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PRELIMINARY CONCEPTUAL DESIGN STUDY OF A
SUPPRESSED-FISSION TOKAMAK HYBRID

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Lawrence
Livermore
Laboratory

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Preliminary Conceptual Design Study of a Suppressed-Fission Tokamak Hybrid

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ABSTRACT

A preliminary design concept for a commercial-size tokamak fusion breeder with a suppressed fission blanket and emphasis on ^{233}U breeding has been formulated. The design is based upon a similar tandem mirror hybrid concept and addresses particular concerns relating to the use of a tokamak for the suppressed fission blanket application.

The single most important departure from the tandem mirror reference blanket concept is the substitution of FLIBE for the liquid lithium used for cooling and in-situ tritium breeding. A concern for excessive MHD-related problems drove the decision to replace the more highly conductive lithium. As a result of the new coolant selection, material compatibility issues mandated changes in the composition of the mobile fuel pellets. In addition, the higher operating temperatures associated with the FLIBE placed more stringent constraints on structural requirements and reduced several design margins. Neutronics analyses predicted relatively poor blanket performance with tritium breeding of 1.02 and fissile ^{233}U breeding of 0.34.

Consequently, the most significant conclusion of this effort is that FLIBE has inherently poor neutronics characteristics when applied to the suppressed fission hybrid option, and therefore, an alternate coolant must be chosen. Future efforts towards the development of an attractive suppressed fission tokamak hybrid will concentrate on the feasibility of helium and/or liquid metal coolants. Included in these efforts will be the investigation of plasma engineering options which hold promise of reducing the magnetic field in the blanket regions.

CONTENTS

	<u>Page</u>
1. INTRODUCTION AND DESIGN OVERVIEW	1
1.1 Goals	1
1.2 Organization	2
1.3 Design Requirements	2
1.4 Design Overview	3
2. REACTOR DESIGN	6
2.1 Fusion Driver System	6
2.1.1 Configuration	6
2.1.2 Magnet Systems	9
2.1.3 Plasma Systems	12
2.1.4 Vacuum Vessel	18
2.2 Mechanical Design	18
2.2.1 Introduction and Design Guidelines	18
2.2.2 Blanket Arrangement	20
2.2.3 Blanket Materials	22
2.2.4 First Wall	22
2.2.5 Fuel Zone	25
2.2.6 Divertor	26
2.2.7 Module Comparison and Assembly Considerations	26
2.3 Heat Transfer and Fluid Mechanics	28
2.3.1 Coolant Selection	28
2.3.2 First Wall Cooling	28
2.3.3 Blanket Cooling	30
2.3.4 Issues	34
2.4 Nuclear Performance	34
2.5 Materials Compatibility	35
2.5.1 Overview	35
2.5.2 Compatibility Problems Related to LiF-BeF ₂	38
2.5.3 Solid Breeder Materials	41
2.5.4 Summary of Issues	42
3. ADDITIONAL RECOMMENDATIONS FOR FY83 WORK	44

FIGURES

	<u>Page</u>
2-1 Cross-Section Elevation View of Preliminary Suppressed Fission Blanket Concept for a Tokamak Reactor	7
2-2 Radial Build-Up of Tokamak Hybrid	8
2-3 Typical Blanket Module Cross-Section	21
2-4 Outer Blanket/Shield Region	23
2-5 Inner Blanket/Shield Region	24
2-6 FLIBE Viscosity	29
2-7 Cylindrical Model - Geometry and Composition	36
2-8a Beryllium/Thorium Oxide Fuel Form	37
2-8b Heterogeneous Model of Beryllium/Thorium Oxide Fuel Form	37

TABLES

	<u>Page</u>
1-1 Differences Between the Tokamak and TMHR Reference Blankets	4
2-1 Tokamak Breeder Magnet Systems	10
2-2 Comparison of Principal Parameters of Tokamak Fusion Reactors	13
2-3 Key Design Guidelines and Parameters for the Suppressed Fission Tokamak Blanket Concept	19
2-4 FLIBE Coolant Parameters for First Wall Tubes	31
2-5 First Wall Tube Dimensions and Temperatures	32
2-6 FLIBE Cooled Tokamak Blanket Design Temperatures	33

1. INTRODUCTION AND DESIGN OVERVIEW

1.1 GOALS

Several goals were identified for this design effort. Primary among these was the development of a preliminary point design concept for a tokamak fission suppressed fusion breeder. As a starting point, an adaptation of the FY82 reference blanket concept for the tandem mirror hybrid reactor⁽¹⁾ was investigated. Several special concerns which were identified as unique to the tokamak driver case included (1) geometric constraints on the use of mobile fuel; (2) flow of coolants in high magnetic field regions; (3) high thermal wall loading; and (4) reduction in blanket coverage by a magnetic divertor and other wall penetrations. With these and other less critical concerns identified, the reference tandem mirror blanket concept was modified by FLIBE cooling and fit around the torus of a commercial scale tokamak driver. The resulting preliminary design concept is presented.

A second objective of this effort was the identification of major design issues. The current design concept is a limited budget, first attempt at the conceptual design of a suppressed fission tokamak. Assumptions have been made as to the fabricability and integration of this device, the compatibility of the coolant, breeder and structural materials, and the physics performance of the basic tokamak driver. Consequently, each section of this document includes a discussion of the major design issues and offers some alternatives for future consideration.

One final goal of this project was the development of recommendations for future investigations. Several recent physics innovations hold promise for more compact, less complicated and higher breeding efficiency hybrid concepts. In addition, a few design concepts which more radically modify the basic breeding blanket design have been suggested for future consideration.

1.2 ORGANIZATION

This report is presented in three major sections. Section 1 provides an introduction which describes the program goals and briefly reviews the design concept.

Section 2, Reactor Design, is divided into five major subsections. The first is a discussion of the tokamak driver. Topics included in Section 2.1 include the general configuration of the machine, the plasma subsystems and the magnet subsystems. Subsection 2.2 contains a detailed discussion of the mechanical design. Particular emphasis has been placed on the first wall and blanket design concepts. In subsection 2.3, the heat transfer and fluid mechanics analyses performed to develop a feasible first wall and packed bed blanket concept are described. The critical analyses which ultimately led to the selection of a lithium-beryllium fluoride salt (FLIBE) coolant are provided. Subsection 2.4 addresses an initial evaluation of the nuclear performance of the FLIBE cooled, packed bed blanket. Unfortunately, this analysis indicates a substantial reduction in breeding relative to the liquid metal coolant used for the tandem mirror hybrid blanket. Subsection 2.5 addresses the varied materials compatibility issues arising from this design. A detailed discussion of the interactions of the FLIBE coolant, the HT-9 structure and the Be/ThO₂ breeding pebbles is presented.

The third section contains recommendations for near term investigations.

1.3 DESIGN REQUIREMENTS

Although fission product waste and decay afterheat handling problems are minimized with the fission suppressed blanket concept, neutron economy becomes a critical design constraint. In order to be attractive the tokamak fission suppressed hybrid concept should have a tritium breeding ratio of slightly greater than one and a fissile fuel breeding ratio of 0.6 or better. In the ideal case, that requirement translates into one fusion neutron being multiplied to about 1.65 neutrons. Streaming losses and parasitic neutron captures require that the multiplication be even higher. Clearly the design concept must maximize neutron multiplication and conservation and minimize any loss mechanism.

