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OVERVIEW OF FIRST WALL/BLANKET/SHIELD TECHNOLOGY*

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ABSTRACT

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

INTRODUCTION

Our understanding of the requirements for the technological support needed to advance fusion as a commercial power source have progressed significantly in several areas related to first wall, blanket and shield (FWBS) components, since the mid 1970's. This progress will be briefly summarized in this paper with the following three types of observations: (1) major changes and trends that have influenced our evaluations of requirements in technology; (2) current perceptions of critical issues for FWBS components; and (3) U.S. programs with objectives of developing FWBS technology.

Our assessments of the technological development needed for advanced reactors actually draw very little from experience on existing devices. This is true because our progress to date has been largely in the areas of physics and plasma engineering whereas the next generation of 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.

Most of our assessments of these R&D needs come 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 DEMO[3].

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, as we should be, in an "idea" stage. It is especially true in an endeavor as technologically sophisticated as fusion, that this conceptual development stage has its own type of intoxication in which there is a premium on conceptual evaluation. There is also some difficulty in appreciating the engineering tasks implied in these concepts, basically for two reasons. First, there is insufficient refinement of detail from which to glean a complete inventory of the engineering tasks that will be needed. This must await a detailed design effort that will come later. 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.

SUMMARY OF PROGRESS - IN FWBS DESIGN

Major changes since the mid 1970's in the development of first wall, blanket and shield components are summarized in Tables 1 - 4. Table I lists changes that have had a general impact on all these components and Tables 2, 3, and 4 are specific to in-vessel components, blanket and shield respectively. ("In-vessel components" include first wall, limiter, armor and plasma dumps.) In the text these changes are grouped into three areas: global analysis and blanket concepts, plasma engineering and first wall concepts, and engineering evaluations.

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Global Analysis and Blanket Concepts

Important progress of a global or systemic 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 significantly (10-20%) lower. Uncertainties of this magnitude are unacceptable, as discussed elsewhere[4,5] and the progress and future refinement of 3-D analysis are important advances.

For reactor operation the prediction of tritium breeding ratios is important in order to balance fuel production with consumption. For design development, 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 Li(n,nα)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-III[6] 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 Study[7] (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] This has had a profound impact on advanced reactor design. 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, $LiAlO_2$, Li_5Al_4O, Li_2SiO_2, Li_4SiO_4, Li_2SiO_3, $Li_2Z_rO_3$ and Li_8ZrO_6 are current candidates being evaluated by the U.S. Fusion Reactor Materials Program.[9]

The advancement of liquid lithium alloys (17Li83Pb) 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 evaluations[10] 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 probably most 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 MW-y/m^2 to 10-15 MW-y/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]

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. In mirrors, advancements in plasma engineering and end plug design, from thermal barriers[15] 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 literature[17-22] and non-inductive current drive methods had a firm theoretical basis but somewhat 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 (discussed later), transport and redeposition of material by the plasma was also studied (and recently reviewed)[29] and in STARFIRE, a redeposition scenario which substantially mitigated the erosion problem was postulated.[30] About this same time, the impact of disruptions, previously perceived as catastrophic, was being reevaluated. The new characterizations of disruptions were potentially manageable by attention to the materials and designs for in-vessel components, notably thicker first walls and limiters and, in ETF[31] 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 estimates[33] 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 graphite[34] 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 DEMO[35] and FED/INTOR[36] reports. Relaxed requirements on first wall lifetime, prompted in part by the difficulty in achieving 40 MW/m^2 especially with stainless steel, brought more widespread use of stainless steel (SS) in preference to "advanced alloys" such as vanadium, niobium, molybdenum and their alloys. Ferritics (HT-9) have also been advanced as candidate first wall materials,[37] for example in recent tandem mirror designs WITAMIR[38] and MARS.[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 detail with (dynamic) 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 requirement[43] (in FED) for high current (non-welding) electrical contacts between sectors that can be remotely separated as need for maintenance would dictate.

