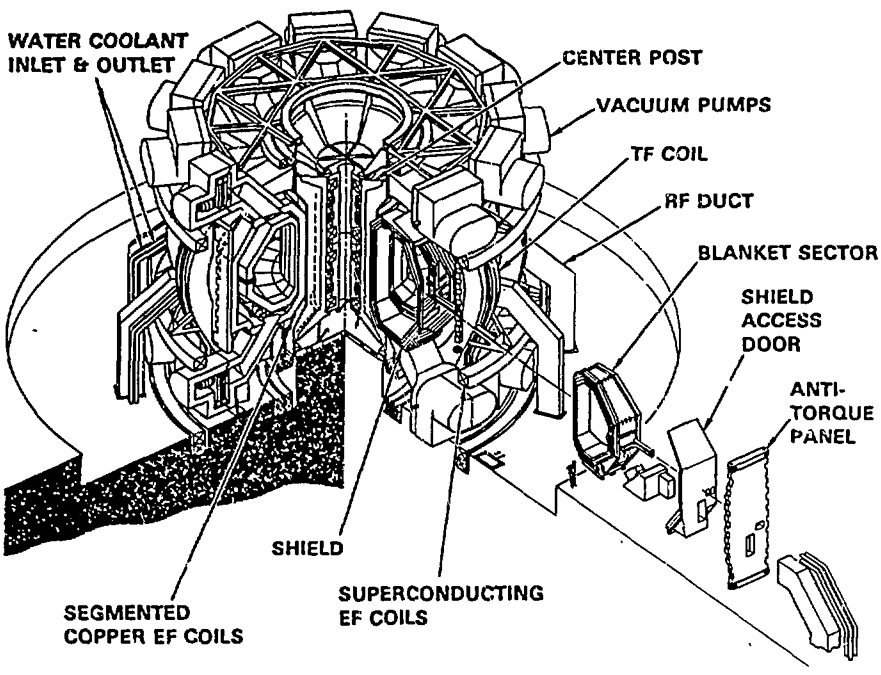


Figure 2. STARFIRE - A concept for a commercial tokamak power reactor prepared in major design study in 1979-1980. Cutaway view shows principal components. First wall and blanket components are shown in more detail later in Figure 7.

TABLE 1. PARAMETERS FOR STARFIRE, DEMO AND MARS

| | STARFIRE | DEMO | MARS (final) |
|---|---|---|---|
| Fusion power (MW) | 3510 | 1069 | 2600 |
| Thermal power (MW) | ~ 4000 [a] | ~ 1300 [b] | ~ 3100 [c] |
| Neutron wall load ($MW/m^2$) | 3.6 | 2.1 | 4.3 |
| First wall heat load ($MW/m^2$) | 0.9 | 0.25 | 0.06 |
| Blanket power multiplication factor | 1.14 | 1.26 | 1.36 |
| Thermal power in first wall/blanket (MW) | 3866 | 1176 | 2820 [c] |
| Power to limiter/end dump (MW) | 200 | 151 | 573 [c] |
| First wall material | Be/SS | Be/SS | HT-9 |
| Limiter/end dump material | Be/Cu or V [d] | Be/Cu or V [d] | V/Cu [d] |
| Neutron multiplier | $Zr_5Pb_3$ (or Be) | None | None (Pb in Li-Pb) |
| Breeding material | $LiAlO_2$ (90% $^6Li$) | $Li_2O$ (natural) | 17Li83Pb (90% $^6Li$) |
| Blanket structure | SS | SS | HT-9 |
| Blanket coolant | $H_2O$ | $H_2O$ | 17Li83Pb (90% $^6Li$) |

[a] Includes 200 MW of low grade power from the limiter which is used as feedwater heating.

[b] Power from limiter is not used in the energy conversion cycle.

[c] Blanket power includes 730 MW from reflector. Thermal power also includes 281 MW of thermal power from plasma end dump. 292 MW from direct conversion is not included.

[d] Vanadium here includes vanadium base alloys such as V-15Cr-5Ti.

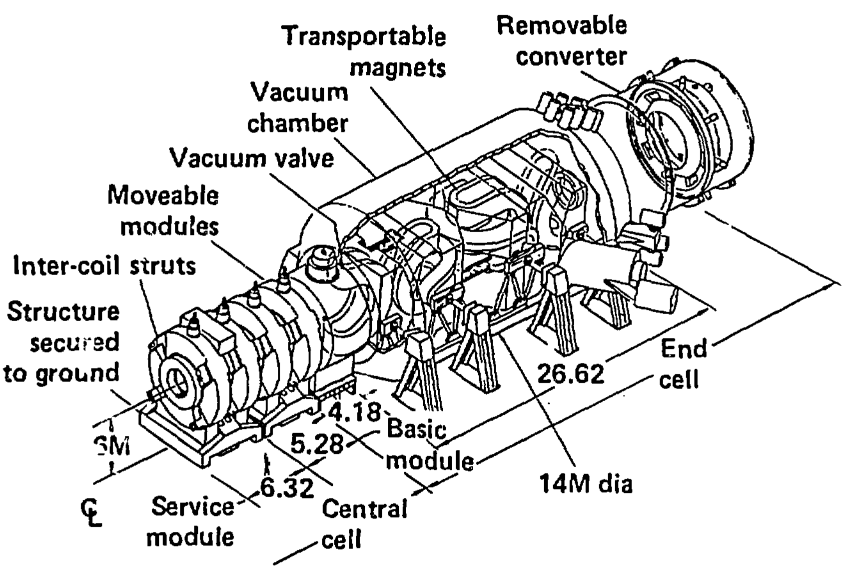


Figure 3. MARS - A concept for a commercial tandem mirror reactor prepared in the Mirror Advanced Reactor Study (MARS) in 1982-83. Cutaway shows principal components of the end portion of the reactor. Only three of the 84 modules in the central cell are shown. Details of module are shown later in Figure 8.