A second requirement is that the design must also allow for the generation and efficient collection of thermal energy. These requirements are difficult to satisfy in the tokamak geometry. On one hand, the design must accommodate all of the plasma and magnetics subsystems, both functionally and geometrically, necessary to produce the fusion power. On the other hand, the blanket breeding concept and configuration must maximize neutron multiplication and utilization. These two objectives are in conflict and compromises are required.

1.4 DESIGN OVERVIEW

As mentioned, the design effort began with the liquid metal cooled tandem mirror blanket concept as a model⁽¹⁾. Very early in the design study, it became apparent that MHD pressure drops resulting from the higher magnetic fields and complex geometry of the tokamak would mandate a change in coolants to avoid the need for structural thicknesses which would be neutronically prohibitive. The liquid lithium employed in the tandem mirror design is far too conductive. FLIBE (LiF-BeF_2) was chosen as the tokamak coolant/tritium breeding material and its incorporation resulted in further deviation from the reference blanket model. Table 1-1 summarizes this and other differences between the tokamak and the tandem mirror reference blanket designs. The remainder of this section is a brief overview of the tokamak fission suppressed hybrid reactor.

Major parameters of the fusion driver are $R = 8.2$ m, $a = 2.4$ m, $B_t = 4.8$ T, maximum B at coils = 10 T, $\langle \beta \rangle = 0.06$, plasma current = 11 mA. Steady-state current drive is assumed using 1.5 MeV negative ion based neutral beams, which permit a fusion energy amplification, Q_p , of about 15.⁽²⁾ The total wall area required for 200 MW of neutral beam injection, for pellet injection and for plasma monitors is only 1% of the torus wall area.

TABLE 1-1. Differences Between the Tokamak and
TMRR Reference Blankets

Differences	Reason
FLIBE coolant vs. lithium	Higher magnetic field, but maintains in-situ tritium breeding
Be/ThO ₂ vs. Be/Th fuel pebbles	FLIBE/Be compatibility with bred uranium
Separately cooled first wall	Higher wall loading and su. face heat flux and poor FLIBE thermal conductivity
Irregular shaped modules	toroidal geometry
More complicated blanket piping	Limited access to inboard blanket, longer coolant paths, provision for separately cooled first wall and poloidal divertor

The first wall is constructed of steel tubes oriented in the poloidal direction and containing a separate FLIBE coolant, also at about 500°C. Because of the heat removal limitations of FLIBE, the non-neutron thermal wall loading is kept below 25 W/cm², corresponding to a neutron wall loading of 2.2 MW/m² when about 70% of the non-nuclear power is removed by a divertor. Consequently, the reactor size is somewhat larger than STARFIRE, which has a comparable fusion power rating (= 3000 MW), but a higher neutron wall loading (3.6 MW/m²). The 25 W/cm² limit is somewhat arbitrary and should be explored further in future work.

For disposal of particles and thermal outflux from the plasma, an INTOR type single-null poloidal divertor is included at the bottom of the torus. The inboard and outboard blankets are extended to cover the bottom of the plasma chamber, leaving a 1-m wide slot through which the diverted plasma flows into the pumping chamber. The inboard blanket is extended underneath the pumping chamber to capture neutrons that penetrate the divertor hardware. By these methods the effective blanket coverage can be greater than 0.9.

The blanket uses a mobile fuel of Be/ThO₂ composite pebbles. The volume ratio of Be to ThO₂ is approximately 15 to 1. As mentioned above, anticipated difficulties in using a liquid metal coolant in the high-field regions led to the choice of a molten fluoride salt (LiF-BeF₂) with a melting point near 460°C as the primary coolant and tritium breeder. The pebbles are introduced from the top of the reactor and flow down the inboard and outboard blankets to outlet ports at the bottom. The blanket structural material is HT-9 and the maximum temperature is about 560°C. The magnetic interaction associated with the use of ferritic steel in the tokamak blanket was not considered.

*The 21 W/cm² includes a contribution due to injected neutral beam power.

2. REACTOR DESIGN

2.1 FUSION DRIVER SYSTEM

2.1.1 Configuration

Figure 2-1 shows a schematic elevation view and Figure 2-2 shows the radial build-up of the tokamak breeder. This design combines general features of INTOR and of STARFIRE. There are 12 toroidal-field (TF) coils of modified D shape, with 24 separate vacuum vessel and blanket sectors. The radial thickness of the blanket segments carrying the fertile material is about 60 cm, while the thickness of the shield immediately behind the multiplying-breeding region is about 80 cm.

An INTOR-type single-null poloidal divertor is used for particle control and heat removal from the plasma. Nearly complete blanket coverage is achieved by extending the outboard blanket inward to the edge of the divertor throat, as shown in Figure 2-1. The inboard blanket and first wall are continued underneath the divertor neutralizer plates to capture neutrons penetrating the divertor chamber.

The 5-m diameter central throat of the tokamak contains an ohmic-heating solenoid with sufficient capacity to initiate the plasma current. The current is then sustained in the steady-state by 1.5 MeV deuteron beams. The total duct area required for neutral beam injection, pellet fuel injection and ECRH (used during start-up) is less than 1% of the torus first wall area.

The EF coils are located externally to the TF coils. Another set of 3 coils located below the divertor chamber and external to the TF coils are required to help shape the divertor flux surfaces. All coils are superconducting except for some low-powered control coils that are located near the vacuum vessel.

Figure 2-2. Radial Build-up of Tokamak Hybrid

2.1.2 Magnet Systems

a. Superconducting vs. Resistive TF Coils

During the early part of the FY82 work, a trade study was performed comparing superconducting and resistive TF coils for the commercial tokamak breeder. The conclusion of this study was that superconducting coils should be selected. The following is a summary of the reasons:

Impact on Reactor Cost. The larger capital cost of the reactor when using superconducting TF coils was traded against the cost of the extra power dissipation incurred by resistive coils (325 MW using massive coils). The trade depends on the value of electricity, but the difference was found always to be a minor cost driver - the effect on the annual reactor costs (payback and operations) being at most 15% and probably much less.

Time Scale for Development. SC (superconducting) coils for tokamak reactors are beyond the current state of the art. However, on the multi-decade time scale for introducing commercial fusion breeders, one can expect that large SC coils will be sufficiently well developed on test stands and elsewhere so that they can be fabricated confidently for reliable operation. Furthermore, after several dozen identical SC coils have been fabricated (eg., after the first 3 breeders), the cost per coil should be significantly less than for the coils of the first breeder.

Existing Conceptual Designs. The fusion breeder can make use of existing conceptual tokamak reactor designs using SC coils, such as STARFIRE or the HFCTR.* Literally millions of dollars have been spent on such design studies, while no design of a commercial sized tokamak reactor with resistive magnets has been carried out.

b. Toroidal Field Coils

Table 2-1 gives the principal parameters of the TF coils and other magnet systems of the tokamak breeder.