RECENT FIRST WALL AND BLANKET SYSTEMS

Our current projections of the FWBS technology for power-producing advanced fusion reactors are embodied in the designs for commercial reactors such as STARFIRE[1] and MARS;[2] these are the most comprehensive design studies to date respectively for tokamak and tandem mirror reactors. Table 5 gives some information from these designs and from DEMO[3] which is a "next generation" tokamak rather than a commercial reactor but contains more recent evaluations than STARFIRE. The overall objective in the technology integrated in these designs is the production of electricity and Table 5 provides some insight into the similarities and differences between tokamaks and mirrors in achieving this objective, particularly with respect to the distribution of heat loads on in-vessel components.

One noticeable difference is that the 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 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 (discussed later).

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 particle loading 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 STAFIRE 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 (discussed later).

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 17Li83Pb). For DEMO, 17Li83Pb was also recommended as a preferred alternate blanket system. Materials selections for a variety of blanket concepts have been reviewed by Smith[45] and the DEMO report[46] contains a side-by-side 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 performance of these systems is summarized in this conference.[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 $^6Li(n,\alpha)t$ 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 5, 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 ^6Li and very soft neutron spectrum. With 17Li83Pb the spectrum is heavily moderated by Pb(n,αn) and Pb(n,3n) reactions. For a Li-Pb system, the step of going to an isotopically enriched breeder appears to be desirable. The reader is referred to references 45 and 6 for more detailed discussions of this issue.

CURRENT PERSPECTIVE ON CRITICAL ISSUES AND PROGRESS IN FWBS TECHNOLOGY

"Critical issues" are those technical problems which thwart further progress in advancing the designs of major reactor systems, in this case the first wall, blanket and shield. Our identification of "critical issues" in FWBS technology has evolved from several sources but primarily from design studies. The more comprehensive design studies such as STARFIRE, FED/INTOR, DEMO, MARS and the current Blanket Comparison and Selection Study have each involved large groups of people from various institutions and have provided an active forum for innovation and peer review.

Tables 6, 7 and 8 list various "critical issues" for in-vessel components and solid breeder and liquid metal blankets. Also listed are some possible advances in technology that may (but not necessarily will) be required in order to resolve the concerns listed. Not shown in the tables are the needs for R&D on design methods, and development of engineering data on materials and components. (A review of these needs would exceed the limited scope of this paper.)

Figure 1 shows ongoing U.S. programs with activities related to the development of FWBS technology. References to ongoing work in these activities will be noted as part of the following discussion of critical issues, although the descriptions of these activities are brief and not intended to be comprehensive and activities dealing with engineering and technology development are emphasized. Both the U.S. Fusion Reactor Materials Program (hereafter called the Materials Program) and the First Wall/Blanket/Shield Program (FWBS Program) have several program elements and the abbreviations for programs and program elements in Fig. 1 will be used subsequently in the text.

Disruptions

For tokamaks, the capability of the first wall and limiter to withstand major plasma disruptions is a major design concern. Sources of potential problems are (electromagnetic) forces, arcing, loss of material by vaporization and degradation by surface melting, i.e. whether the surface layer of material melted during a disruption will remain in place and resolidify with sufficient mechanical integrity such that repeated disruptions will not result in a catastrophic loss or degradation of material.

The disruption phenomenon is characterized in two phases. In the first phase (thermal quench), the kinetic energy of the particles is dissipated after their loss of confinement and impingement on some part of the plasma chamber; the limiter and inboard wall of the plasma chamber are preferred sites. In the second phase (current quench), the electrical inductance previously stored in the plasma is transferred to the torus structure and dissipated by eddy currents in the torus structure. Important parameters in characterizing the response of the limiter (or first wall) to the intense transient heat load of the thermal quench are the plasma energy, the time and the affected surface area. The latter two parameters are not well defined and disruption analyses tend to treat the problem parametrically. Typical values are times of 5 ms and 20 s and energy densities in the range of 250 - 1000 J/cm^2. Much of the effort to characterize the parameters in order to define engineering requirements was done in conjunction with the FED/INTOR Study and has been published.[47-50] Excellent characterization of the issues and related analyses are included in the FED/INTOR[36] and DEMO[51] reports. In the FWBS Program (I), the response of materials to disruptions is being simulated experimentally using electron beam heating.[52,53]