## HEAT TRANSFER ISSUES

The primary source of heat in a DT fusion reactor is the fusion reaction which releases 80% of the fusion power as 14.1 MeV neutrons and the remainder as 3.5 MeV alpha particles. Additional sources of sensible heat are the energy multiplication in the blanket from the transmutation of $^{6}Li$ and power supplied to the plasma by RF systems or neutral beams. Figure 4 shows the power cycle for DEMO.

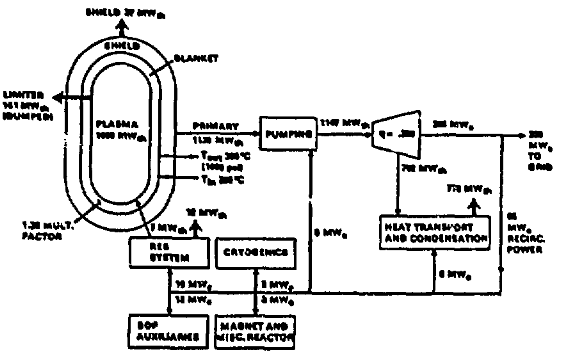


Figure 4. Power flow diagram for the DEMO reactor. Transmutations in blanket add to fusion power (1069) MW to produce 1138 MW of primary thermal power.

Heat is transferred to the reactor components in the following ways. The energetic alpha particles born in the plasma are ions and tend to be confined by the magnetic field and to exchange energy with the plasma particles. This "alpha power" plus power supplied to the plasma by RF or neutral beams reaches various parts of the vacuum vessel either through bombardment by particles or by (electromagnetic) radiation, primarily from impurities.

The energetic neutrons penetrate the first wall and are absorbed primarily by the blanket. Significant nuclear heating also occurs in the first wall and its superstructure such as a limiter, armor or divertor.

Table 2 lists various design issues associated with heat transfer in fusion nuclear systems. Several examples from this list are described below beginning with those components that interact directly with the plasma.

### Limiters

In tokamak reactors, the imperfect magnetic confinement allows particles from the plasma to escape radially outward. A limiter or divertor forms a partial aperture that defines the edge of the plasma and serves the dual functions of protecting the first wall and of controlling impurities. Figure 5 shows a section of the limiter from the DEMO design. The curved shape follows the edge of the plasma and spreads out the intense heat load on the surface of the limiter. Vacuum pumping ports connected to the limiter (or divertor) collect neutral particles.

The limiter and the collector plate for a divertor are specialized heat sinks. Most reactor designs do not attempt to recover this power for the primary cycle because the implied requirement for a coolant at high pressures and high temperatures would further compound design objectives that are already difficult.

The constraints on selection of materials are also severe. Radiation losses that would sap energy from the plasma tend to increase with atomic number. Only very light elements ("low Z") will be fully ionized in the plasma. The practical choices for plasma-side structural materials in an advanced reactor appear at this time to be limited to beryllium or graphite unless the edge temperature of the plasma can be kept quite low (<50 eV) so that low-sputtering refractories (e.g. W) could be used.

Designs for limiters, armor and first walls in tokamaks have typically utilized a composite structure with low-Z tiles or cladding on a structural substrate. An extensive investigation of the design options and constraints on plasma-side materials was performed for INTOR (International Tokamak Reactor) (18,19), a design study with international collaboration.

Severe neutron radiation and loss of material through erosion are additional considerations for plasma-side materials in tokamaks. For losses due to erosion, allowances in increased thickness of plasma-side materials of up to 1 cm are used in current design studies. Steady-state erosion (and redeposition) is anticipated from sputtering by neutrals and ions from the plasma. Vaporization and melting can

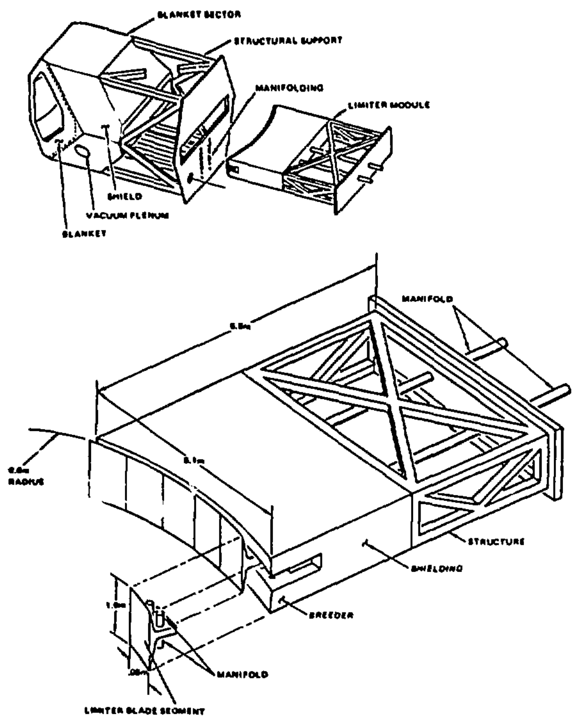


Figure 5. Limiter - Position of the limiter (for the DEMO design) is shown in elevation view. Modular design permits removal for remote maintenance.

