OVERVIEW

BLANKET COMPARISON AND SELECTION STUDY

OVERVIEW OF THE BLANKET COMPARISON AND SELECTION STUDY

DALE L. SMITH, CHARLES C. BAKER, and DAI KAI SZE
Argonne National Laboratory, 9700 South Cass Avenue, Argonne, Illinois 60439

GROVER D. MORGAN *McDonnell Douglas Astronautics Company*
P.O. Box 516, St. Louis, Missouri 63166

M. A. ABDOU *University of California-Los Angeles*
Los Angeles, California 90024

STEVEN J. PIET *EG&G Idaho, Inc., P.O. Box 1625*
Idaho Falls, Idaho 83401

K. R. SCHULTZ *GA Technologies, Inc., P.O. Box 85608*
San Diego, California 92138

RALPH W. MOIR *Lawrence Livermore National Laboratory*
Livermore, California 94550

JAMES D. GORDON *TRW Energy Development Group*
One Space Park, R1/2128, Redondo Beach, California 90278

Received November 21, 1984
Accepted for Publication April 4, 1985

The Blanket Comparison and Selection Study (BCSS) was a 2-yr, multilaboratory project initiated by the U.S. Department of Energy/Office of Fusion Energy. Its primary objectives were to (a) define a limited number of blanket concepts that should provide the focus of the blanket research and development (R&D) program, and (b) identify and prioritize critical issues for the leading blanket concepts. The BCSS focused on the mainline approach for fusion reactor development, namely, the D-T-Li fuel cycle, tokamaks and tandem mirror reactors (TMRs) for electrical energy production, and a reactor parameter space that is generally considered achievable with modest extrapolations from the current data base. The STARFIRE and Mirror Advanced Reactor Study reactor and plant designs, with a nominal first-wall neutron load of 5 MW/m², were used as reference designs for the study.

The study focused on

1. *development of reference design guidelines, evaluation criteria, and a methodology for evaluating and ranking candidate blanket concepts*
2. *compilation of the required data base and development of a uniform systems analysis for comparison*
3. *development of conceptual designs for the comparative evaluation*
4. *evaluation of leading concepts for engineering feasibility, economic performance, and safety*
5. *identification and prioritization of R&D requirements for the leading blanket concepts.*

Sixteen concepts (nine TMR and seven tokamak) that were identified as leading candidates in the early phases of the study were evaluated in detail. The overall evaluation concluded that the following concepts should provide the focus for the blanket R&D program (breeder/coolant/structure):

1. *lithium/lithium/vanadium alloy*
2. *Li_2O/helium/ferritic steel*
3. *LiPb alloy/LiPb alloy/vanadium alloy*
4. *lithium/helium/ferritic steel.*

I. INTRODUCTION

The Blanket Comparison and Selection Study (BCSS) was a 2-yr, multilaboratory project initiated by the U.S. Department of Energy/Office of Fusion Energy (DOE/OFE). Its primary objectives were to (a) define a limited number of blanket concepts that should provide the focus of the blanket research and development (R&D) program, and (b) identify and prioritize critical issues for the leading blanket concepts.[1,2] The BCSS focused on the mainline approach for fusion reactor development, namely, the D-T-Li fuel cycle, tokamaks and tandem mirror reactors (TMRs) for electrical energy production, and a reactor parameter space that is generally considered achievable with modest extrapolations from the current data base. The STARFIRE (Ref. 3) and Mirror Advanced Reactor Study[4] (MARS) reactor and plant designs, with a nominal first-wall neutron load of 5 MW/m^2, were used as reference designs.

The BCSS was led by Argonne National Laboratory (ANL) with major support from industry, universities, and other national laboratories (see Table I). The major support organizations provided teams of experts, while the other special support consisted of experts in selected areas. A nine-member executive committee, which consisted of managers from ANL and the major support organizations, served as the decision-making body for the study. A review committee was appointed by DOE/OFE to provide periodic evaluation of the progress and direction of the study.

