

Overview of Current Research in Fusion First Wall, Blanket and Shield Technology

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Summary

This brief overview of first wall, blanket and shield technology reviews the changes and trends in important design issues in first wall, blanket and shield design and related technology from the 1970's to the 1980's. The emphasis is on base technology rather than either systems engineering or materials development. The review is limited to the two primary confinement systems, tokamaks and mirrors, and production of electricity as the primary goal for development.

1. Introduction

The next generation of fusion devices, after TFTR and MFTF-B, will require major advances in supporting technology, for example, the introduction of breeding blankets and actively-cooled first walls. These are areas of new technology in which we can draw very little from our experience on existing devices.

Our understanding of the requirements for technological support needed to demonstrate fusion as a power source have progressed significantly in several areas related to first wall, blanket and shield (FWBS) components, since the mid 1970's. This insight into the needs for supporting technology comes primarily from design studies where new design concepts evolve and are subjected to some scrutiny in preliminary evaluations of their feasibility. This review will draw extensively upon three recent design studies, STARFIRE,¹ MARS,² and DEMO³.

The design studies are particularly valuable for elucidating R&D needs and they play an important role in formulating an overall strategy for the advancement of fusion. However it also prudent to remember that we are in an "idea" stage and there is some difficulty in appreciating the engineering tasks implied in these concepts, basically for two reasons. First, there is insufficient detail from which to complete an inventory of the engineering tasks that will be needed. Second, there is a personality to the conceptual development phase which tends to relegate engineering development to a somewhat straightforward exercise of manpower. A balanced vision of our future progress must recognize that, real as our progress is in the advancement of ideas, there also awaits a major challenge in engineering and R&D in order to make these ideas work.

2. Summary of Progress - In FWBS Design

Several important changes in the development of first wall, blanket and shield technology have occurred since the mid 1970's. The following discussion groups these changes into three areas: global analysis and blanket concepts, plasma engineering and first wall concepts, and engineering evaluations.

2.1 Global Analysis and Blanket Concepts

Important progress of a global nature has come from the advancement of 3-D neutronics codes, from evaluations of various concepts for breeding tritium and from systems studies that have provided a method for design integration and cost analysis. 3-D analysis has also been developed for shielding and activation codes. In comparison with 1-D models, 3-D analysis generally gives a value for the breeding ratio that is roughly 10-20% lower. Uncertainties of this magnitude are unacceptable, as discussed elsewhere^{4,5}

The accurate prediction of breeding ratios (BR) is important to show the feasibility (BR > 1.05 - 1.1) of the concepts. Recent experiments have resulted in about 15% lower cross-sections for $\text{Li}(n,\alpha)\text{t}$ in ENDF/B-V (mod. 2) which may render tenuous some attractive blanket systems, such as Li_2O (without multiplier) or Li alone, where the predicted breeding ratios are only marginally acceptable.

UWMAK-III⁶ was the first major adaptation of 3-D neutronic codes to design studies in order to obtain more accurate predictions of tritium breeding ratios. Since then neutronics and engineering analysis in combination with the material evaluation of various breeder concepts has proceeded through reactor design studies and special blanket studies such as the Blanket and Shield Design Study⁷ (1978/79) and the current Blanket Comparison and Selection Study, which began in late 1982 as a two-year project with the objectives of reviewing all blanket concepts and recommending the more promising candidates. Among the changes since the late 1970's in preference for blanket concepts are the utilization of solid breeders, general dissatisfaction with molten salts, emergence of liquid lithium alloys, and the concern and recent reconsideration of safety issues related to use of pure lithium.