Table 2. Summary of Heat Transfer Issues in Fusion Nuclear Systems

| Component/Critical Feature | Major Constraints | Other Factors | Typical Power Deposition |
|---|---|---|---|
| **A. Heat Sinks/Superstructure** | | | |
| Limiter: combined heat load and particle flux | 1. thickness (erosion by plasma)<br>2. low reflux of impurities (low Z or cool plasma edge)<br>3. integrity of composites/cladding | PMI*, N*, V*, M*<br>PMI, N, V, M | 4-13 $MW/m^2$ peak BCSS**<br>2.4 $MW/m^2$ peak DEMO, INTOR |
| Collector (Divertor): combined heat load and particle flux | 1. thickness (erosion by plasma)<br>2. integrity of composites/cladding | PMI, V | 3-5 $MW/m^2$ peak |
| Direct Convertor Bins: combined heat load and particle flux | 1. thickness (blistering, exfoliation)<br>2. high voltage standoff<br>3. integrity of composites/cladding | PMI, V | 7.6 $MW/m^2$ peak MARS |
| Armor: combined heat load and particle flux | 1. thickness (erosion by plasma)<br>2. low reflux of impurities (low Z or cool plasma edge)<br>3. method of attachment | PMI, N, V, M | |
| **B. First Wall** | | PMI, N, V, M | .5.1 $MW/m^2$ (tokamak) BCSS<br>.25 $MW/m^2$ (tokamak) DEMO<br>.06 $MW/m^2$ (mirror) MARS |
| Water Coolant: surface heat transfer | 1. thickness (pressure)<br>2. tube reliability | | |
| Liquid-metal Coolant: surface heat transfer | 1. liquid metal MHD effects<br>2. temperature limit (corrosion) | | |
| Helium Coolant: surface heat transfer | 1. thickness (pressure)<br>2. high temperature coolant<br>3. void volume | | |
| **C. Neutron Multiplier** | | N, V, M | 29 $W/cm^3$ peak STARFIRE |
| Be Multiplier: bulk heat removal | 1. resources<br>2. joining/fabrication<br>3. severe radiation damage (ductility)<br>4. helium production and release | | |
| **D. Blanket** | | | |
| Solid Breeders - monolith: interface heat transfer | 1. temperature range for breeder<br>2. control of breeder/coolant-tube contact | N, V, M | 40-85 $W/cm^3$ peak $LiAlO_2$ STARFIRE<br>~35 $W/cm^3$ peak SS STARFIRE<br>170 $W/cm^3$ peak $LiAlO_2$ BCSS<br>13.4 $W/cm^3$ peak $LiAlO_2$ DEMO |
| Solid Breeders - sphere-pac: effective conductivity | 1. temperature range for breeder | N, V, M | |
| interface heat transfer | 1. temperature range for breeder | | |
| Liquid Metal Blankets: internal heat transfer | 1. liquid metal MHD effects<br>2. corrosion | N, V, M | 50 $W/cm^3$ peak in SS BCSS<br>11.5 $W/cm^3$ peak in HT-9 DEMO<br>20 $W/cm^3$ peak in V BCSS<br>26 $W/cm^3$ peak in Li BCSS<br>22.2 $W/cm^2$ peak in 17Li-83Pb DEMO<br>36 $W/cm^3$ peak in 17Li-83Pb MARS |
| Molten Salt Coolant: stability | 1. peak temperature<br>2. MHD induced breakdown | N, V, M | |
| **E. Shield** | | | |
| heat removal | 1. large volume | N, V, M | |

* other factors
PMI — plasma materials interaction, e.g. sputtering and vaporization from disruptions

N — neutron radiation, specifically activation and severe radiation damage in structural materials

V — vacuum, including both plasma side vacuum and integrity of boundary with reactor environment

M — magnetic field

** Typical parameter values were taken from the following design studies. Many parameters scale with neutron wall loading (power density).

| Design Source | Average Neutron Wall Load |
|---|---|
| BCSS — Blanket Comparison and Selection Study (1984) | 5 $MW/m^2$ |
| MARS — Mirror Advanced Reactor Study (1983) | 4.3 $MW/m^2$ |
| DEMO — Demonstration Tokamak Reactor (1982) | 2.1 $MW/m^2$ |
| INTOR — International Tokamak Reactor (1982) | 1.3 $MW/m^2$ |
| STARFIRE — A Commercial Tokamak Fusion Power Plant Study (1980) | 3.6 $MW/m^2$ |

result from plasma disruptions. (Plasma disruptions are unique to toroidal devices and involve gross motion of the plasma after loss of confinement.)

Although the basic physical principles are understood we do not yet have the ability to predicate local erosion rates with much accuracy. During steady-state operation the distributions of energy for the impinging particles, the amount of redeposition and the integrity of the redeposited material are poorly characterized variables. For disruptions, the time period of the disruption and the degree of localization of the energy deposition are not well known. In a parametric analysis on disruptions for INTOR, energy densities of 270-1070 $J/cm^2$ (on the limiter) and times of 20 ms and 5 ms were used in the case where the limiter receives most of the heat load. In the case labeled "reasonable expectation" these energy densities were reduced by 50%.