Some examples of evaluations of electromagnetic effects on in-vessel components are presented in this conference.[54,55] The general area of electromagnetic effects is being studied in the FWBS Program (III) and an experimental facility (FELIX) will be completed in 1983 for tests on electromagnetic effects.[56]

Damage from arcing due to plasma disruptions could result in vaporization of material and self-welding between adjacent components. A (hypothetical) problem would be if adjacent limiter sections were welded together in that their removal was not possible without an unplanned remote cutting operation within the plasma chamber in order to proceed with normal remote maintenance. There are still many unknowns in the resolution of this problem. A highly conducting first wall, or conductive cladding such as beryllium, will lower the induced voltage differences between sectors, perhaps below the threshold voltage to initiate arcing which is not well established for the conditions in a plasma disruption.

The uncertainties concerning arcing have prompted another type of design solution -- to simply close the circuit between sectors. A design solution for arcing proposed for FED was to place preferred paths for current conduction between sectors by the use of electrical contactors that could be remotely disengaged to allow sector or limiter removal for maintenance. Experimental evaluations of materials for such contacts performed in the FWBS Program (IV) are reported in this conference.[57]

Erosion/Redeposition

The consequence of erosion of in-vessel components, due to sputtering by energetic particles from the plasma, is a potentially substantial loss of material. This loss may be mitigated in part by redeposition of eroded material. This subject has been evaluated extensively in several design studies[36,58,59] and is reviewed elsewhere in this conference.[60] Evaluations of erosion have been based on ion sputtering data at relatively low fluxes so there is little data to resolve questions about surface conditioning and redeposition at high particle fluxes. Facilities now in preparation at Sandia National Laboratories (Albuquerque) as part of the High Heat Flux Component Development Program[61] and at the University of California (Los Angeles)[62] will become available in the near future to study surface effects at high particle fluxes.

Composite Structures

The trend toward composite structures for in-vessel components raises general design concerns about a method for assuring a sound bond between the clad and structure. The structures will be subjected to repeated temperature cycles in the next generation of experimental devices and even in "steady state" reactors. Adequate mechanical integrity implies considerations for strain isolation, to reduce differential thermal strains at the interface, and for high thermal conductivity in order to control the temperature of the cladding. Plasma sprayed coatings that retain some porosity may offer both adequate compliance to accommodate differential thermal strains and, with techniques now being developed in the Materials Program (PMI),[63] thicknesses of several millimeters as specified in current designs.

Another approach is to join cladding to the structure through an intermediate layer such as metal felt or fingers. Some development of this concept was pursued and small pieces were fabricated in conjunction with the design development for TFTR limiters.[64] The design requirements for such composite structures are governed by the operating conditions and also in the case of a brazed joint by the fabrication cycle that includes a brazing step at some temperature significantly higher than the operating temperature.

In first wall applications such as STARFIRE and DEMO which use Be/SS composite structures and for the plasma end dumps in MARS, an important design consideration is the high temperature (> 300° C) of the water coolant and associated high pressure (> 8.6 MPa or 1250 psi) required for efficient power extraction. For limiters, the intensity of the thermal and particle loads has generally lead to the use of cool (low pressure) water to reduce the temperature of the structure and stresses from the coolant. The design requirements for the MARS plasma end dumps[16,65] include both high thermal loads (~ 2.8 MW/m^2 on the halo scraper) and high temperatures. In addition, the end plate is negatively biased and must include standoff insulators in the cooling system. The current preference for the cooling structure of both the end plate and the halo scraper is a composite coolant tube sheathed with TZM and coated with porous, plasma-sprayed vanadium.

The thermal and mechanical performance of composite structures, and in-vessel components in general, are subjects of investigation in both the High Heat Flux Component Development Program and the FWBS Program (I). The latter program will include tests on Be/SS composites this year (1983).