* High field compact tokamak reactor

TABLE 2-1. Tokamak Breeder Magnet Systems

<u>Toroidal Field Coils</u>	
Conductor	NbTi (superconductor)
Coolant	Liquid helium, 3°K
Number of TF coils	12
Max. B at conductor	10.1 T (R = 3.9 m)
Max. B on plasma axis	4.8 T (R = 8.2 m)
TF ripple at plasma edge	0.5% peak-to-average
TF coil shape	Modified D
Horizontal bore	9.2 m
Vertical bore	16 m
Radial build	1.4 m inboard, 1.7 m outboard
Width	2.4 m outboard, nested inboard
<u>Poloidal Field Coils</u>	
Type	NbTi (superconducting)
Coolant	Liquid He, pool-boiled for external coils
Number	8 external, three dedicated to single-null divertor
<u>Ohmic Heating Solenoid</u>	
Type	NbTi (superconducting)
Coolant	Liquid He, pool-boiled
Max. B in solenoid	± 8 T (double swing)
Max. flux swing	190 Wb

c. Poloidal - Field Coils

All of the PF coils will be superconducting and have been located external to the TF coils for ease of maintenance and replacement in the event of failure. This configuration has the additional benefit that in the event of failure of a TF coil, repair or replacement does not involve the dismantling of internal PF coils. External placement of the three divertor-dedicated PF coils, in particular, allows a relaxation of some of the space constraints associated with neutron shielding requirements. As the TF coils must be shielded regardless of the PF coil location, external placement of the PF coils provides for adequate internal space to shield both the TF and PF coils.

The PF coils maintain a D-shaped plasma with considerable triangularity as indicated in Figure 2-1. The five outer PF coils provide most of the vertical field necessary for equilibrium positioning of the plasma column. The remaining three divertor-dedicated PF coils are vital for creating the magnetic separatrix, outside of which magnetic field lines are "diverted" into the divertor chamber. This PF coil subset provides the required magnetic field lines for impurity control in the tokamak driver.

d. Ohmic-Heating Transformer

The flux swing of 190 Wb that can be delivered by the OH solenoid is more than adequate to provide start-up of the full plasma current (11 MA). The solenoid current is programmed to zero during the first 1000 s of the burn, with the neutral-beam-driven current compensating for the very small reverse emf during this process.

The maximum rate of change of field in the solenoid during start-up is about 1 T/s. This value is thought to be within the state-of-the-art, so that the solenoid could be fabricated without a special R&D program.

To date, there has been no definitive demonstration of neutral-beam current drive (although promising results have been obtained). If beam current drive turns out to be impractical, the OH solenoid can be made capable of driving the plasma current for at least 3 h. At the

operating temperatures and profiles, the loop voltage will be about 0.02V, requiring a flux swing of 220 Wb to sustain the current. A total of 370 Wb is required including 150 Wb for current ramp-up. This capacity can be obtained by using a force-cooled solenoid with Nb₃Sn conductor to permit ± 12 T field swing, and by increasing the diameter of the central throat of the tokamak from 5.1 m to 5.8 m. Assuming 100 s downtime between pulses, the duty factor would be 99%. The chief disadvantage of this operational mode is the aggravation of thermal and mechanical fatigue due to the 2,000 pulses per year.

2.1.3 Plasma Systems

a. Plasma Parameters

The basic design requirement is to generate 3,000 MW of fusion power (the same as in the tandem mirror hybrid) while keeping the non-neutron wall loading sufficiently low to accommodate the heat removal limitations of FLIBE (i.e., ~ 25 W/cm²).

Table 2-2 lists the principal plasma parameters of the tokamak breeder and compares them with STARFIRE and INTOR parameters. Assuming that the poloidal divertor can remove 70% of the thermal flux from the plasma (which has been verified in ASDEX and PDx experiments), the spatially averaged thermal wall loading is 21 W/cm² and the uncollided neutron wall loading is 2.2 MW/m². Consequently, the breeder reactor size is somewhat larger than that of STARFIRE, which has a comparable fusion power rating but higher neutron wall loading. The requirements on magnetic field and beta for the breeder are somewhat relaxed from the STARFIRE values.

The operating regime is sub-ignition ($Q_p = 15$) as suggested by the continuous injection of beam power to drive the plasma current. The heating power supplied by the beams is sufficiently large to eliminate the need to identify and demonstrate the effective thermal stability control that would be required for ignited operation.

TABLE 2-2. Comparison of Principal Parameters of Tokamak Fusion Reactors

	INTOR	STARFIRE	Comm'l Tokamak Breeder
Major radius (m)	5.2	7.0	8.2
Minor radius (m)	1.2	1.94	2.4
Plasma elongation	1.6	1.6	1.7
Max. B at coil (T)	11.0	11.1	10.1
Max. B at plasma axis (T)	5.5	5.8	4.8
Plasma current (MA)	6.4	10.1	11
N beam energy (keV)	175	RF heating	1,500
Heating power (MW)	75 (start up)	90 (contin.)	200 (contin.)
Impurity control	Single-null pol. divertor	Pumped limiter	Single-null pol. divertor
$\langle \beta \rangle$ = plasma pres./magnetic field pres.	0.056	0.067	0.06
Pulse length (s)	200	steady state	steady state
Fusion gain, Q_p	ignited	39	15
Fusion power (MW)	620	3,500	3,000
Uncoill. neutron wall loading (MW/m ²) ^a	1.3	3.6	2.2
Non-neutron thermal wall loading (W/cm ²) ^a	10	70	21 (28 max)
Thermal power (MW)	-	4,000	5,100
Power consumpt. (MWe)	240	450	550
Net elect. power (MWe) ^a	-	1,200	1,600
Net plant effic.	-	0.30	~ .30

a) Average quantities

b. Heating Systems

Neutral-beam injection has been chosen as the primary plasma heating system because the largest plasma temperatures and β -values to date have been obtained with neutral beams, and theoretical analyses indicate that the same beam systems can be used for plasma heating and current drive.

The specified beams are 1.5 MeV D^0 beams formed from D^- ions. Between 60 and 100 MW of injected power is required for heating the plasma to ignition conditions, but 200 MW of injection is needed to drive the total plasma current. The injected beam current of 140 A-equivalent utilizes 3 beam ducts, each 0.4 m wide and 1.0 m tall and located in the outboard midplane. The beam orientation is 35° normal to the outer plasma surface.

Neutral-Beam Technology. The ion beams would be obtained from self-extracting surface conversion D^- sources. In development work at LBL, such sources have to date produced more than 1 A of D^- current in multi-second pulses (this source was 0.25 m in length which gives 4 A/m), compared with about 15 A (6 A/m) that would be required of each ion source for the 1.5 MeV beams. The accelerated D^- beams would be neutralized by photo-detachment using an oxygen-iodine chemical laser such as is under development by the AFWL. The laser efficiency, anticipated to be $>7\%$ is unknown. A neutralization efficiency of 97% is expected, but a great uncertainty concerns the ion beam accelerator. Electric quadrupole focused RF accelerators are well suited to the requirement, but such accelerators have power efficiency of only about 35%. It appears feasible to develop electrostatic quadrupole accelerators for the 1.5-MeV range, and such accelerators would have a power efficiency of 80%. In this study we assume that the electrostatic accelerator can be developed, and that the overall injector efficiency will be 60% (333 MW electric power). A transverse field focusing accelerator being developed at LBL should give improved efficiency and put the ion source out of the direct line-of-sight from neutrons, making hands-on maintenance possible.

Electron Cyclotron Heating (ECRH). The only other non-ohmic heating system required is 3 MW of ECRH power at 120 GHz that would be used for preionization and electron heating at the beginning of each pulse. This power can be supplied by gyrotrons which are currently under development.

c. Operational Scenario

The TF coils are operated at constant current, while the currents in the EF and divertor coils undergo large changes at the beginning and end of the pulse. Neutral-beam current drive allows the pulse to be arbitrarily long, but in practice it may be limited to tens of hours because of the need for maintenance or some other regularly scheduled activity.