TABLE I

BCSS Team

Lead laboratory
Argonne National Laboratory
Major support organizations
McDonnell Douglas Astronautics Company (MDAC)
GA Technologies, Inc. (GA)
TRW, Inc. (TRW)
EG&G Idaho, Inc. (EG&G)
Lawrence Livermore National Laboratory[a] (LLNL)
University of California, Los Angeles[a] (UCLA)
Special contributors
Grumman Aerospace Corporation[b]
Energy Technology Engineering Center[b]
Westinghouse Electric Corporation[b] (WEC)
University of Wisconsin[b] (UW)
Rensselaer Polytechnic Institute[b]
Hanford Engineering Development Laboratory[b] (HEDL)
Oak Ridge National Laboratory (ORNL)

[a]FY 1984 only.
[b]FY 1983 only.

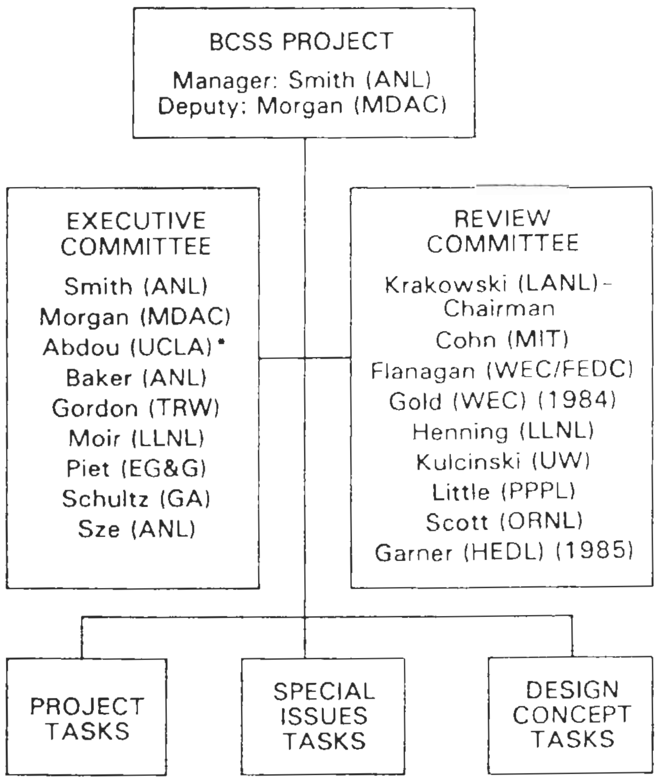
The BCSS was conducted with a two-phase approach. Phase 1 involved

1. developing reference design guidelines
2. developing evaluation criteria and methodology for ranking concepts
3. identifying concepts for considerations, including mainline and alternate concepts
4. compiling a data base and developing a uniform systems analysis
5. developing conceptual designs for evaluation purposes.

Phase 2 involved

1. identifying critical issues for each concept
2. evaluating blanket concepts
3. identifying and prioritizing R&D requirements for leading concepts.

The BCSS project was organized into three major tasks, namely, project tasks, special issues, and blanket design, as indicated in Fig. 1. The project tasks included such areas as design guidelines; evaluation criteria; and engineering, economics, safety, and R&D evaluations. Special issues tasks provided a consistent



*Project Manager in fiscal year 1983.

Fig. 1. The BCSS project organization.

data base and analysis methodology to be used for all design concepts. The design concepts tasks developed reference designs with sufficient detail to provide meaningful evaluations.

Design guidelines and evaluation methodology and criteria, based on the reactor parameter space for tokamak and TMR concepts that is generally considered reasonably achievable, were developed for the study. Using the STARFIRE and MARS reactor and plant designs as a basis, blanket concepts were developed for evaluation. A blanket concept is defined by the selection of materials for the primary components, namely, breeder, coolant, structure, and neutron multiplier if required, and by the geometric characteristics of the design. Table II lists the candidate materials that served as a focus for the BCSS.