A whole new type of breeding blanket evolved with the use of solid breeders in the UWMAK-II design.⁸ The advent of solid breeders has offered the possibility of using high pressure water coolant for heat extraction from the first wall and blanket with the attendant advantage of its relatively well established technology, especially in power conversion systems. Continuing evaluations of potential solid breeders have resulted in many possible candidates (and a general preference for oxides). Li_2O , LiAlO_2 , $\text{Li}_5\text{Al}_4\text{O}$, Li_2SiO_3 , Li_4SiO_4 , Li_2TiO_3 , Li_2ZrO_3 and Li_3ZrO_6 are current candidates being evaluated by the U.S. Fusion Reactor Materials Program.⁹

The advancement of liquid lithium alloys ($^{17}\text{Li}^{83}\text{Pb}$) has offered the attractiveness of combined breeder-coolant transport with much-reduced concerns about their hazard potential in case of spills. A general concern about large circulating power requirements for pumping liquid metals in high magnetic fields has prompted evaluations of MHD effects. Recent evaluations¹⁰ of such effects have indicated the need for some type of mixing of liquid metal coolant near the first wall in order to effectively transfer and distribute heat from the first wall (in tokamaks) into the bulk of the liquid metal coolant.

The integrated treatment of reactor designs was widely noticed first with the UWMAK-I design study.¹¹ The early series of design studies from the University of Wisconsin provided several "point" designs. As these and other point designs evolved, the capability to do parametric studies as variations from a point design was developed. The "ANL Parametric Systems Studies"¹² was among the first comprehensive code evaluations and the current "FED Systems Code"¹³ is the most sophisticated step in the evolution of these codes. The codes have provided several kinds of useful information, including predictions of operating economics and component costs. Two important results from such studies have been the reduction in goal lifetimes for first wall exposure from 20-40 $\text{MW}\cdot\text{y}/\text{m}^2$ to 10-15 $\text{MW}\cdot\text{y}/\text{m}^2$ and a recognition of the potentially large cost of shielding, for example, shielding cost was reduced from 32% of the cost of reactor equipment in STARFIRE to 11% in DEMO.¹⁴

2.2 Plasma Engineering and First Wall Concepts

Advancements in plasma engineering in both mirror and tokamak programs have improved the attractiveness of their confinement concepts and their respective in-vessel components (limiters, divertors, armor, halo scrapers, end plates, etc.) In mirrors, advancements in end plug design, from thermal barriers¹⁵ to beam pumping to drift pumping,¹⁶ have revitalized the mirror confinement concept and reduced the total particle load that passes the end plugs to the plasma dumps. In tokamaks the advancements of pumped limiters and pseudo-steady state

operation through non-inductive current drive have simplified magnet (and machine) configuration and largely mitigated problems with fatigue of structural members. The STARFIRE design represents the first comprehensive attempt to develop a detailed design, supported by physics and engineering analysis, that incorporated these features. (The pumped limiter had previously appeared in other variants in the literature¹⁷⁻²² and non-inductive current drive methods had some theoretical basis with very limited experimental confirmation.)

As the FED/INTOR (then ETF and INTOR) and STARFIRE concepts were evolving and pumped limiters were incorporated into these designs, the impact of surface erosion due to sputtering by energetic particles from the plasma was also being recognized.²³⁻²⁸ Along with the consequences of erosion, transport and redeposition of material by the plasma was also studied²⁹ and in STARFIRE, a redeposition scenario was postulated which substantially mitigated the erosion problem.³⁰

The impact of disruptions, previously perceived as catastrophic, was also being reevaluated. Disruptions became potentially manageable by attention to the materials and designs for in-vessel components, notably thicker first walls and limiters and, in ETF³¹ and initially in FED,³² armor as a sacrificial surface to protect the first wall from disruptions.

During this period of design development, selections of materials for in-vessel components were also influenced by estimates³³ of orders of magnitude higher tritium permeation through the first wall, when implanted tritium was used as the source rather than adsorbed tritium, and by new data on enhanced sputtering of graphite at high temperatures. The data on enhanced (chemical) sputtering of graphite³⁴ indicated that at nearly all temperatures above about 500° C, graphite sputtering was roughly an order of magnitude greater than at room temperature, a finding that severely limited its application in design and rendered infeasible many uses in previous designs. Graphite had been the "workhorse" for many in-vessel components in the late 1970's because its combination of thermal and physical properties offered the promise of radiatively-cooled, low Z surfaces facing the plasma.