The plasma current, I_p , density, n_e , and temperature, T_e or T_i , are increased by a series of processes of total duration approximately 20 s. The flux linkage from the OH solenoid and the EF coil system is adequate to supply the requirements for full current start-up (11 MA). However, to reduce the initial loop voltage, about 3 MW of ECRH is used to preionize the filling gas and heat the electrons to a temperature between 10 and 50 eV. The OH solenoid is then activated; full current is reached over a period of 12 s while T_e is raised to about 1 keV and $\bar{n}_e$ is increased to about 10^{14} cm^{-3} .

This density is sufficiently large to trap the 1.5 MeV D^0 neutral beams. The beams are activated one by one over the next 5 to 10 s. Together with fusion alphas that are produced in increasing amounts as T_i approaches and exceeds 10 keV, the beams heat the plasma electrons and bulk ions to their operating temperatures. The rate of change of flux in the OH solenoid is reduced as the injected fast ions carry an increasing portion of the plasma current.

The plasma is fueled by continuous injection of frozen deuterium and tritium pellets. During the burn, the beam power and pellet injection rates are varied in order to keep the plasma current and fusion neutron output at the required levels. Particle pumping and plasma heat removal are accomplished by the magnetic divertor.

d. Current Drive

Steady-state current drive is to be accomplished by the 200 MW of 1.5 MeV D^0 beams. The STARFIRE design specified RF power at the lower hybrid frequency for current drive. However, stability concerns related to surface plasma current generation appear to disqualify lower hybrid heating. In the case of 1.5 MeV beams, adequate penetration of the plasma

is assured and the interaction with the plasma is well understood. The modest current drive efficiency needed (0.06 A/W) is expected to be readily attainable.

As noted previously, the neutral beams will also provide thermal stability control of the plasma. Furthermore, current profile control by the fast ions is expected to insure stable operation at the required β of 6%.

Full current drive by the beams requires that $Z_{\text{eff}} \sim 1.5$. While this level of impurity content ought to be prevalent because of fusion alphas and a certain amount of sputtering, it may be necessary to inject pellets of frozen high-Z gas (such as xenon) to insure the required Z_{eff} in the center of the plasma

e. Impurity and Particle Control

For disposal of particles and thermal outflux from the plasma, a single-null poloidal divertor is included at the bottom of the torus (see Figure 2-1). The magnetically diverted plasma flows into the pumping chamber through a 1-m wide slot between the inboard and extended outboard blankets.

A poloidal divertor is chosen rather than a bundle divertor because the effectiveness of both single-null and double-null poloidal divertors has been demonstrated successfully in the large PDX, ASDEX and DOUBLET III tokamaks, while the performance of the bundle divertor is still in doubt. Extensive analyses carried out in the INTOR and FED projects showed that a pumped limiter, the only option to a divertor, would have less certain long-term performance.

In-torus components such as the divertor baffle and neutralizer surfaces must be protected from excessive plasma sputtering and the effects of a major plasma current disruption (in which the plasma current disappears in a fraction of a second and the plasma energy is dumped). Exhaustive evaluations of candidate materials were carried out in the INTOR program in 1981 and 1982, comparing sputtering, thermo-mechanical properties, heat conduction, surface melting and other characteristics.

It was concluded that Be and BeO are the preferred materials to serve as protective coverings for the first wall, baffles and divertor collector plates. The beryllium neutron multiplication is expected to partially compensate for neutron losses in the divertor hardware. The thickness appropriate for this application will probably be no more than 1 cm. A surface heat load $\sim 1 \text{ kW/cm}^2$ is expected. In the tokamak breeder design, protective beryllium coating of the first wall is not required except at the bottom of the torus near the divertor where the particle flux is high. Beryllium coverage will also be used on the baffle surfaces defining the throat of the divertor chamber and on the neutralizer plates in the divertor chamber itself.

Two significant issues must be investigated if the beryllium course is to be pursued. Attachment of the beryllium to the various heat removal surfaces is a materials problem in itself and it is further complicated by the environment in which the attachment must successfully function. A second problem arises from the swelling properties of beryllium. The lifetime of beryllium tiles may be controlled by internal stresses which can build as a result of differential neutron swelling and the ΔT across a beryllium tile. As a result, methods of fabrication, attachment, and accessibility will be important considerations. Provisions for periodic replacement of the tiles and servicing of the divertor modules will be necessary because of surface erosion as well as the above two reasons. An ~ 1 year service life is anticipated.

Based on present day tokamak experiments in PDX and ASDEX, the divertor is expected to remove up to 75% of the thermal (non-neutron) power flowing from the plasma. All particle pumping is carried out by the divertor except for a tiny fraction that is performed by the 3 beam injector ducts. The divertor chamber is comprised of 24 modules each with its own vacuum pumping system. Each module is removed by radial extraction when the internal components must be replaced because of severe erosion or swelling failure.

2.1.4 Vacuum Vessel

There is no vacuum vessel per se. Rather, the first wall coolant tubes are attached to each of the 24 blanket/divertor sectors and the vacuum seal welds will be made at the outside of each sector. This configuration provides the advantage of modularity with a minimum number of welds near the plasma. The key issues, which must be resolved in a more detailed design are access to remove the sectors and fabricability.

The connections between the sectors are designed to give a vessel resistance in the toroidal direction of $0.1 \text{ m}\Omega$. This value permits full start-up of the plasma current in 20 s. This resistance is sufficient to prevent significant damage to the vessel in the event of a major plasma disruption. Vacuum pumping is carried out by a duct attached to each of the 24 sectors. Replacement of a divertor module or of a complete vessel sector is accomplished by radial (and in half the cases - rotational) translation between adjacent TF coils.

2.2 MECHANICAL DESIGN

2.2.1 Introduction and Design Guidelines

The fission suppressed blanket concept for a tokamak reactor is a single zone direct cooled blanket. The blanket design evolved in accordance with a set of initial design guidelines. These guidelines and some of the blanket parameters are presented in Table 2-3. Because the limited amount of time allotted to this task did not permit extensive iteration of the design, considerations such as remote maintenance and service, structural material optimization and thermal hydraulic and neutronic performance were not optimized. The component arrangement and configurations, to the extent possible, do attempt to address those considerations.

TABLE 2-3 Key Design Guidelines and Parameters for the
Suppressed Fission Tokamak Blanket Concept

Fusion power	3000 MW
Major radius of device	8.0 m
Plasma size and shape	
Major radius	8.2 m
Minor radius	2.4 m
Elongation	1.7
First wall neutron loading	2.2 MW/m ²
First wall surface heat flux	0.21 MW/m ²
Number of toroidal field coils	12
First wall coolant pressure	750 psi (in tubes)
Blanket coolant pressure	<75 psi
Magnetic field on axis	4.8 Tesla
Blanket/first wall structure	Ferritic steel (HT-9)
Maximum structural temperature	575°C
Blanket coolant	FLIBE
Fuel form	Composite Be/ThO ₂ pebbles (spheres)
Sphere diameter	~ 3 cm
Fuel processing	Continuous or batch
Blanket thickness	~ 60 cm
Shield thickness	~ 80 cm
Toroidal field coil size	
Horizontal bore	9.2 m
Vertical bore	16 m
First wall configuration	Tubular wall 1.5 cm diameter

2.2.2 Blanket Arrangement

The overall tokamak blanket arrangement is shown in Figure 2-1, which is a cross-section elevation view of the reactor. The relationship between the blanket and shield relative to the D-shaped plasma and modified D-shaped toroidal field (TF) coils is shown. In this blanket concept, the blanket is divided into twenty-four equal sized sectors (a number compatible with the 12 TF coils). When assembled, these form a complete torus with seals between the sectors to isolate the interior vacuum environment from the outside. This precludes the need for a separate vacuum vessel.