First Wall Coo'ing (Liquid Metals)

In liquid metal systems, MHD forces tend to encourage lamillar flow and suppress mixing in the coolant. This phenomenon would result in large thermal gradients adjacent to the first wall because, without mixing, the heat penetration into the coolant essentially relies only on conduction. When there is a comparatively large heat flux on the first wall (as for tokamaks), the permissible residence time for coolant to remain near the first wall is rathei short. A time of .1 sec was estimated for 17Li83Pb with a heat load of 0.5 MW/m^2 in evaluations performed in the Blanket Comparison and Selection Study.[10] This constraint on heat penetration will require some method of mixing the boundary layer with the bulk of the coolant. Flow mixing, for example with internal fins, and convolution of the flow path by staggered radial baffles that force the flow radially into the first wall and then away from it (somewhat like the UWMAK-I design but on a smaller scale) have been proposed in the Blanket Comparison and Selection Study.

Tritium Permeation

Tritium permeation from the plasma into the water system of a water-cooled first wall has received considerable attention since the realization that tritium *implanted* as energetic particles from the plasma could increase the source term for diffusion, compared with the relatively slower process of surface adsorption, enough to increase the permeation rate by several orders of magnitude.[33] A similar problem may be encountered in the plasma end dump for MARS.[66]

At the free surface use of porous cladding can retard permeation by providing the opportunity for migration back to the plasma side through the interconnected porosity. Permeation barriers on the coolant side, for example oxides on the surface of stainless steel, will also tend to reduce permeation. The FED/INTOR report[67] contains an extensive evaluation of tritium permeation.

Breeding/Multipliers/Tritium Recovery

For solid breeding blankets the general design issue of most concern is obtaining an adequate tritium breeding ratio with minimum reliance on a neutron multiplier (specifically beryllium). Various solid breeders have been considered and Li_2O and $LiAlO_2$ have received extensive evaluations in the DEMO and STARFIRE studies respectively. (See also papers by Jung and Abdou[6] in this conference.)

Although the usable range of temperature for Li_2O is believed to be rather narrow, cited as 410 to 660°C in the DEMO report,[68] its lithium atom density (0.93 g/cm^3, 100% dense) is roughly twice that of other solid breeders and its good breeding capability has made Li_2O a preferred candidate for solid breeding blankets. The perceived low temperature limit for Li_2O is based in part on a concern that diffusion of tritium out of the breeder (for recovery) may be slow at low temperatures, especially after irradiation has created traps. This concern pertains to other solid breeders as well but there is as yet little data to confirm these modeling predictions.

The general precaution taken in design is to specify breeder densities below 80% so that interconnected porosity will assist transport of the tritium to the purge gas collection system. For STARFIRE a special bimodal porosity was specified and samples of $LiAlO_2$ in this form were developed by the Materials Program.[69] Fabrication of Li_2O pellets in large quantity was initiated as part of the Lithium Blanket Module Experiment.[70] Pellets of this type were the starting stock for previously mentioned tests in the FWBS Program (II).

Results from the TRIO experiment,[71] which began irradiation in March of 1983, will help clarify the issues of tritium retention, the time constant for tritium release and the form of the recovered tritium (water, hydrogen gas, hydroxide?). The characterization of mechanical and physical properties of solid breeders, thermodynamic equilibria and data on radiation effects is the subject of ongoing work in the Materials Program. A review of this work will be published shortly after this conference.[9] Also, some early scoping tests have been conducted in the FWBS Program (II)[73] to study the stability of a breeder system with regard to material transport by vaporization and the thermomechanical affects of cycling. The tests use Li_2O in a flowing stream of helium. Details are described in this conference.[73]

Concerns about heat transfer and mechanical integrity stem from a general requirement to preserve the physical form of the breeder against densification, vaporization and material transport and cracking under thermal stresses. Heat transfer at the interface between the breeder and heat sink sets the minimum operating temperature for the breeder. For designs with Li_2O this is particularly important because the precipitation of LiOH (LiOT) expected below about 410°C could lead to problems with mass transport and corrosion. Maintaining roughly a 100°C temperature difference between water-cooled structure at about 320° C and the adjacent Li_2O may require some type of compliant thermal bond at the interface, for example metal felt brazed between the breeder and the heat sink; other design concepts such as double-walled heat sink with a gas gap and wire wrap have also been proposed. The heat transfer coefficient between the breeder and the heat sink and its sensitivity to the gap or to contact pressure is uncertain so the design tolerance for an unbonded system is in doubt. As part of the FWBS Program (II), 1-D heat transfer experiments are being performed to determine the heat transfer coefficient between Li_2O and stainless steel, as described elsewhere in the conference.[72,73]