In the initial phases of the study, various blanket concepts were designated either as "mainline" or "alternate" concepts. The mainline concepts included those blanket concepts that had been developed to a greater extent and that were generally believed to offer the greatest potential. All other concepts, including more innovative and less well-developed concepts, were considered alternate concepts. The phase 1 effort focused on (a) development of reference designs for the mainline concepts and (b) a concept screening of alternate concepts to determine whether any alternate concepts should be evaluated in detail. The mainline concepts considered included the following breeder/coolant combinations with each of the three candidate structural alloys and with/without a neutron multiplier:

Li/Li	Li_2O/H_2O
Li/He	Li_2O/He
LiPb/LiPb	$LiAlO_2$/He
LiPb/He	$LiAlO_2/H_2O$
LiPb/H_2O	Li_8ZrO_3/He
LiPb/Na	Li_8ZrO_3/H_2O.

From all of the alternate concepts, only those with nitrate salt (NS) as a coolant or Flibe as a breeder with helium coolant were selected for more detailed evaluation.

A major part of the effort was devoted to compiling a materials data base and developing the analytical capability required for blanket concept development and evaluation. From the various breeder/coolant/structure/neutron multiplier combinations listed above, ~130 concepts were included in the initial evaluation. These blanket concepts were developed in sufficient detail to evaluate the performance and safety characteristics of each concept.

Sixteen leading concepts (seven tokamak and nine TMR blankets) selected in phase 1 of the study were evaluated in detail in phase 2 of the study. A detailed evaluation methodology was developed in each of four areas:

TABLE II

Candidate First-Wall/Blanket Materials

Breeding Materials	Coolants	Structure	Neutron Multiplier
Liquid metals Lithium 17Li-83Pb	H_2O Lithium 17Li-83Pb Helium NS[d]	Austenitic steel PCA Manganese steel[a]	Beryllium Lead
Ceramics Li_2O Li_8ZrO_6 $LiAlO_2$[b]		FS HT-9 Modified FS[a]	
Salt Flibe[c]		Vanadium alloy V-15 Cr-5 Ti	

[a]Low-activation structural alloys; V-15 Cr-5 Ti is inherently low activation.
[b]$LiAlO_2$ is representative of ceramics, including Li_2SiO_3, Li_2ZrO_3, etc.
[c]Fluoride salt.
[d]Nitrate salt.

1. engineering feasibility
2. economics
3. safety
4. R&D requirements.

Based on the blanket designs developed and the analyses performed, a relative ranking of the leading blanket concepts was developed in each of the four evaluation areas. Critical issues associated with each concept were identified, an R&D assessment was performed, and the R&D requirements for the leading concepts were identified and prioritized.

The results of this study are summarized in this overview. Detailed results are presented in Refs. 1, 2, and 5 through 18.

II. DESIGN GUIDELINES AND EVALUATION METHODOLOGY

Uniform design guidelines and evaluation criteria were developed to provide a consistent basis for comparison of the various blanket concepts.

II.A. Design Guidelines

The purposes of the design guidelines were

1. to establish the value (or range of values) of parameters and to specify assumptions that require consistency in evaluating the various blanket concepts

2. to provide uniform guidance on the approach to handling issues that impact blanket design and/or performance.

Table III lists the key design guidelines used in this study. In many cases sensitivity studies were conducted to evaluate the impact of variations of the reference guidelines on the performance characteristics of selected blanket concepts. Other design guidelines such as structural and breeder material temperature limits, tritium breeding requirements, and fluence limits to the toroidal field (TF) coils are discussed in Sec. III.

II.B. Evaluation Methodology

An important part of the study was the development of detailed evaluation criteria and a methodology for uniform comparison of the various blanket concepts. In the early phases of the study, initial screening criteria, minima or maxima, were established for several important parameters: breeding ratio, thermal efficiency, tritium inventory, lifetime, tritium loss rate, and minimum wall loading.

Approximately 130 concepts were developed in sufficient detail for a qualitative comparison by the executive committee. These concepts were ranked as follows:

$R = 1$: *Potentially attractive.* These concepts were recommended for further development.