The heat loads from plasma disruptions will vaporize material and cause surface melting and the decay of the large current (several megamps) in the plasma gives rise to eddy currents in the structure. The motion of a melted layer of material subjected to currents in the presence of a magnetic field during the short time of a disruption represents another uncertainty in the allowance for loss of plasma-side materials.

## Direct Converters

In mirror devices, there is no equivalent to the limiter located in the central cell. The effective radial confinement is much greater than in tokamaks and the thermal load on the first wall is expected to be much lower in mirrors than tokamaks. However, particles tend to escape confinement by leaking out the ends of the central cell. The end losses are deemed large enough in current designs (e.g. MARS) to necessitate the use of direct convertors to tap this source of otherwise lost power.

The direct convertor grids are the primary targets for bombardment by energetic ions from the plasma. The heat loads are higher though roughly similar to those for limiters. The energies of the bombarding ions are fairly high (~50-950 keV plus some prompt alphas at ~3.5 MeV) because of the nature of the magnetic and electrostatic barriers at the ends of the central cell that help contain the plasma. At high energies the ions tend to penetrate and sputtering losses are low but there is concern about subsurface damage and the possibility of spallation.

The direct convertors are located far from the central cell so neutron radiation damage is not a problem as it is for limiters (in tokamaks) and extracting the thermal power from the direct convertor with high pressure water coolant as well as collecting the electrical current of the ions has been suggested.

In the MARS design, 292 MW of electrical power is recovered directly in the direct convertors and 96 MW of electrical power is recovered from the 281 MW of thermal energy carried by the ions and electrons. High pressure water is used as a coolant. The total electrical power of 388 MW recovered from the direct convertor is 14% of the gross electrical power of 1500 MW.

The characterization of the plasma edge and its interactions with materials is the mission of the Plasma Materials Interaction Task in the Fusion Reactor Materials Program. Experimental development of plasma-side materials and prototype components is carried out under the High Heat Flux Components Development Program. These and other experimental programs mentioned later are funded through the Development and Technology Division of the U.S. DOE Office of Fusion Energy.

## Blankets - Solid Breeders

The blanket collects most of the heat in the reactor. The possibilities for blanket concepts are numerous but can generally be classified with the following two elements: a) solid or liquid breeder and b) static or circulated breeder. The STARFIRE blanket (Figure 6) and the lithium-lead blanket used in MARS (Figure 7) are respective examples of a static solid breeder and a circulating liquid-metal breeder. Designs with static (or slowly circulated) liquid metal cooled by molten salt have also been proposed as have solid breeder designs where the breeder is circulated in a fluidized bed.

A succession of design studies has clarified the technological issues for many blanket concepts as well as for other components. Until recently there has been little consistency in the treatment of design issues from one study to the next because the levels of effort in these studies has varied considerably and each study focused on a single or perhaps two blanket concepts. During 1983-84 a systemmatic and comprehensive comparison of candidate blanket systems for fusion reactors, the Blanket Comparison and Selection Study (BCSS), analyzed many blanket designs and identified the more promising candidates. A significant but not unexpected result was the discovery of potentially problematic engineering issues for all design concepts.

For solid breeder blankets the primary issue in heat transfer is the uncertainty in the heat transfer coefficient at the interface between the breeder and the coolant pipe or breeder cladding.

The temperature at each point in the solid breeder will increase or decrease directly with any incremental change in the temperature difference across the interface between the breeder and coolant pipe. A typical value for the heat flux is 65 $W/cm^2$ and estimated heat transfer coefficients are in the range of 0.5-1 $W/cm^2{}^\circ C$ so temperature differentials across the gap of roughly 100°C are anticipated.

The heat transfer coefficient is a critical parameter because there is a temperature window for acceptable performance of solid breeders. For example, the respective guidelines for $Li_2O$ and $LiAlO_2$ adopted in the BCSS are 410-800°C and 350-1000°C.

Control of the peak temperature of the breeder is important for maintaining long-term stability against closing of porosity by grain growth or mass transfer. Designs for solid breeders have some form of controlled porosity to facilitate migration of tritium to the purge flow system where it is transported for recovery.

There is usually also a low temperature limit set by the desirability to maintain the inventory of tritium in the blanket below some set level (e.g., 10 kg) and, in the case of $Li_2O$, to avoid possible formation of (corrosive) LiOH. The tritium inventory in solid breeders arises because even with negligible solubility a concentration gradient is needed to force tritium to diffuse out of the solid breeder to the purge stream. The "diffusive inventory" of tritium in