The DEMO design (like STARFIRE) uses a blanket configuration of coolant tubes imbedded on the breeder material. Analysis of the thermal stresses indicates a propensity for cracking to form at the breeder/tube interface in the radial/axial planes (perpendicular to the surface of the tubes). Potential problems that could arise from cracking are discontinuities in the thermal conduction path could cause hot spots or that displacement of material could cause stresses on the coolant tube during thermal cycles. However, the most likely "crack planes" are parallel to the heat flow path and the impact of cracking or crazing is as yet not known.

Water and Helium and Tritium Permeation

The use of water coolant with solid breeders is convenient for coupling to the power conversion system but also presents two general areas of concern related to reliability and safety. Minimizing any potential hazard for accidental release of tritium is a design goal. With solid breeder blankets, two potential areas of concern are permeation of tritium into the coolant system and blanket failure that might breach the primary blanket containment and force tritium either into the coolant stream via a breached tube or out of the blanket wall via a wall rupture.[74] Analysis of the potential breach problem has been a part of ongoing evaluations in the Fusion Reactor Safety Program. The contribution of tritium retained in the solid breeder to the releasable inventory in such an event is unclear; however, the engineering of some types of fail-safe systems for blanket containment will clearly be a design requirement and may necessitate some adaptations in blanket technology such as double-walled systems, overpressure plenums, etc. A related concern with the overall reliability of solid breeder blankets is the potential for repairing any minor failures in the blanket system (discussed later).

For helium-cooled blanket designs the generic concern is the mechanical integrity of the structure at the high temperatures (generally above 600°C) needed to obtain good thermal conversion efficiency for helium. The requirement for high temperature generally drives the design of the structure to advanced alloys (e.g. vanadium) as in a high temperature blanket design for MARS.[75] Issues in the selection of materials for high temperature blankets are also included in the previously cited review of blanket materials.[70]

MHD Effects and Compatibility

Table 8 briefly summarizes major design concerns for (flowing) liquid metal blankets. MHD effects are an issue because the frictional losses and pressure losses due to MHD forces potentially represent a large parasitic demand on the reactor circulating power. The design studies show acceptable values (e.g. 0.4-0.6% electrical power in DEMO). However, there is general concern that our current capabilities to perform MHD thermal hydraulic analysis do not utilize sufficiently detailed physical configurations (complex geometries-bends, expansions, etc.) nor adequate data on MHD effects at high magnetic interaction parameters (high field, high flow rates) to estimate MHD losses with confidence.

Were experimental data and more sophisticated analyses to show unacceptable requirements for pumping power, one design option to reduce this power would be the use of (electrically) insulated structure in the blanket. The requirement on dielectric integrity of these insulated walls is not yet well defined; however their use would be a noticeable departure from currently conceived technology for blankets. To date MHD data have been drawn primarily from outside the fusion programs. In 1984, the FWBS Program will implement scoping tests on liquid metal MHD effects.

Compatibility with the structure and the environment of reactor vault in case of a spill is a general concern for liquid metal systems. The extreme reactivity of liquid lithium with water and concrete raises safety concerns which might require considerable redundancy in containment for a liquid lithium system. Nickel is soluble in both lithium and lead and non-nickel bearing alloys such as ferritic steels and vanadium alloys are generally preferred in recent designs. Good reviews of compatibility issues for liquid metal systems are available in the literature.[45,46,76] The experimental lithium system for FMIT is providing some large scale experience with lithium corrosion.[77] The DEMO report also summarizes issues concerning corrosion in LiPb systems.[78]

Tritium Containment/Extraction Safety

Li and Li-Pb systems present quite different requirements in tritium recovery. Lithium has a very high solubility for tritium and separation of the tritium from the bulk lithium is the primary design concern. Several methods for tritium recovery have been proposed (gas sparging, membrane diffusion and gettering systems) and batch processing by molten salt extraction has been proven successful for laboratory-scale tests.[79] Were lithium selected as the breeder for an advanced fusion device, some large scale demonstration of an effective tritium recovery process would be required.