$R = 2$: *Set aside for possible future consideration.* These concepts were judged to be potentially acceptable but less attractive than the $R = 1$ concepts.

$R = 3$: *Rejected.* These concepts did not meet initial screening criteria. They were judged to be clearly inferior to other concepts and eliminated from further consideration.

The $R = 1$ concepts were then evaluated in more detail, partially on a comparative basis, to reduce the number of top-rated concepts to an acceptable number for detailed evaluation. The nine concepts (breeder/coolant/structure/neutron multiplier) listed in Table IV were finally rated $R = 1$ and evaluated in detail.

A detailed methodology was developed for evaluation of these concepts in each of four areas:

1. engineering
2. economics
3. safety
4. R&D requirements.

The evaluation methodologies and results are summarized in Sec. V. Details of the procedures and rankings are presented in Refs. 1, 2, 5, and 6. The overall ranking of concepts was based primarily on the engineering, economics, and safety evaluations.

TABLE III

Design Guidelines

	Tokamak	TMR
Reactor design basis	STARFIRE	MARS
Peak magnetic field (T)	10	5
Neutron wall load (MW/m^2)	5	5
First-wall heat flux (W/cm^2)	100	5
First-wall erosion (mm/yr)	1	0.1
Dose to TF coils (rad)	10^{10}	10^{10}

TABLE IV

Top-Rated Blanket Concepts Given Full Evaluation

(Breeder/coolant/structure/neutron multiplier)

Li/Li/V	Li_2O/He/FS
Li/Li/FS[a]	$LiAlO_2$He/FS/Be
LiPb/LiPb/V[a]	$LiAlO_2$/H_2O/FS/Be
Li/He/FS	$LiAlO_2$/NS/FS/Be
Flibe/He/FS/Be	

[a]Not rated $R = 1$ for tokamak configuration.

III. SPECIAL ISSUES

Several special issues important to more than one blanket concept were evaluated separately to provide a common base for all concepts. The special issues include

1. a materials data base assessment (structural materials, corrosion limits, breeder materials, and special materials)
2. tritium containment
3. structural and electromagnetic analyses
4. neutronics analyses (tritium breeding, shielding, and activation)
5. reliability, resource, and high-power density blanket considerations
6. auxiliary components (limiter/divertor, energy conversion system, etc.) (see Refs. 1, 2, and 12 through 18).

III.A. Structural Materials

Three classes of alloys are currently considered as leading candidates for the first-wall/blanket structure of a commercial fusion reactor: austenitic stainless steels, ferritic (martensitic) steels (FS), and vanadium-base alloys. For the BCSS program, one reference or baseline alloy was selected from each class and one low-activation counterpart for the austentic and FSs was identified for evaluation as part of the study; the reference vanadium alloy is inherently low activation.

Austenitic stainless steels have been used extensively in fusion reactor applications, and, therefore, possess the most developed data base for nuclear applications. For this reason, the austenitic steels are generally regarded as a reference to which other alloys are compared. The primary candidate austenitic alloy (PCA), which is under development in the U.S. alloy development program, was selected as the reference austenitic alloy. This alloy, which is a modification of Type 316 stainless steel, in the 20 to 25% cold-worked (CW) condition is the product of several years of development to provide a radiation damage resistant alloy for fusion reactor applications.

The low-activation counterpart to PCA is a manganese-stabilized steel with very low nickel and molybdenum in order to qualify for class "C" radioactive waste disposal per 10CFR61. The manganese steels are noted for their "hardenability" and were developed primarily for wear resistance applications. Although most of the compositions commercially available today contain significant amounts of nickel and molybdenum, and are difficult to fabricate, a manganese steel with a composition Fe-15 Mn-15 Cr-0.05 C-0.01 N was proposed for evaluation in the present study. Major concerns regarding the use of this alloy relate to corrosion and safety because of the high mobility/volatility of manganese. Other properties are assumed to be similar to those of PCA.