The first wall and blanket are cooled by FLIBE (which also serves as the tritium breeding medium). The first wall coolant is directed poloidally through narrow tubes (see Section 2.3). The blanket coolant enters from a supply manifold at the outer edge of the blanket (furthest from the plasma), is directed through a narrow duct to the front of the blanket, and flows outward through a fucled region containing the composite Be/ThO₂ spheres. Heat transfer considerations dictated that three passes of the coolant through the bed be utilized. Therefore, inner baffling and flow channels are required within the blanket. This cooling arrangement is discussed in more detail later.

A cross section of a typical sector is shown in Figure 2-3, a view of the blanket at the horizontal centerline (Section A-A of Figure 2-1). Only one of the twenty-four sectors are shown. The difference in widths of the outer region of the blanket (outboard of the plasma) and the inner region (inboard of the plasma between the plasma and the inner leg of the TF coils) can clearly be seen. The width of the blanket in the toroidal direction varies from ~ 3 m to ~ 1.5 m for the outer and inner regions of the blanket, respectively. Figure 2-1 shows that the outer blanket region extends from the centerline of the plasma clockwise around the plasma to the top of the divertor. The inner region extends from the top of the plasma, counterclockwise along the inside of the plasma. In order to absorb any neutrons which leak through the divertor, the inside blanket is extended further to fully cover the bottom of the divertor.

FIGURE 2-3. Typical Blanket Module Cross-Section (Section A-A of Figure 2-1)

2.2.3 Blanket Materials

FLIBE was considered as a blanket coolant for two reasons. First the use of liquid metals would lead to prohibitively high coolant pressures, high pumping power, and poor neutronic performance resulting from the structure necessary to contain the pressures. Secondly, the FLIBE serves as a medium for breeding tritium. The use of composite Be/ThO₂ spheres for breeding fissile fuel is a natural extension of the tandem mirror reference design. The primary difference is that thorium oxide is required in this application (rather than thorium metal) because of its superior compatibility with the FLIBE. Similarly HT-9, also the candidate mirror blanket material, was selected because of its operating temperature regime and compatibility with the FLIBE coolant. Compatibility considerations are discussed further in Section 2.5.

2.2.4 First Wall

As discussed earlier, the first wall is of a tubular construction. The first wall is divided into two halves -- one cools the inner blanket and the other, the outer region. Both first wall coolant loops are supplied by circular manifolds at the top of the blanket. The FLIBE cooled tubes penetrate the shield and follow the contour of the plasma until they reach the divertor. At that point, the outer tubes curve outward and follow the contour of the top of the divertor. These tubes terminate with a circular manifold which collects the first wall coolant. The inner blanket tubular first wall similarly follows the shape of the inner edge of the plasma, however, the tubes extend below the divertor and exit near the pebble outlet. Another option in which the first wall coolant is mixed with the main blanket coolant near the bottom of the divertor was also considered. The tubular first wall is attached to the blanket structure behind to provide 24 integral first wall/blanket/shield assemblies. This arrangement is shown in Figures 2-4 and 2-5, enlarged cross-sections through the outer and inner blanket regions, respectively.

The divertor region is cooled separately with high velocity water in copper swirl tubes. The details of this arrangement have not yet been considered in detail.

FIGURE 2-4. Outer Blanket/Shield Region (Section A-A enlarged)

FIGURE 2-5. Inner Blanket/Shield Region (Section A-A enlarged)

2.2.5 Fuel Zone

Outer Blanket Region. The fuel zone of the blanket, together with the blanket coolant concept are shown in Figures 2-4 and 2-5. As shown in Figure 2-4, the blanket coolant enters the blanket from pipes extending from a poloidally oriented circular manifold at the bottom of the figure. (The inlet manifold from the outer blanket is not shown in Figure 2-1 because the view is through the center of a blanket segment.)

The blanket coolant flow enters through a plenum consisting of the wall of one side of the blanket and another parallel plate to provide a coolant path to the inside of the blanket just behind the first wall. At the first wall the coolant turns $\approx 90^\circ$ to flow toroidally through the inner plenum for about a third of the width of the blanket. As the flow proceeds through this plenum, it is bled off through holes in the outer wall of the plenum and enters the fuel zone. In the fuel zone it flows radially outward 60 cm and discharges into an outer plenum at the back (outside) of the blanket. The flow is collected in the plenum, flows toroidally, and is directed into another parallel plate radial flow channel, and flows inward toward the first wall. This velocity multiplying flow process is repeated until three passes through the blanket are completed. The flow is then channeled through a series of radial pipes connecting the outside of the blanket to a coolant outlet manifold (shown at the top of the figure). The outlet manifold, similar to the inlet manifold, runs toroidally around the outside of the blanket. The outlet manifold, shown in Figure 2-1, terminates near the top of the divertor and discharges the coolant into a circular manifold located above the divertor. The composite fueled spheres for the outer region of the blanket are loaded through the port(s) shown at the top of the blanket. The fuel is removed through the port(s) just above the vacuum duct shown in the figure. The shielding outboard of the blanket (between the blanket and TF coils) can be seen in these figures.

Inner Blanket Region. The inner blanket region coolant arrangement (Figure 2-5) is similar to that for the outer region, except that the blanket segments, as mentioned earlier, are narrower in width. In Figure 2-1, only the outlet manifold is shown on the inboard blanket on the left side of

the figure. The coolant collected in the outlet manifold is discharged into a circular manifold near the fuel discharge under the divertor

2.2.6 Divertor

The divertor as shown is a scaled-up version of the INTOR divertor and reflects some effort devoted to modifying its design concept. The throat of the divertor has been narrowed to 1 m in width. Structure required to define this aperture cannot be adequately actively cooled. As a result, these structures will be fabricated of high temperature materials (Be or BeO) which can be radiatively cooled by the first wall and cooled portions of the divertor. Use of the beryllium will also provide for enhanced neutron multiplication in the divertor region where neutron utilization will be a critical performance parameter.

The vacuum pumping duct has been inclined from the original horizontal orientation to both permit more complete blanket coverage (i.e., neutron utilization) and facilitate flow of the Be/ThO₂ composite spheres through the blanket.

2.2.7 Module Comparison and Assembly Considerations

A first wall/blanket/shield assembly module which surrounds the plasma is wedge shaped to conform to the toroidal shape of the plasma when all modules are assembled. A cross-section through a typical sector is shown in Figure 2-3. For assembly, removal and maintenance of the modules, the torus is divided into equal size tapered sections as shown. The reactor contains 12 TF coils, two of which are shown in the figure. It is anticipated that the TF coils would remain in place for the life of the machine, or at least not require as frequent replacement as the first wall/blanket assembly. Therefore, the module sectors must be narrow enough to be extracted through the available space between the coils. Since a twelve sector design would not permit removal of the sector without interference with the outer leg of the TF coils, more smaller sectors are required. As can be seen in the figure, with 24 sectors, only one of which is shown, there is adequate space to permit a sector to be extracted between coils. Every other sector is centered between coils and can be radially extracted. Since the remainder of

the sectors are located directly under TF coils, removal of these first requires removal of a sector between coils. The adjacent sector under the coil then has to be translated into the opening left by the coil which was removed and then extracted radially. A 12 sector arrangement that would eliminate a need for the translation should also be considered.