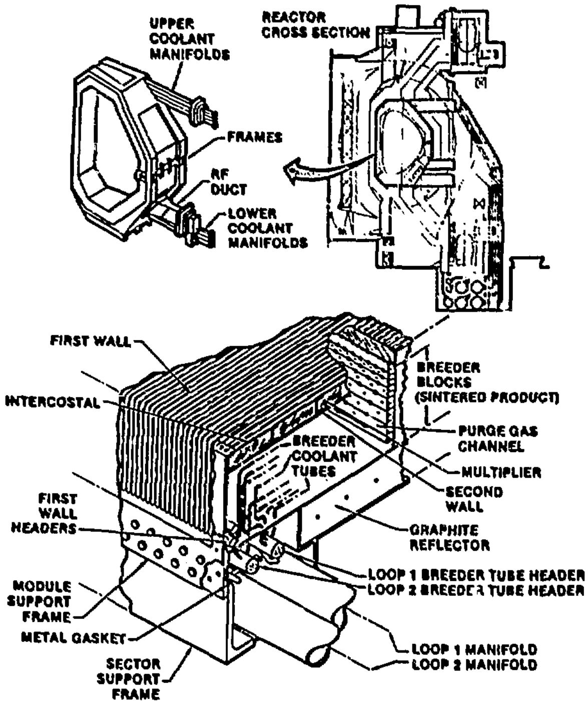


Figure 6. STARFIRE Blanket--First wall, neutron multiplier and water-cooled solid breeder ($LiAlO_2$) blanket are integral assembly. Sector is also shown in Figure 1.

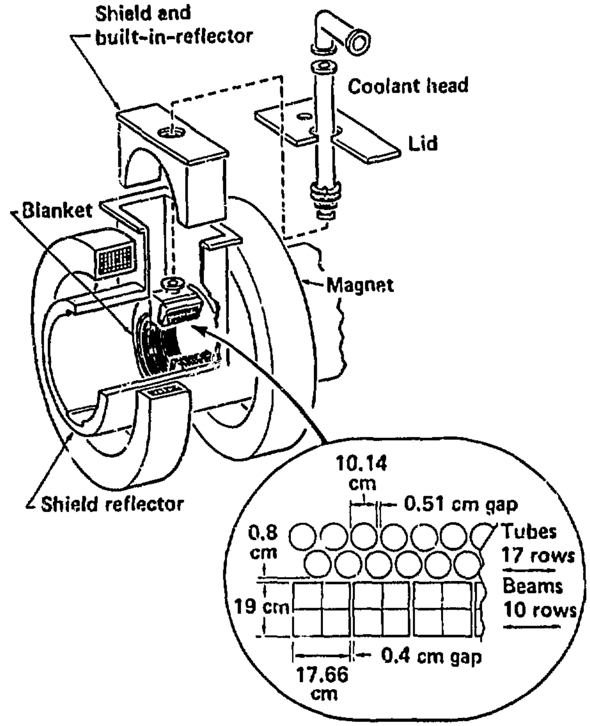


Figure 7. MARS Blanket--Liquid metal 17Li-83Pb flows from top to bottom through circular and square ducts. First wall is simply surface of inner row of circular ducts. Modules are also shown in Figure 2.

solid breeders varies strongly with temperature. For a drop in the minimum temperature of $LiAlO_2$ from 450°C to 400°C the inventory in a STARFIRE-type blanket would increase by a factor of six.*

For the STARFIRE blanket, the current concensus among designers has been that the "as-assembled" contact between the breeder and the coolant tubes would probably not provide well controlled heat transfer. There has also been concern about the reliability of the single-walled tube design. A recently suggested solution is double-walled tubes and "sphere-pac" form for the solid breeder.

Sphere-pac is a mix of spherical particles of three sizes that gives fairly dense packing and good thermomechanical performance and has proven successful in the fission industry. A typical mixture of particle sizes for sphere-pac in (fission) fuel is 58% with 1200 micron diameter, 20% with 300 micron diameter and 22% with 40 micron diameter. The expected packing density based on experience with fabrication of sphere-pac fuel in the fission industry is 85-88%. Sphere-pac for fusion blankets offers the following potential benefits: ease of blanket assembly; all void volume as interconnected porosity for tritium migration; adequate heat transfer; and capability to accommodate swelling and mechanical loads.

*Based on data generated for the BCSS (11).

In another design approach, helium is used as the coolant with the breeder clad in plate-like cans. The high pressure of the helium coolant forces the cladding tight against the breeder to control and to improve heat transfer.

Blankets-Liquid Metal

In self-cooled liquid metal blankets, the liquid metal serves as both the breeder and the coolant. This arrangement simplifies many considerations regarding both materials and design since the blanket requires only structure plus the breeder-coolant. Liquid metals have good heat transfer characteristics with high thermal conductivities and heat capacities. The design constraints are driven primarily by magneto-hydrodynamic (MHD) effects on flow that severly restrict cooling of the first wall.

The Blanket Comparison and Selection Study (BCSS) has highlighted two important problems related to MHD effects. The first problem is the suppression of turbulence which drastically reduces heat transfer from the first wall. In the presence of a strong magnetic field perpendicular to the flow of liquid metal, as in a fusion reactor, the flow is lamellar and the flow elements adjacent to the first wall will remain adjacent to the first wall and be heated along its length. For the assumption of slug flow, the heat penetration into the liquid metal during its passage along the first wall is basically the same as a solid metal heated from one surface for the same time interval. The engineering problem is that flow rates consistent with bulk heat removal for the blanket may produce excessive temperatures at the first wall.

The second problem is the tradeoff between stress limits in the blanket structure and temperature limits based on allowances for corrosion. Coolant pressure and stress generally increase in proportion to the coolant flow rate. Although corrosion generally increases with flow rate, the corrosion rates are much more sensitive to the temperature of the structural material which decreases with increasing flow rate.

The competing design requirements for limits on temperature and stress present the following dilemma. Structural temperatures could be reduced by increasing the flow rates but the higher velocities would also increase the pressure drop due to MHD effects. Increasing the thickness of the structure to reduce this stress is an ineffective solution because the increased electrical conductance of the thicker structure exascerbates the MHD problem. (Also the neutronic performance is degraded by a thicker structure.)