In Li-Pb systems, the solubility of tritium is low and the resulting high tritium partial pressure encourages tritium permeation. Consequently, the primary concern is to minimize any permeation to the environment. In DEMO, the proposed solution is to use double jacketed piping and either an intermediate heat exchanger or a double jacket steam generator.[80] Molten salt extraction is also recommended for tritium extraction. Discussions of tritium recovery and extraction are generally included in reactor design studies. The Blanket and Shield Design Study[7] also contains descriptions for several types of systems.

The Fusion Reactor Safety Program is investigating the risk and potential consequences of various (hypothetical) scenarios that involve liquid metal breeders and spills. Experimental studies include monitoring controlled spills of lithium and lithium lead.

Remote Maintenance of FWBS Systems

The absence of issues concerning remote maintenance from Tables 6, 7 and 8 is conspicuous. There are many major design concerns regarding remote maintenance; however, the format used here of brief descriptions of critical issues does not permit a coherent rendering of these issues. The reader is referred to several discussions of remote maintenance issues in the literature[81,82] and to the integration of remote maintenance into STARFIRE,[39,82] and MARS[41,2] and FED[40] design concepts. These discussions generally cover the overall philosophy of remote maintenance such as coil placement, modularization of structures, motion of major components during removal, etc.

Another aspect of remote maintenance is the specific hardware development needed to accomplish the various tasks implied in the overall maintenance philosophies. One effort to consolidate some of this type of information, primarily maintenance equipment, was a workshop held at the Fusion Engineering Design Center.[83] Typically, the development of clever design features that facilitate remote maintenance, such as remotely coupled joints, in-situ remote welders, in-vessel viewing devices and remote leak detectors are reported in conferences such as this series of Topical Meetings on the Technology of Fusion Energy (formerly the "Technology of Controlled Thermonuclear Reactors"), the IEEE series on Engineering Problems in Fusion Research, and at ANS meetings for which complete papers are published separately by the Remote Systems Technology Division, for example the 30th Conference on Remote Systems Technology (1982). Two concepts of remotely actuated vacuum joints developed as part of the FWBS Program (IV) are presented in this conference.[84,85]

The investigation of alternatives to welded joints for remote operations is one subject of investigation in the FWBS Program (IV). Within its rather limited scope, the FWBS Program (IV) is seeking to advance the capability for remote maintenance where appropriate by simple hardware demonstrations of proof-of-principal for untested design concepts. For example, in 1984 the program will develop and operate a remote latching mechanism for a concept selected from development work in 1983. In-vessel viewing systems will also be a subject of the study. The FWBS Program (IV) has also created a repository for information on remote maintenance and design, the "Designers Guidebook for First Wall/Blanket/Shield Remote Maintenance and Repair."[86] This volume will be revised intermittently to add new information. A draft of the first edition, with many sections incomplete, was distributed to a limited group for review in April of 1983.

While the FWBS Program (IV) was chartered to develop hardware for remote maintenance and is making progress, the pace and scope in this program cannot yet be construed as a viable assault on the immense challenge of providing comprehensive development of engineering methods for remote assembly, maintenance and repair. As the time nears for use of D-T fuel cycles in upgrades of TFTR and MTFT-B, we can expect expanded interest in R&D in this area.

CONCLUDING REMARKS - DEVELOPMENT STRATEGY

The preceding perspectives on historical development and critical issues in FWBS technology have reviewed our understanding of concepts and issues. A key factor missing from this discussion is an overall R&D strategy.

We are now in a stage of grappling with such a strategy. Our long range plan must include the facilities and resources to bring FWBS technology to an acceptable state of readiness for a success-oriented demonstration of fusion power production. Our increasing ability to gage the difficulties of the engineering challenge in building "next generation" fusion devices and our broader perspective on engineering development, as a combination of parallel programs in fusion, fission and non-nuclear facilities, are indications of our progress in this area.