The high chromium (martensitic) FSs, e.g., HT-9 and Fe-9 Cr-1 Mo, offer possible advantages over the austenitic steels in the areas of radiation swelling resistance, lower thermally induced stresses, and better compatibility with liquid lithium and LiPb alloy. The HT-9 (Fe-12 Cr-1 Mo-V-W) alloy in the normalized and tempered condition is selected as the reference ferritic alloy for this study primarily on the basis of the extensive nonirradiation data base and strength at high temperatures. Although this alloy exhibits good radiation swelling resistance, the composition and thermomechanical treatment have not been optimized for radiation damage resistance as in the case of the PCA. Welding and radiation embrittlement are primary concerns.

The low-activation FS proposed for evaluation is Fe-11 Cr-2.5 W-0.3 V-0.15 C. Tungsten is substituted for molybdenum in this alloy. While this specific alloy has not been made, alloys with similar compositions have been produced. As a result, there is a high degree of confidence that the proposed alloy can be fabricated with properties similar to commercial HT-9. In this study, both unirradiated and irradiated properties were assumed to be equivalent to those for the HT-9 alloy.

Vanadium-base alloys represent an advanced alloy system that offers advantage with respect to higher temperature operation, better corrosion resistance in lithium (and probably LiPb), and possibly better radiation damage resistance. The V-15 Cr-5 Ti alloy, which was originally developed under the fast breeder reactor program, is selected as the reference alloy. The titanium provides improved radiation damage resistance and the chromium provides improved high-temperature mechanical properties. Although this alloy was developed partially on the basis of good radiation damage resistance, it does not necessarily represent an optimized composition. Because of the limited data base, this alloy system requires a larger R&D effort.

The reference vanadium-base alloy also meets the "low-activation" definition in terms of waste management. Therefore, an alternate low-activation alloy is not required for this system.

It is important to note that for all low-activation alloys, the long-term activation will be dominated by activation products from trace impurities. Therefore, very low concentrations of certain impurities, e.g., niobium, molybdenum, and nickel, must be maintained to meet class "B" or "C" waste disposal criteria.

A grooved structural first wall was found to provide substantial benefits for the tokamak concepts, which must accommodate relatively high-heat fluxes and erosion rates on the first wall.[1,15] Analyses indicate that significant reductions in thermal stresses result from orthogonal grooves on the plasma side of the first wall.

Table V provides a summary of predicted performance characteristics and limitations of the candidate structural alloys. Key conclusions from the study are summarized in Table VI.

III.B. Corrosion/Compatibility

Critical aspects of liquid-metal, molten salt, water, and gaseous corrosion/compatibility with candidate structural materials were evaluated in detail.[1,13] The present study included the following assessments:

1. liquid-metal corrosion/compatibility with lithium and 17Li-83Pb
2. molten salt (NS and Flibe) corrosion/compatibility
3. water (200 to 350°C) corrosion of vanadium alloys and CW PCA
4. gaseous corrosion/compatibility of vanadium.

III.B.1. Liquid-Metal Corrosion/Compatibility

Corrosion and compatibility issues are a major consideration in assessing the viability of the different liquid-metal blanket designs. The most important compatibility concerns in any application of liquid metals are corrosion/mass transfer and the effect of environment on the mechanical properties of the containment material. Corrosion can lead to significant wall thinning/wastage and deposition of corrosion products in cooler areas of the circuit. Deterioration of mechanical strength of structural materials can result from the influence of the environment itself and the effects of microstructural and compositional changes that occur in the material during long-term exposure to the liquid metal. Reference 1 provides an assessment of the corrosion behavior of austenitic PCA, ferritic HT-9, and vanadium V-15 Cr-5 Ti alloys in liquid lithium and eutectic 17Li-83Pb environments.