The higher fields and longer flow lengths for the inboard portions of blankets in tokamaks make design requirements for flowing liquid metal blankets significantly more difficult than for mirror reactors. The simple flow-through style of the MARS blanket (Figure 7) cannot be directly adapted for tokamaks.

A more complicated flow path that combines fast flow in small first wall passages parallel to the toroidal magnetic field with lower flow rates in most of the thickness of the blanket was developed in the BCSS as one potential solution for a tokamak liquid metal blanket. Figure 8 shows this design.

A major obstacle to further design development is lack of data. The analyses to date are based on correlations that may not accurately model flow in the regime of high magnetic interaction parameters of $10^3$–$10^5$ appropriate for fusion. (The magnetic interaction parameter is equal to the Hartmann number squared divided by Reynold's number and is the ratio of magnetic to inertial forces.) Data with any relevance is very scarce and limited to interaction parameters in the range of 10–90. In 1984 the Blanket Technology Program began an analytical and experimental task to supply needed data on liquid metal MHD behavior at high interaction parameters.

MECHANICAL INTERACTIONS

The sources for various steady-state and transient mechanical loads in fusion nuclear systems are summarized in Table 3. Thermal and hydraulic stresses and radiation damage are important factors for plasma-interactive components and for the blanket. Two other important considerations that have significant impact on mechanical interactions are (1) the need for remote disassembly, reassembly and repair of fusion nuclear systems and (2) responses of the structure to electromagnetic transients. Both affect mechanical interactions through their global effects on reactor design. In addition, the eddy currents produced in electromagnetic transients, for example plasma disruptions in tokamaks or the rapid loss of plasma in mirrors, are direct sources of significant body forces on components.

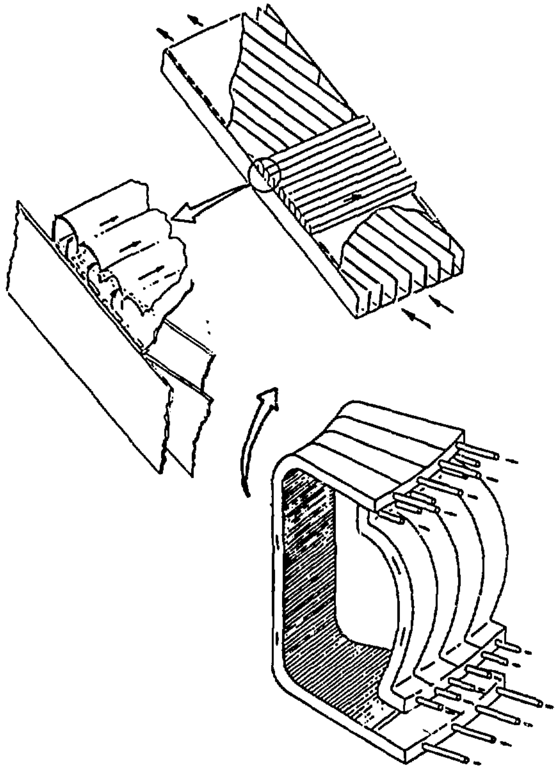

Figure 8. Liquid metal blanket for tokamak—the combination of fast toroidal flow at the first wall and slow poloidal flow (perpendicular to toroidal field) permits more rapid coolant flow past first wall while maintaining lower bulk velocity in the large ducts to reduce MHD pressure drop. The design was developed in the Blanket Comparison and Selection Study.

Remote Maintenance

Components in fusion reactors (first wall, blanket, shield, etc.) will become activated either directly from neutron radiation or indirectly from tritium or activated corrosion products transported by the coolant. Consideration of remote maintenance and repair shapes the general design philosophy as well as the detailed designs of components. The range of operations to be performed remotely is large, from the lifting and transport of reactor segments weighing perhaps hundreds of tons to the delicate and precise work of coupling electrical leads or locating vacuum leaks.

The different types of hardware, notably the preponderance of electrical equipment and the somewhat complicated configurations, make the requirements for remote handling for fusion plants more vast than for fission plants. One obvious difference is in the size and weight of components. For example, the estimated

Table 3. Summary of Mechanical Interactions in Fusion Nuclear Components

| Effect | Source | Limiter, Armor, Divertor | Direct Convertor | First Wall | Blanket/Breeder Solid | Blanket/Breeder Liquid | Shield |
|---|---|---|---|---|---|---|---|
| thermal stress | - surface heat flux (plasma) | T* | M* | T,M(1) | - | - | - |
| | - bulk heating (neutrons) | T | - | T,M | T,M | T,M | T,M |
| | - mismatch clad/substrate | T | M | T | - | - | - |
| | - mismatch breeder/cladding steady-state | - | - | - | T,M | - | - |
| increased stress | - steady-state erosion by plasma (reduced thickness) | T(2) | M | T(2),M | - | - | - |
| | - ablation/melting from plasma disruptions | T | - | T | - | - | - |
| | - crack growth | T | M | T,M | T,M | T,M | T,M |
| hydraulic stress | - high pressure coolant including liquid metal MHD effects | T | M(2) | T,M | T,M | T,M | - |
| stress/distortion | - differential swelling | T | - | T,M | T,M | T,M | - |
| | - irradiation creep | T | - | T,M | T,M | T,M | - |
| changes in thermal & mechanical properties | - radiation damage | T | M(3) | T,M | T,M | T,M | - |
| body forces/torques | - eddy currents from disruptions or loss of plasma | T | - | T,M | T,M | T,M | - |
| | - magnetic loads on ferromagnetic materials | - | - | T,M | T,M | T,M | - |
| | - seismic loads | T | M | T,M | T,M | T,M | T,M |
| fatigue, crack growth | - cyclic operation | T(1) | - | T(1) | T(1) | T(1) | - |