2.2.8 Design Issues and Possible Alternate Concepts

The blanket concepts have not been examined in sufficient depth to identify many of the design issues. The divertor, required to maintain reasonable heat flux at the first wall, complicates the design (as compared to the tandem mirror design). The design of seals between the divertor and the blanket will also need investigation. Seals between modules have not been investigated, but probably are a lesser concern than module/divertor interface seals. The design will have to be examined to determine that the configuration has the proper shapes designed into the fuel sphere movement paths to assure that the spheres will flow when required. Penetrations will be required in the blanket for the injection of neutral beams and RF heating. Special coolant provisions will be required around the penetrations, but only small penetrations are anticipated, and not much of the blanket coverage is impacted.

With three radially inward coolant flow passages required to achieve the three coolant passes through the blanket (see Section 2.2.5 and Figure 2-4). Some loss of neutron multiplication will result, thereby reducing the blanket effective coverage. An alternate arrangement may be to flow the coolant in radially through the fueled region to the first wall annulus, then outward through the next section of spheres. This could be repeated twice to execute four passes through the spherical bed without any coolant path streaming areas. If the neutronic and thermal performance are satisfactory, modification of the basic concept should be pursued in the next study phase.

2.3 HEAT TRANSFER AND FLUID MECHANICS

2.3.1 Coolant Selection

Three low pressure, high temperature coolants with internal breeding (lithium and two FLIBE coolants, 47% LiF - 53% BeF₂ and 66% LiF - 34% BeF₂) were considered. Because of its high electrical conductivity and the high (10T) magnetic fields, the use of lithium coolant could result in extremely high pressure drops (~ 1500 PSI or ~ 100 ATM).

The 47-53 FLIBE has a lower melting point than the 66-34 FLIBE, but was eliminated because of its high viscosity in the temperature range of interest (450-550°C, see Figure 2-6). High viscosity results in high pumping power and requires high velocities (aggravating the pumping power requirement) to ensure turbulent flow. Turbulent flow is necessary to reduce the film drop temperature (between the steel structure and/or beryllium/thorium fuel pebbles and the FLIBE) which results because of the low thermal conductivity of FLIBE. The laminar film drop temperature for FLIBE with a 30 W/cm² surface flux is over 800°C. A pumping power on the order of 80 MW would be required to cool the first wall with 47-53 FLIBE at a maximum structure temperature of 572°C.

The 66-34 FLIBE chosen for the reference design concept has a melting point of 460°C, requiring a blanket inlet temperature of $\sim 500^\circ\text{C}$. This results in high operating temperatures for the HT-9 ferretic steel structure, which is lifetime limited by creep rupture strength at these temperatures. HT-9 at 575°C has roughly a three full power year lifetime at a design stress of 13 ksi⁽³⁾. This is an issue for future design consideration. It is possible that INCONEL, with its higher operating temperature, range could be used instead of the HT-9, but some consideration must be given to irradiation damage effects.

2.3.2 First Wall Cooling

Cooling the first wall, which has a surface heat flux of 28 W/cm² (direct heat flux of 21 W/cm² plus a 33% safety factor to account for radiant heat load peaking due to high temperatures of some divertor surfaces), will be difficult. As discussed in Section 2.2, the first wall is constructed of 1.5 cm o.d. tubes with FLIBE running through them. Coolant viscosity, allowable structure temperature, and MHD effects

FIGURE 2-6. FLIBE Viscosity

compete with one another, resulting in approximately 16 Mw of pumping power to cool the tubular reactor first wall. The first wall cooling parameters are given in Tables 2-4 and 5. The heat transfer coefficient and viscous pressure drops were calculated using standard pipe flow equations (4,5). MHD pressure drops were calculated using the Hartman equation (6,7).

2.3.3 Blanket Cooling

Cooling the pebble bed with FLIBE (66 mole percent LiF and 34 mole percent BeF₂) does not appear to present any major problems and considerable flexibility is possible in this area. Since preliminary calculations yielded very low pressure drops in the pebble bed, more detailed analysis and parameterizations were not deemed necessary. Viscous pressure drop in the packed bed was calculated using the Ergun equation (8,9). The Hartmann equation for pressure drop for flow perpendicular to a magnetic field (6,7) was used for the MHD pressure drop. Flow is from front (first wall area) to back through the blanket. The coolant parameters in Table 2-4 result in viscous and MHD pressure drops of 3035 Pa and 100 Pa, respectively. Calculating flow in a packed bed with the Hartmann equation is an extremely crude approximation; however, the calculated pressure drop is so low that an increase of two orders of magnitude would have no significant impact on the design.

Temperatures in the packed bed are summarized in Table 2-6. Differences in the inboard and outboard blanket zones were not taken into account in this analysis, but a conservative design point was picked. A single pass of FLIBE coolant through the packed bed would give a film drop temperature of 57°C and a bulk temperature rise of 10°C, however, improved blanket operating point could require a lower film drop temperature (materials considerations). Unfortunately, the film drop varies inversely with the Nusselt number for the packed bed (8,10).

$$Nu = (0.4 Re^{1/2} + 0.2 Re^{2/3}) Pr^4$$

where

Re = Reynolds number, $\rho y d / \mu$

Pr = Prandtl number, $C_p \mu / K$

TABLE 2-4. FLIBE Coolant Parameters for First Wall Tubes

LiF - BeF ₂ composition	Mole % 66-34
Viscosity	.014 Kg/m-s
Velocity	6 m/s
Reynolds number	10,300
Nusselt number	134
Viscous pressure drop	$1.08 \cdot 10^5$ Pa/meter of tube
MHD pressure drop	$1.35 \cdot 10^5$ Pa/meter
Total pressure drop	5.1 MPa
Pumping power	165 w/meter of tube
Total pumping power (energy lost in fluid)	16 Mw

TABLE 2-5. First Wall Tube Dimensions and Temperatures

<u>Tube Dimension</u>	
Radius	6.0 mm
Thickness	1.5 mm
Total length	21.0 m
Length exposed to plasma	17.0 m
<u>Temperature</u>	
FLIBE inlet temperature	500°C
Minimum steel temperature	>500°C
FLIBE outlet temperature	530°C
Film drop temperature	27°C
Maximum steel/FLIBE temperature	557°C
ΔT in steel tube	15°C
Maximum steel temperature	572°C

TABLE 2-6. FLIBE Cooled Tokamak Blanket Design Temperatures

<u>Pebble Bed</u>	
FLIBE inlet temperature	500°C
Minimum steel temperature	>500°C
FLIBE outlet temperature	530°C
Film drop temperature (to pebbles)	57°C
Maximum beryllium surface temperature	<587°C
Film drop temperature (to steel liner)	45°C
Maximum steel temperature	<575°C

Thus increasing the velocity will slightly decrease the film drop temperature and more dramatically decrease the fluid bulk temperature rise. In the current design the fluid is channeled through the blanket three times to achieve a total temperature rise of 30°C which is compatible with the 30°C rise in the first wall tubes. The resulting hottest ball surface temperature is less than 587°C. Lower heat generation rates in the structure result in lower temperatures there, with the maximum structure temperature in the back of the bed predicted to be about 575°C.

2.3.4 Issues

The major thermal issues with this design are the structural operating temperature, the fuel and beryllium temperatures, and temperature fluctuations or areas of stagnation affecting the tubes and packed bed supporting structure. Bulk temperatures have been calculated and appear reasonable (with some chemical compatibility issues - see the following Section 2.5), but a more detailed analysis is required to understand temperature distributions on a smaller scale in the fuel, beryllium multiplier and structure. Also, the next step in the thermal hydraulic analysis of this blanket design would consider the inboard blanket section, outboard blanket section, and the divertor region on an individual basis. Finally, the primary power associated with cooling the first wall tubes and the blanket could be reduced by slowing the flow and allowing a higher bulk fluid temperature. However, the problems associated with a high temperature and possible flow stagnation in the bed are key. The pumping power is not a sensitive performance issue with the hybrid. Therefore, the higher flow, lower temperature regime was selected.