Factors that affect corrosion include: liquid-metal purity, composition and microstructure of the containment material, temperature, exposure time, velocity

TABLE V

Structural Materials Assesment

Candidate Alloys	Austenitic Steel PCA CW	FS HT-9	Vanadium V-15 Cr-5 Ti
Physical properties			
melting temperature (°C)	1400	1420	1880
Nuclear characteristics[a]			
(dpa/MW·yr/m^2)	11	11	11
(appm He/MW·yr/m^2)	174	130	57
(appm H/MW·yr/m^2)	602	505	240
Heating rate (W/cm^3)	40	40	25
TBR[a]	1.23	1.23	1.28
Thermal stress factor [MW/m^2·mm (500°C)]	3.2	4.8	9.8
Maximum surface heat flux (MW/m^2)[b]	0.3	0.4	1.8
Design stress limit			
S_m (MPa) 500°C	205	175	220
S_m (MPa) 550°C	192	160	235
S_{mt} (MPa) (2×10^4, 100 dpa)			
500°C	100	155	165
550°C	85	100	165
700°C	---	---	165
Maximum allowable temperature, °C ($\sim 0.5 T_m$) (irradiation embrittlement)	550	550	720
Corrosion rate (mg/m^2·h)[c]			
Lithium (500°C)	60	2	<0.01
LiPb (500°C)	>100	100	0.01
Radiation lifetime (swelling) (5%)	100 dpa (500°C) 150 dpa (400°C)	190 dpa[d]	220 dpa[d]
Critical design issues	• Limited lifetime (swelling) • High-thermal stress • Liquid-metal corrosion • Radiation creep • Operating temperature limit	• Weld procedure (postweld heat treatment) • Ductile-to-brittle transition temperature (DBTT) above room temperature • Operating temperature limit • Liquid-metal embrittlement • Ferromagnetic properties	• R&D requirements • Weld procedure (inert environment) • Oxidation characteristics • High-tritium permeation rates • Costs

[a]For lithium blanket.
[b]Idealized 5-mm-thick flat plate with 50°C film coefficient, $T_{out} = 400$°C.
[c]Predicted for 1.5 m/s.
[d]Not well defined; may be higher.

TABLE VI

Key Conclusions of BCSS

- Ferritic steel and/or vanadium alloys were selected as structures for all leading blanket concepts.
  1. There is a higher risk than PCA.
  2. The probability that they will work is high.
  3. They provide significant advantages compared to PCA.
  4. Vanadium provides temperature and heat load advantages.
- Low-activation structure is feasible.
  1. Modified FS with properties similar to HT-9 can be developed.
  2. Vanadium alloy is inherently low activation.
  3. Manganese-stabilized steel performance is similar to PCA with additional problems.
- Grooved first wall provides significant lifetime/erosion advantage for tokamak.
- Except for reactivity problems, lithium is generally superior to LiPb.
- Tritium recovery from solid breeders appears feasible.
  1. Hydrogen swamping appears necessary to facilitate tritium release (tritium released as HT).
  2. Long-term radiation effects are unknown.
  3. Swelling of Li_2O presents a design problem.
- Tritium containment/recovery is a major concern for all concepts except lithium.
  1. Tritium is in reduced form (HT or T_2) at relatively high pressures.
  2. Effective tritium barriers are necessary to contain tritium.
- Acceptable tritium breeding is attainable for all leading concepts except possibly Li_2O. Ternary oxides require an effective neutron multiplier.
  1. Li_8ZrO_6 will not provide sufficient breeding without a neutron multiplier.
  2. Major uncertainties in tritium breeding requirements relate to plasma burnup fraction, required doubling time, and tritium processing efficiency.
- Beryllium is the only reasonable neutron multiplier option.
  1. Resources are adequate for hundreds of reactors.
  2. Efficient reprocessing is required.
  3. Swelling can probably be accommodated.

[including magnetohydrodynamic (MHD) effects], system ΔT, surface area and temperature profile, and system containment (e.g., bimetallic system). In general, the data base is inadequate for both lithium and LiPb to define the importance of each of these factors. However, the data for austenitic and FSs are sufficient to provide reasonable projections of corrosion rates for lithium and LiPb under anticipated conditions. Only limited data exist for corrosion of vanadium in lithium and LiPb. Based on these data and data for other refractory metals, very low corrosion rates are predicted for vanadium alloys in lithium and LiPb.