* Notes: T = Tokamak type reactor.
T(1) = Existing tokamaks operate in cyclic mode but pseudo-steady-state is anticipated in reactors.
T(2) = Erosion is most severe on limiters but also occurs at the first walls of tokamaks and mirrors.
M = Tandem mirror type reactor.
M(1) = First wall heat flux will be lower in mirrors than in tokamaks.
M(2) = Extraction of thermal power (with pressurized water) as well as direct current has been considered.
M(3) = Ion bombardment may produce exfoliation.

weight of one sector of the blanket in STARFIRE is 65 tons and the shield door is 179 tons. For handling large components the most directly applicable technology from the fission industry probably lies in fuel reprocessing plants and in the old NIRVA (nuclear rocket) program.

Design studies have been the forum for improvements stemming from concerns about remote maintenance and particularly from the need to provide access to the blanket, first wall and components such as limiters and armor that may be more subject to damage. Examples of design decisions based on concerns for remote maintenance are: the relocation of the poloidal (usually resistive) coils outside the toroidal (usually superconducting) coils, the relocation for major joints of the vacuum boundary away from the first wall and the segmentation and modularization of fusion nucler components for removal between the toroidal field coils. The modularization in STARFIRE is emphasized in Figure 2.

In mirrors the modularization in the central cell can be accomplished either by removing the magnet and blanket/shield segment as a unit, or by making smaller solenoidal segments that could be removed between the coils. The latter approach was adopted for MARS. The blanket modules shown in Figure 7 are designed for axial translation to a nearby "service module" where the blanket can be lifted out between the coils. Twelve of the 84 modules in the MARS central cell are the slightly wider service modules.

The limiter design shown in Figure 5 also has a modular style. The complete limiter runs toroidally around the mid-plane of the DEMO plasma chamber and is composed of a eight modules that can be removed or inserted, much like drawers, into openings between the toroidal field coils. The cantilevered structure necessary for the limiter configuration is susceptible to torques and care must be taken in the design to minimize the radial components of eddy currents produced during plasma disuptions.

Among the many challenges in remote maintenance for fusion are: (1) the mobilization and transport of large, heavy sections weighing several hundred tons; (2) routine inspection of the plasma chamber for damage; (3) replacement of minor elements such as fasteners or individual armor tiles in the plasma chamber; (4) location of leaks after their detection; (5) coupling and uncoupling of large pipes; and (6) breaking and sealing of an assortment of ports, many of which are large and non-circular. Also, the difficulty and expense implied in making repairs to components in the nuclear system of a fusion reactor forces a high value on reliability.

As one example from this list, let us briefly examine the number of ports that are associated with the first wall, blanket and shield systems for fusion reactors. Table 4 shows estimates based on the conceptual designs for the Fusion Engineering Device (FED - a tokamak), STARFIRE and the Tandem Mirror Reactor (TMR). The estimates are for disconnectable joint systems external to the vacuum vessel and specifically for the first wall, blanket and shield systems. The numbers are incomplete and thus probably underestimated. The large number of such joint systems is impressive. Manipulating even a small fraction of these joint systems during planned or unplanned reactor outages has the obvious potential impact of extremely long times and difficult if not impossible tasks unless these systems are designed and the hardware developed to insure adequate remote maintenance.

Table 4.* FUSION REACTOR JOINT SYSTEMS

| | Number of Joint Systems | | |
|---|---|---|---|
| | Vacuum | Coolant | Other |
| FED (FEDC, 6/81) | 138 | 376 | 208 |
| STARFIRE (ANL, 9/80) | 228 | 404 | 246 |
| TMR (LLNL, 9/79) | ? | 688 | 554 |

*Courtesy of McDonnell-Douglas Astronautics Corp.

## Electromagnetic Transients

In tokamaks, the segmentation of the first wall and blanket and its thick inhomogeneous structure brought a new class of problems in electromagnetic analysis of field penetration, particularly for the poloidal field coils. With plasma disruptions introduced as a plausible event in the operation of tokamak reactors, the electromagnetic problems are exacerbated by the possibilities of arcing between sectors of first walls or limiters and body forces in these components due to eddy currents. (Current judgment holds that although disruptions are undesirable, they are likely, and the consequences are tolerable rather than catostrophic.)

One hurdle in the resolution of design issues concerning the electromagnetic responses of components is the limitation of the design codes that deal with transient electromagnetic behavior. The segmentation and complex configurations of the components make 3-D calculations desirable, but the existing codes have not yet progressed to the sophistication of 3-D treatments seen in fields like stress analysis and thermohydraulics. Another limitation has been the lack of experimental data to establish the limits for 2-D approximations of 3-D problems and for development of 3-D codes. Some feeling for the nature of these problems can be gained by reviewing some experimental results on magnetic damping from an electromagnetic test stand, FELIX, recently constructed at Argonne National Laboratory.