The curtailed utilization of graphite forced tokamak designers toward actively-cooled in-vessel components with provisions in design for material loss due to disruptions and to erosion. This trend in design produced in-vessel components with thick composite structures. For example in DEMO the first wall is 4 mm of SS with a 2 mm cladding of Be and the limiter uses beryllium-strengthened copper or vanadium as the substrate and 25 mm thick Be tiles on the surface, except at the leading edge where a tantalum coating is recommended. The issue of materials selection for pumped limiters is well documented in both the DEMO³⁵ and FED/INTOR³⁶ reports.

Relaxed requirements on first wall lifetime brought more widespread use of stainless steel (SS). Ferritics (HT-9) have also been advanced as candidate first wall materials,³⁷ for example in recent tandem mirror designs WITAMIR³⁸ and MARS.²

2.3 Engineering Evaluations

Two other major changes related to FWBS components are the evolution of a general philosophy of remote maintenance and the emergence of calculational methods to deal with electromagnetic effects. Concern about remote maintenance forced attention on machine

configurations that would permit access to the FWBS components. Among the design changes this philosophy wrought in tokamaks was placement of EF coils outside the TF coils, placement of the vacuum boundary away from the first wall and segmentation and modularization of FWBS components for removal between the TF coils.^{39,40} In mirrors, the modularization in the central cell was accomplished either by removing the magnet and FWBS segment as a unit, or by making smaller solenoidal FWBS segments that could be removed between the coils.^{2,41,42}

In tokamaks, the segmentation of the first wall and blanket and its thick inhomogeneous structure presented 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 were exacerbated by the possibilities of arcing between sectors of first walls or limiters and body forces in these components due to eddy currents. One impact from these concerns has been a design requirement⁴³ (in FED) for high current (non-welding) electrical contacts between sectors that can be remotely separated as need for maintenance would dictate.

3. Recent First Wall and Blanket Systems

Our current projections of the FWBS technology are embodied in the designs for commercial reactors such as STARFIRE¹ and MARS;² these are the most comprehensive design studies to date respectively for tokamak and tandem mirror reactors. Table 1 gives some information from these designs and from DEMO³ which is a "next generation" tokamak rather than a commercial reactor but contains more recent evaluations than STARFIRE.

Current estimations of heat load to the first walls in mirrors is much lower than for tokamaks. If realized, the lower thermal load on the first wall may lead to mirror designs

TABLE 1 PARAMETERS FOR STARFIRE, DEMO AND MARS

	<u>STARFIRE</u>	<u>DEMO</u>	<u>MARS</u>
Fusion power (MW)	3510	1069	2570
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	1.39
Thermal power in first wall/blanket (MW)	3866	1176	2880
Power to limiter/end dump (MW)	200	151	231
First wall material	Be/SS	Be/SS	HT-9
Limiter/end dump material	Be/Ta ^d	Be/Cu or V ^d	V/Cu ^d
Neutron multiplier	Zr ₅ Pb ₃ (or Be)	None	None (Pb in Li-Pb)
Breeding material	LiAlO ₂ (90% ⁶ Li)	Li ₂ O (natural)	17Li83Pb (90% ⁶ Li)
Blanket structure	SS	SS	HT-9
Blanket coolant	H ₂ O	H ₂ O	17Li83Pb (90% ⁶ Li)

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. Includes 231 MW of thermal power from plasma end dump. Direct conversion also produces 324 MWe (not included in the thermal power).

d. Ta and V here imply alloys such as Ta-SW and V-15Cr-5Ti or V-20Ti.

with higher neutron wall loadings and the use of higher power density blankets. The lower first wall heat load in mirrors would also mitigate concerns with heat buildup in the boundary layer adjacent to the first wall in systems where the first wall is cooled with liquid metal.