The basis for a temperature limit from corrosion considerations can be radioactive mass transport, wall thinning/wastage, or mass transfer and deposition. The specified corrosion limit for hands-on maintenance, based on fission reactor experience, is 0.5 μm/yr. The corrosion limit to avoid problems from excessive deposition of corrosion product in localized regions is generally believed to be ~5 μm/yr. Because of the specific design dependency and uncertainties associated with this limit, a more liberal limit of 20 μm/yr is specified for this study. The allowance for wall thinning is specified as 10% of the wall thickness; however, this limit is not likely to be important for section thicknesses >3 mm during a service life of 2 to 4 yr. In most cases, the most important consideration in establishing the operating temperature limits for fusion reactor blankets is mass transfer and deposition.

Table VII lists the proposed design temperature limits based on mass transfer/deposition and radioactive mass transfer for the three structural materials in flowing lithium and LiPb. The corrosion rates for PCA in LiPb are clearly excessive for acceptable thermal-hydraulic performance. The rates for PCA in lithium and FS in LiPb pose severe constraints that would generally make such systems unattractive. The

TABLE VII

Design Temperature Limits (°C) for Liquid-Metal Systems at 1.5 m/s

Liquid Metal	Criteria[a] (μm/yr)	Austenitic Steel, PCA	FS, HT-9	Vanadium Alloy, VCrTi
Lithium	20	470	580	>750
	5	430	550	>750
	0.05	370	460	>750
LiPb	20	410	450	>750
	5	375	415	>750
	0.05	320	360	650

[a]Reference criteria for mass transfer/deposition and radioactive mass transfer in this study are 20 and 0.05 μm/yr, respectively.

corrosion rates for FS in lithium meet the mass transfer/deposition criteria; however, radioactive mass transfer is sufficient to require remote maintenance. The predicted corrosion rates for V-15 Cr-5 Ti in lithium and LiPb, although highly uncertain, satisfy both the mass transfer/deposition and the radioactive mass transfer criteria with considerable margin. Therefore, remote maintenance would not be dictated by corrosion considerations.

III.B.2. Molten Salts

An adequate data base exits for design of nonnuclear NS heat transfer systems with austenitic steels to temperatures of 600°C. No data have been reported on the corrosion of HT-9 or Fe-9 Cr-1 Mo. The dominant corrosion effect observable in austenitic steel heat transfer systems with the NS is the formation of a duplex spinel/magnetite oxide film and an uptake of oxidized chromium (+6) by the molten salt. For nonnuclear applications, a corrosion allowance of 13 μm/yr appears adequate up to 600°C. For purposes of this study, it is assumed that the corrosion behavior of FS is similar to the austenitic steels; however, this must be verified.

The salts are somewhat conducting and, when moved through a magnetic field, generate a voltage that can cause dissociation of the salt. Increasing the ionic content of the salt increases the corrosion of the structure. Although very little data exist, preliminary estimates indicate that this effect is not serious.

The majority of the relevant corrosion data for Flibe has been obtained with austenitic stainless steels. Based on the Molten Salt Reactor Experiment, corrosion rates of Type 316 stainless steel loops containing a $2LiF$-BeF_2 mixture under heat transfer conditions average ~8 μm/yr at 650°C. These rates can be lowered significantly by chemically buffering the salt and/or reducing the chromium content of the steel. The high nickel alloys generally exhibit superior corrosion resistance compared to the austenitic steels. The high chromium FSs have not been investigated. Currently, there is a great uncertainty in the Flibe corrosion properties in a magnetic field. Further tests are required to evaluate the potential for electromagnetic effects on the corrosion by the salts.

Vanadium is not considered compatible with the salts above ~400°C because of oxidation problems.

III.B.3. Water Corrosion

Vanadium alloys have generally not been proposed for use in pressurized water-cooled systems because of corrosion considerations. Evaluation of recent scoping data, however, concludes that selected alloys such as VCrTi may be acceptable for use in pressurized water.