Several groups have recently attempted to offer the elements of a comprehensive strategy. In 1981, the FED Technology Group was formed as a working group under the FED Technical Management Board and in 1982, the Testing Group was formed as part of a larger "Critical Issues" working group in the FED/INTOR activity. Both efforts produced published results[87,88] with useful information. These and other efforts are also typically subject to constraints such as the types of facilities considered, i.e. FED/INTOR plus fission reactors, and the lack of time and collective engineering experience to develop and evaluate sets of reasonable alternative facilities. Another recent effort[89] is the Workshop on the Future Engineering Needs of Magnetic Fusion conducted by the National Research Council. An overall plan ("Fusion Technology Development Plan") is being prepared by the DOE Office of Fusion Energy.

We are, as we should be, in an idea stage. Our increasingly more sophisticated evaluations of concepts for advanced fusion reactors give a progressively better understanding of the commitment needed toward engineering and technology. With some prudent empericism to complement our refinement of critical issues, resolutions and choices should be forthcoming and our efforts in FWBS technology will focus onto selected reactor systems with sufficient depth to identify the details of engineering that will make commercialization of fusion possible.

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TABLE 1 MAJOR CHANGES - GENERAL

MAJOR CHANGES - GENERAL	IMPACT ON DESIGN
3-D Neutronics Code	- Sophisticated estimates/comparisons of breeding ratios possible
System Codes/Studies	- Inter-dependence of systems better recognized
Relaxation of Goal Lifetimes ($20-40\ MW\text{-}y/m^2 \rightarrow 10-15\ MW\text{-}y/m^2$)	- Increased confidence in stainless steel
Simple Calculation of Electromagnetic Effects	- Concern for arcing, body forces on components and time response of field penetration in tokamaks
Remote Maintenance Philosophy	- Segmentation/modularization of components - Vacuum boundary away from first wall

TABLE 2 MAJOR CHANGES - IN-VESSEL COMPONENTS

MAJOR CHANGES - IN-VESSEL COMPONENTS	IMPACT ON DESIGN
Pumped Limiters	- Simplified configuration - Improved accessibility
Mirror end cell design	- Reduced total particle load to plasma end dumps - Relocation of beam armor to end cells
Disruption Characterization	- Thermal loads and body forces on components
Erosion/Redeposition	- Thicker sacrificial materials (claddings)
Characterization	- Emergence of composite first wall and limiter structures
Enhanced Sputtering of Graphite	- Actively cooled heat sinks replace graphite
Non-Inductive Current Drives	- Relaxation of fatigue concerns
Liquid Metal MHD Evaluations	- Boundary layer heat transfer problem at first wall
Implanted Tritium Issue	- Problem with potentially high tritium permeation to water coolant in first wall

TABLE 3 MAJOR CHANGES - BLANKET

MAJOR CHANGES - BLANKET	IMPACT ON DESIGN
Solid Breeders	- Use of solid breeders - Interest in water coolant systems - Heat transfer, tritium recovery and breeding issues
Breeder Evaluations	- Dissatisfaction with salts - Safety concerns of liquid lithium - Interest in liquid lithium alloys - Lithium safety reconsidered
Li^7 Cross Section Decrease	- Possible need for multipliers with Li and Li_2O systems
Tritium Permeation Evaluations	- Appreciation for tritium permeation to water coolant - Consideration of permeation barriers
Detailed MHD Evaluations	- Concern for high circulating power requirements for pumping - Appreciation for data limitations

TABLE 4 MAJOR CHANGES - SHIELD

MAJOR CHANGES - SHIELD	IMPACT ON DESIGN
3D Shielding Analysis	- Reduced uncertainty, especially in penetration shielding
Activation Codes	- Better quantification of activation transport, shutdown doses, etc.
Shielding Cost Analysis	- High cost driver; optimization and interest in less expensive materials

TABLE 5 PARAMETERS FOR STARFIRE, DEMO AND MARS

	STARFIRE	DEMO	MARS
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/SS	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. Includes 231 MW of thermal power from plamsa end dump. Direct conversion also produces 324 MWe (not included in the thermal power).