The in-vessel components in any D-T reactor must collect and discharge about 20% of the fusion power produced by the reactor (as alpha particles which transfer energy to other particles and ultimately to the in-vessel components as particle bombardment or radiation). In STARFIRE and DEMO this alpha power is balanced between the limiter and the first wall. In DEMO the power to the limiter is about 14% of the fusion power and this thermal power (which also includes nuclear heating) is not collected in the power cycle in order to simplify the design. In STARFIRE the ratio is less than 6%. There the heat load to the limiter was reduced and transferred to the first wall by injecting a controlled impurity (iodine) into the plasma. By comparison, the first wall in STARFIRE receives 82% of the alpha power versus 48% in DEMO.⁴⁴ The impurities increase the fraction of alpha power dissipated as radiation with two beneficial effects. First, the radiation is a more benign surface loading condition (no sputtering). Second, the power from the limiter is low grade heat (low pressure, low temperature water) and is used less efficiently for feedwater heating than the high grade heat in the first wall, so there is also an advantage in efficiency in increasing the fraction of power to the first wall.

In the MARS design surface heat load on the first wall (per MW of fusion power) is about 9% of that of STARFIRE and the power collected on the plasma end dump is proportionally about 60% more (than the STARFIRE limiter) and is almost entirely particle heating. The large amount of power to the end dump requires that it be collected and processed as high grade heat and the MARS design utilized direct conversion, with the halo scraper as the (grounded) anode and the end plate as a biased cathode. The implied design conditions are challenging and require high voltage isolation and structures at high temperature with high pressure coolant.

The introduction of blankets for collection of heat and production of tritium is probably the single greatest change in technology between near term devices and future self sufficient D-T reactors. The basics of blanket design do not differ significantly between tokamaks and mirrors. Although there are some differences in piping and manifolding, these are minor compared to the gross differences in the types of breeding systems being considered, as is evident in the choices of blankets for STARFIRE (solid breeder LiAlO_2 with a neutron multiplier), DEMO (solid breeder Li_2O without a neutron multiplier), and MARS (liquid metal $^{17}\text{Li}^{83}\text{Pb}$). For DEMO, $^{17}\text{Li}^{83}\text{Pb}$ was also considered as an alternate blanket system. Materials selections for a variety of blanket concepts have been reviewed by Smith⁴⁵ and the DEMO report⁴⁶ contains a comparison of the merits of these breeder systems (and pure lithium) for use in DEMO and includes some parametric analysis of variants within each system, such as the use of neutron multipliers (except for Li-Pb), full or partial coverage of the breeding blanket, the use of secondary coolants with liquid metal systems and methods for tritium recovery. The comparative breeding performances of these systems are summarized in reference 6.

With regard to their thermal power for production of electricity, blanket concepts range from about 110% to 140% of the virgin (14.1 MeV) neutron power born in the plasma. This

amplification of power in the blanket comes from $\text{Li}(n,\alpha)t$ reaction which produces 4.76 MeV and has a high cross section for thermal neutrons. As is evident in the comparison of blanket power multiplication factors in Table 1, the blanket power multiplication is particularly high for the Li-Pb system. The high value comes from the use of a breeder highly enriched in ${}^6\text{Li}$ and a soft neutron spectrum. With ${}^{17}\text{Li}{}^{83}\text{Pb}$ the spectrum is heavily moderated by $\text{Pb}(n,2n)$ and $\text{Pb}(n,3n)$ reactions.

TABLE 2 US PROGRAMS/ACTIVITIES RELATED TO FWBS TECHNOLOGY

First Surface Activities

Plasma edge studies
Materials Program - Plasma Materials Interactions Task
High Heat Flux Component Development Program
FWBS Program - disruptions

First Wall Activities

FWBS Program - surface heating
Materials Program - Alloy Development for Irradiation Performance Task

Blanket Activities

FWBS Program - blanket simulation studies
Tritium Recovery in ORR (TRIO) Experiment
Breeder neutronics
Materials Program - Solid Breeder Development Task
Tritium System Test Assembly (TSTA)
Lithium Blanket Module Experiment (LBM)

Shield Activities

Shielding experiments

General

Design studies
Neutronics
FWBS Program - electromagnetic effects
- remote assembly, maintenance, repair
Safety Program

4. Current Activities

Table 2 lists current programs and activities in the U.S. that are directly concerned with R&D on first wall, blanket and shield technology. A more extensive version of this review published recently^{4,7} includes a discussion of critical issues in technology and refers to work in ongoing programs.

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