Researchers at the Princeton Plasma Physics Laboratory (PPPL) discovered the beneficial effects of "magnetic damping" in reducing the deflections of components and included this effect in their design calculations for the limiters for the Tokamak Fusion Test Reactor now in operation at PPPL. The particular conditions of interest for "magnetic damping" occur when a component, such as a limiter blade, deflects and the deflection causes a conducting surface to intercept additional toroidal flux. This flux change produces eddy currents and a related force that opposes further motion, i.e., a restoring force (see Figure 9). The effect is greatest when the surface is initially parallel to the field lines (e.g., a limiter blade).

The phenomenon of magnetic damping was investigated experimentally during 1984 in tests where the eddy currents were introduced into a flat plate by pulsing a (dipole) magnetic field orthogonal to the plate (and to the background field.) Some results are shown in Figure 10. The angle of rotation about the axis (y) through the middle of the plate and the integrated current density in the plate were continuously monitored during the test. The accompanying figures show the time dependence of the total current J and the angle of rotation θ.

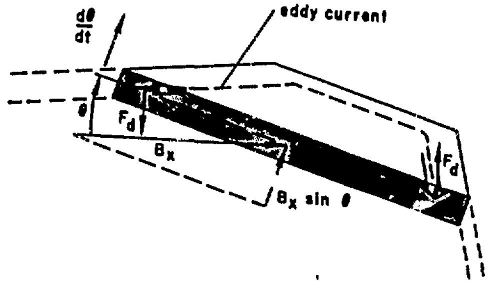


Figure 9. Rotation of the plate out of parallel to magnetic field B produces eddy currents and a resulting torque that opposes the rotation.

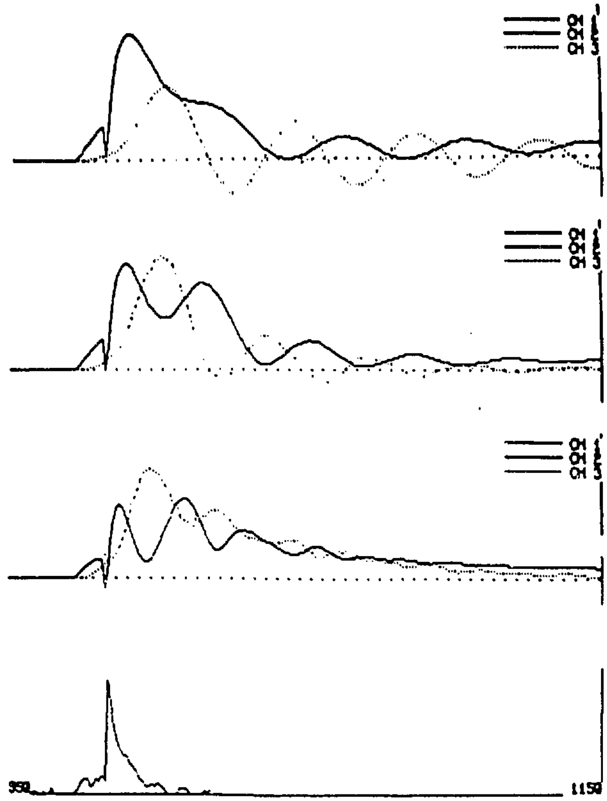


Figure 10. Three runs from tests on magnetic damping. Current (solid line) and angle of plate (dotted line) versus time all shown (top to bottom) for fields of 0.2, 0.4 and 0.8 tesla. Total time across frame is 400 milliseconds. Spike at bottom shows decay of orthogonal field that produced initial eddy currents in plate.

In the top curve, the field $B_x$ of 0.2 tesla (T) or 2000 gauss is relatively low. The current profile is readily recognizable as the sum of the initial current (from the dipole field pulse) that dies out exponentially plus a smaller periodically varying current from the motion of the plate.

In the bottom curves there is heavy damping. The initial current profile is strongly influenced by the motion of the plate and at the peak in angle θ the current has been reduced to near zero by the eddy currents that were generated by the motion of the plate. After the first large deflection, subsequent motion is severly restricted. The time-averaged behavior is one of the spring slowly returning the plate to the horizontal position against the damping force.

The initial torque on the plate is four times greater at 0.8 T than at 0.2 T but the maximum deflection is less than 40% greater. The amount of energy dissipated by the system through $I^2R$ heating is about half that at the lower field. The additional energy present in the system (that does not appear as $I^2R$ heating in the plate at higher fields) is exchanged with the dipole and solenoidal fields through the mutual inductance of the plate with these fields. This exchange is the key factor that limits the mechanical energy available for deformation of components.

## CONCLUDING REMARKS

The development of fusion has overcome many obstacles and proof of the scientific feasibility of fusion as a source of harnessable power is tantalizingly near at hand. The next major challenge in developing fusion as a practical energy source is a demonstration of "engineering feasibility." Basically this will involve the development and integration of many technologies, and most likely will include large superconducting magnets, RF power injection, tritium processing, etc. Resolutions in design concepts and in the issues in heat transfer and mechanical interactions presented here will of necessity be forthcoming, although for the near term the pace of technology development in the national program is slowing.

The engineering disciplines to achieve the needed progress in fusion technology are not new. However the configurations and requirements for materials in fusion reactors make the applications of these technologies unique.

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