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

TABLE 6 MAJOR DESIGN CONCERNS IN-VESSEL COMPONENTS

Item	Concern	Possible Technology Need
Disruptions	- Material loss or degradation by melting, vaporization - Damage from arcing - Body forces on components	- High current electrical contacts that can be remotely disengaged
Composite structures	- Clad/structure bonding	- Strain isolation interface with high thermal conductivity (e.g. brazed fingers, felt, etc.)
Erosion/Redeposition	- Lifetime in severe environment of high T, high P, high thermal and particle flux	- Use of advanced structural material (e.g. V alloys, Cu-Be, TZM possibly with "compliant" cladding
First wall cooling (liquid metal system) (helium systems)	- High layer thermal gradient due to MHD suppression of mixing - Mechanical integrity of complex geometry (vanes, channels) for flow control	- Flow path convolution or flow mixers - Combination of gas cooled flow technology with advanced material e.g. vanadium)
Tritium permeation	- Implanted tritium from plasma may permeate into coolant	- Use of porous cladding - Tritium permeation barriers

TABLE 7 MAJOR DESIGN CONCERNS - SOLID BREEDER BLANKETS/FIRST WALLS

Item	Concern	Possible Technology Development
Breeding Ratio	- Adequacy (TBR > 1.00+)	
Tritium Recovery Retention	- Prompt recovery of tritium from blanket - Releasable inventory	
Use of Multipliers - Be	- Cost, mechanical integrity	- Materials and fabrication development of water-cooled Be multiplier
	- Toxicity of Be	- Materials and fabrication development of water-cooled Be multipler
Heat Transfer	- Control of breeder temperature	- Breeder/structure thermal bonding method Fabrication of breeder forms with adequate porosity, conductivity and strength
Mechanical Integrity/ Compatibility	- Cracking - thermal, stress impacts	- Fabrication of breeder forms with adequate porosity, conductivity and strength
	- Li_2O - impurities, loss of strength, creep, corrosion	
Use of H_2O Coolant	- Water leak into blanket	- Breach prevention method, e.g. double jacketed cooling system
	- Tritium permeation into blanket coolant	- Breach prevention method, e.g. double jacketed cooling system - Tritium permeation barriers
Use of He Coolant	- Structural integrity at high temperatures	- High temperature (> 600°C) structural material (e.g. vanadium alloys)
	- Large void volume - neutron streaming	

TABLE 8 MAJOR DESIGN CONCERNS - FLOWING LIQUID METAL BLANKETS/FIRST WALLS

Item	Concern	Possible Technology Development
MHD Effects	- Large pump power	- Insulating walls
Corrosion/Compatibility	- Material Degradation - Activation transport	- Coatings, coolant additives
Tritium Extraction - Li	- Processing large volumes	- Adequate large scale extraction
Tritium Containment -LiPb	- Permeation driven by high partial pressure of tritium	- Double containment systems
Reactivity/Safety	- Tritium release	- Double containment systems
	- Consequences of a lithium fire	- Double containment systems
Weight - 17Li83Pb	- Structural support	

FWBS TECHNOLOGY
RELATED U.S. PROGRAMS/ACTIVITIES

FIRST SURFACE

- PLASMA EDGE STUDIES
- MATERIALS (PMI)
- HIGH HEAT FLUX COMP. DEV. PROG
- FWBS PROGRAM (I)

FIRST WALL

- FWBS PROGRAM (I)
- MATERIALS (ADIP, SPM)

BLANKET

- FWBS PROGRAM (II)
- TRIO
- BREEDER NEUTRONICS
- MATERIALS (ADIP, SPM)
- TSTA
- LBM EXPERIMENT

SHIELD

- SHIELDING EXPERIMENTS

DESIGN STUDIES

NEUTRONICS

FWBS PROGRAM (III, IV)

SAFETY PROGRAM

Abbreviations: PMI - Plasma Materials Interaction

ADIP - Alloy Development for Irradiation Performance

SPM - Special Purpose Materials

FWBS Program Elements

I - Surface Heating

II - Blanket Simulations

III - Electromagnetic Effects

IV - Remote Handling

Figure 1. U.S. programs and activites directly related to the development of first wall, blanket and shield technology.