2.4 NUCLEAR PERFORMANCE

The nuclear performance of the FLIBE cooled blanket was estimated in a scoping analysis with the result that the overall breeding performance is substantially lower than that of similarly configured liquid lithium coolant blankets. This discouraging result can be attributed to fluorine inelastic scattering (which competes with the beryllium) and parasitic captures in the fluorine.

Two one dimensional TARTNP Monte Carlo models were used to estimate breeding and to correct for the effects of heterogeneities introduced by the fuel form. The first model, shown in Figure 2-7, predicts a total breeding (i.e., T+F) of 1.60 versus 1.83 for a liquid metal blanket (13% decrease). The second model, shown in Figure 2-8, predicts that the effect of the heterogeneous mixture will adjust the breeding in this ideal case by a factor of $1.90/2.11 = 0.90$. Therefore, the adjusted breeding for 100% coverage would be 1.44 (versus 1.72 for a liquid lithium).

If we assume that 1.1 breeding events will be required to make-up tritium and to compensate for other losses, then the expected net fissile production is 0.34 atoms per fusion. Compared with an expected value of 0.62 for a liquid metal cooled blanket, the FLIBE blanket performs at a 45% disadvantage. This disadvantage is greater than originally expected, and motivates continued study of the ability to use liquid metals in the tokamak (see Section 3).

2.5 MATERIALS COMPATIBILITY

2.5.1 Overview

The blanket concept of the tokamak hybrid reactor uses the molten salt LiF-BeF_2 (composition 66-34 mole %) as the primary reactor coolant and as the medium for breeding tritium. The blanket also contains a solid form of thorium oxide for U-233 production and metallic beryllium for neutron multiplication. Materials questions that require careful consideration in the blanket design include:

- 1) The effect of MHD-induced electric potentials on salt stability and corrosion reactions.
- 2) The compatibility of ferritic steels such as HT-9 with salt in the presence of metallic beryllium.
- 3) The effect of transmutation reactions in the salt on the corrosion of blanket materials
- 4) Solid-solid interactions among breeding, moderating and containment materials.

BED MATERIAL VOLUME FRACTIONS

MATERIAL	VF
ThO ₂ + U-233	0.03 x 0.98 = 0.029
Be	0.57 x 0.98 = 0.55
FLIBE*	0.40 (1% Li 6)
Fe	0.02

*66% LiF + 34% BeF₂

FIGURE 2-7. Cylindrical Model - Geometry and Composition

FIGURE 2-8(a) Beryllium/Thorium Oxide Fuel Form

FIGURE 2-8(b) Heterogeneous Model of Beryllium/Thorium Oxide Fuel Form

The operating temperature of the hybrid is fundamentally important to material compatibility concerns. Although the postulated temperatures ($\leq 600^{\circ}\text{C}$) are actually below those for which the bulk of salt operating experience exists, they are nevertheless higher than those projected for liquid-metal cooled hybrid concepts ($\leq 500^{\circ}\text{C}$). This latter difference inexorably results from the higher melting point of the salt relative to the liquid metals. The most important ramification of the higher operating temperature relates to the interalloying between metallic beryllium and thorium and the stability of barrier coatings that will impede interalloying. Opportunities for coating development at the 600°C temperature level appear much better in the case of thorium than beryllium, given the relative high solid-state diffusion rate of beryllium and the release of neutron-produced helium from the core to surface of the beryllium.

2.5.2 Compatibility Problems Related to LiF-BeF_2

a. Chemical Reactions with Blanket Components. In the molten salt reactor experiment the salt Li_2BeF_4 was used as the secondary coolant, and it was coupled through a heat exchanger to a circulating fertile-fissile fuel salt mixture composed of $\text{LiF-BeF}_2\text{-ThF}_4\text{-UF}_4$ (72-16-12 - <1 mole %). Both salt compositions were shown to be chemically and thermally stable in contact with a reasonably wide range of engineering materials up to at least 700°C . The above fertile-fissile salt mixture appears on first consideration to be ideally suited as a blanket material for a fission/fusion hybrid reactor, since it combines all the nuclides required for the thorium fuel cycle into one homogeneous high temperature fluid. Unfortunately, the incorporation of metallic beryllium in the blanket precludes the use of U^{4+} or U^{3+} containing salt, because the beryllium is a sufficiently strong reductant to reduce virtually all of the uranium ions to insoluble uranium metal. The possibility of coating the beryllium to isolate it from a thorium/uranium bearing salt has been considered, but this possibility appears remote given the extent of internal helium generation in the beryllium.

Eliminating uranium ions from the salt obviously means the elimination of ThF_4 and, in turn, the introduction of thorium only as a separate phase in the blanket. This confers compatibility between the salt and beryllium, but would introduce a serious compatibility problem if the thorium were in metallic form (as in the tandem mirror hybrid⁽¹⁾). A coating or cladding on the thorium would be required to provide a barrier against the fluoridation of the thorium by BeF_2 . Although the possibilities for developing such a barrier appear much better than is the case for beryllium, the approach nevertheless is speculative at the present time. Other forms of solid thorium, either the oxide or the carbide, are compatible with LiF-BeF_2 . Consequently, the oxide form of thorium has been chosen for the current design.

b. FLIBE/Steel Compatibility Considerations. The compatibility of LiF-BeF_2 salt with commonly-used high temperature alloys is a function of the nature and concentration of solute ions carried in the base salt. The degree of dissolution of a given alloy constituent, M, can be controlled chemically by redox reactions such as

where the $[\text{Ce}^{4+}]:[\text{Ce}^{3+}]$ concentration ratio in the salt is maintained at a preset level. In the tokamak hybrid blanket design, the redox reaction

will be the controlling corrosion reaction, and the dissolution of container alloy constituents will depend on the Gibbs free energy of BeF_2 relative to that of the respective constituent metal fluoride. Since the free energy of BeF_2 is so much greater than that of fluorides of commonly-used alloying elements (Fe, Cr or Ni), the dissolution and, hence, the corrosion of these elements will be quite negligible. In effect the metallic beryllium acts as a sacrificial anode to prevent oxidation of the containment wall. Acceptable corrosion rates of type 316 stainless steel at 650°C have been demonstrated in LiF-BeF_2 salt in the presence of beryllium.

Based on the above, HT-9 appears well suited to the containment of LiF-BeF_2 in the proposed design. The major compatibility problem in the use of this alloy will arise from its direct contact with beryllium. Some interdiffusion between iron and beryllium can be expected at the proposed operating temperature ($\leq 600^\circ\text{C}$). This problem was also addressed in the tandem mirror blanket study,⁽¹⁾ and an experimental program is underway to test diffusion couples of steel and HT-9 with beryllium. If the interaction between beryllium and HT-9 does prove excessive, a potential solution will be to apply a nonreactive cladding, such as vapor-deposited molybdenum, to the exposed HT-9 surfaces.

c. Magnetohydrodynamic Effects on Molten Salt Corrosion. In electrolytes flowing across strong magnetic fields, the Lorentz forces acting on the ionic components cause charge separation. Thus, an emf is induced that can affect both the free energy change and kinetics of chemical reactions occurring between the salt and electrically-conducting blanket components. The magnitude of the emf depends on the dimensions of piping cross-sections, the magnetic field strength, and the flow velocity. In the proposed blanket, the MHD effect will be strongly dependent on the electrical impedance across the packed bed of thorium oxide/beryllium spheres. If the impedance is low, then the bed will act as a short-circuit to limit the voltage from wall to wall. Furthermore, the relatively large number of flow channels in the packed bed effectively divide the induced emf into a series of relatively small potential drops per channel. Nevertheless, the MHD effect on corrosion cannot be

quantitatively assessed without a more definitive blanket design and additional electrochemical data on the LiF-BeF₂ system.

d. Transmutation Effects. The most important transmutation reactions in the salt are those involving the production of tritium and can be written as:

The TF produced in these reactions will react with oxidizable metals in the blanket, which in the present concept will be effectively limited to metallic beryllium.

There will also be transmutation of beryllium ions in the salt according to the reactions:

Again, the beryllium metal in the blanket can be expected to react with the free fluorine produced in the above reactions and to effectively buffer the HT-9. If the "wastage" of beryllium metal attributable to the above reactions were deemed too high for the lifetime of the beryllium in the reactor, a redox couple comprised of solute atoms such as Ce³⁺ and Ce⁴⁺ can be considered. However, this would also require some form of coating (beryllide or oxide) on the beryllium to reduce its chemical activity.

2.5.3 Solid Breeder Materials

Most of the compatibility issues affecting the performance of thorium and beryllium in the blanket have been addressed in the preceding discussion of molten salt compatibility. One additional problem, however, concerns the potential self-welding between adjacent beryllium spheres.

The problem is similar to that being addressed in the lithium-cooled blanket design being developed for the mirror fission-fusion concept,⁽¹⁾ although it may be compounded in the present concept because of the higher blanket temperature. Many factors can affect the rate of self-welding, as, for example, any corrosion reactions at the beryllium surface; accordingly the problem cannot be gauged without data from self-welding tests now underway.

2.5.4 Summary of Issues

The extensive testing of LiF-Bef_2 salts for fission reactor applications provides an effective data base for evaluating these salts as coolants for fusion reactor blankets. Using this data base, it appears that the use of Li_2BeF_4 as the tokamak hybrid blanket coolant could accommodate the incorporation of beryllium metal and HT-9 containment as well or better than a liquid metal coolant. However, metallic thorium could not be accommodated in the salt concept without the development of a highly reliable coating or cladding. As the thorium is introduced in the less reactive form of an oxide, many issues can be avoided. However, a composite pebble fabricated from beryllium and thorium oxide would, most likely require a plug rather than a thorium metal snap ring (see Reference 1). The plug design results in a higher thermal gradient in the pebble and can lead to shorter pebble life.

Therefore, the compatibility issue of highest priority is whether the reference salt concept can be adapted to the use of thorium metal rather than thorium oxide. The issue is threefold: (1) is there a workable scheme for protecting the thorium from the salt, (2) is that scheme more tractable than one that would protect metallic beryllium from a ThF_4 containing salt which has small concentrations of UF_3 , and (3) how do the costs of reprocessing ThO_2 or ThC , which are compatible with salt, weigh against the costs of protecting thorium? The answers hinge on the kinds of thicknesses of barrier coatings or claddings for the thorium that can be applied without over-complicating the design or affecting the breeding gain. Once the barrier coatings or claddings have been defined, a development program would be needed to provide their feasibility. However, it is premature to begin such a program without a fuller exploration of salt blanket concept alternatives.

Because of the relatively high melting point of the LiF-BeF_2 salt, the FLIBE cooled blanket must operate at a higher temperature overall than that of alternative liquid metal concepts. This higher temperature will almost certainly enhance the interdiffusion between adjacent beryllium spheres and between HT-9 and beryllium, a factor that could ultimately decide the feasibility of the salt concept vis-a-vis liquid metal. Accordingly, it is important to determine the temperature dependence of Be-Be and Be-Fe interdiffusion at an early stage of concept development. Studies are now underway to measure diffusion rates in these couples in contact with lithium, and the studies should be extended to measure the diffusion rates of similar couples in molten Li_2BeF_4 .

A final issue that is important in assessing molten salt blankets concerns the severity of electrochemical corrosion effects produced in flowing salts through a strong magnetic field. The packed bed geometry, by simple analysis, appears to be a very favorable arrangement for minimizing the induced potentials, but a more definitive analysis will be needed once the salt blanket concept has been better detailed.

3. ADDITIONAL RECOMMENDATIONS FOR FY83 WORK

Several new physics developments and design alternatives should be investigated for their potential impact on the tokamak hybrid design effort. Other areas require more detailed design analysis. It is recommended that three recent innovations be analyzed for their possible application to the adaptation of a liquid metal blanket for the tokamak hybrid:

1. Bean-Shaped Plasmas. MHD stability theory has been reasonably successful in predicting the achievable beta-values of circular plasmas in PDX and ISX-B and D-shaped plasmas in Doublet III. According to this same theory, a "bean-shaped" plasma should have a beta limitation as much as twice that of the D-shaped plasma used in the reference tokamak. Then, for the same wall loading, the average B could be reduced by a factor up to 1.4, and the resultant MHD pressure drop (proportional to B^2) could be reduced by a factor of 2).
2. Inboard Blanket Replacement by Multiplier/Reflector. A significant complication in the current mechanical design is the inboard blanket. Space limitations and the magnet shielding requirements make adequate blanket and first wall coverage on the inboard side a significant design challenge. One option for this blanket is a 50 cm layer of liquid lead, flowing slowly downward. 1-D neutronics calculations showed that the best neutron reflector for the inboard wall of a tokamak is lead backed by Zircaloy. It may be possible to eliminate breeding on the inboard side. However, 3-D neutronics analysis of the tokamak neutron situation is required before we can make a definitive conclusion.
3. Polarized Fuel. Although the use of polarized D-T fuel in a tokamak has only recently been proposed and is highly speculative, some potential improvements can result. Two options have been considered.

Option A - Transverse Polarization: If the deuteron spin is polarized transverse to $\vec{B}$ and there is not triton polarization, then the fusion neutrons are emitted with a distribution peaked along $\vec{B}$. In low-aspect-ratio tokamak geometry, only 15% of the fusion neutrons will be directed to the inboard side of the torus. If this side is designed for maximum neutron multiplication and reflection (using Pb or Be), it may be possible to restrict the breeding blanket locations to the lower B field outboard, top and bottom regions with little net loss in breeding.

Option B - Ortho Polarization: The principal advantage of ortho (or parallel) polarization is that a 50% increase in fusion reactivity results. This option allows a reduction of 11% in the magnetic field for the same fusion power, with a corresponding reduction of 23% in the pressure required to pump liquid lithium. The principal disadvantage of ortho polarization is that the $\sin^2\theta$ distribution of the fusion neutrons results in somewhat lower breeding on the outboard blanket, even when maximizing reflection of neutrons from the inboard side. Also, the radiation damage rate to the inboard first wall is increased.

It was tentatively concluded that the transverse-D polarization is preferred for the tokamak hybrid, but the resulting advantages for either mode do not appear to be compelling. An experimental program to address the utility of polarization will soon be underway at PPPL.

An important consequence of these innovations is that it may be possible to use the same liquid lithium coolant system that was designed for the tandem mirror hybrid. The use of a liquid metal blanket would result in markedly improved breeding characteristics.

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