Although austenitic stainless steels have been used extensively in pressurized water systems, stress corrosion problems have frequently been observed under certain conditions. The combination of CW and reduced ductility under irradiation may exacerbate this problem. Further investigations should be conducted to more thoroughly evaluate the seriousness of this problem. For the present study, it is assumed that this problem will not prevent the use of CW PCA in pressurized water systems.

III.B.4. Gaseous Corrosion/Compatibility of Vanadium-Base Alloys

An evaluation of the thermodynamic and kinetic processes for vanadium and VCrTi alloys exposed to helium with low impurity concentrations indicates that oxidation is excessive (unacceptable) if VCrTi is exposed to helium with greater than ~0.1-ppm moisture at temperatures above ~500°C. An evaluation of the helium coolant cleanup indicates that the purities required here are extremely difficult to attain economically in practical systems.

No severe effects are predicted for exposure of vanadium-base alloys to air for a few hours at temperatures <650°C; however, since one oxide of vanadium melts at ~670°C, rapid attack may occur at higher temperatures.

III.C. Breeder Materials

Lithium is the only viable tritium breeding material for a deuterium-tritium (D-T) fusion reactor. Liquid lithium, the 17Li-83Pb eutectic alloy, solid compounds including Li_2O and $LiAlO_2$, and the fluoride salt (Flibe) are the leading candidate breeder materials considered in the BCSS.

III.C.1. Solid Breeder Materials

Li_2O and several ternary lithium oxides are generally considered as the leading candidates for the solid breeder blanket concepts. The Li_2O is of interest because adequate tritium breeding may be attainable without the added complexity of a neutron multiplier. The ternary compound Li_8ZrO_6 was also of interest because of its relatively high breeding potential and the possibility of better thermochemical stability compared to Li_2O. All other ternary ceramics considered require an effective neutron multiplier. Primarily because of the higher melting temperature, and hence better thermochemical stability, $LiAlO_2$ was selected as the reference ternary solid breeder for this study.

Critical issues associated with solid breeder materials relate to the following:

1. fabrication/refabrication of the ceramic
2. property data base
3. tritium release from solid
   a. temperature limits
   b. species

4. radiation effects (swelling)

5. tritium breeding.

Important aspects of the first four issues are discussed in Refs. 1, 2, and 14, and briefly summarized here. The tritium breeding considerations are discussed in the BCSS report and Refs. 17 and 18.

III.C.1.a. Fabrication/Refabrication. Two configurations, pressed and sintered plates and sphere-pac materials, were chosen for the BCSS solid breeder blankets. Considerable experience now exists in powder preparation, in the fabrication of sintered breeders by cold pressing/sintering, and by hot pressing. The latter technique is preferred when grain size is to be preserved to high density. Future development in this area needs to focus on breeder microstructure tailoring and on properties enhancement.

Sphere-pac solid breeders offer the potential in reducing the blanket temperature variability associated with breeder cracking and the gap conductance uncertainty. Sphere-pac requires three sizes of high-density [>98% theoretical density (TD)] spheres to achieve ~88% smear density. These particle sizes have diameter ratios of 40:10:1; the actual diameters currently used for fission fuels are 1200, 300, and 30 μm. The same sizes have been recommended for the sphere-pac solid breeder blankets. There have been few direct experiences in fabricating sphere-pac solid breeders and none regarding their performance characteristics in an irradiation environment. Consequently, there remain several fabrication development issues. Refabrication of recycled irradiated material, which is essential, is a major development problem.

III.C.1.b. Property Data Base. Several thermophysical and mechanical properties of the candidate solid breeder materials are required for blanket design and performance/safety evaluation. Particularly important properties include: hydrogen/tritium solubility and diffusivity, surface desorption characteristics for tritium, thermal conductivity, specific heat, thermal expansion, helium diffusivity, elastic and fracture properties, high-temperature creep, and chemical compatibility. The effect of radiation on some of these properties is of particular importance. In many instances, limited data exist for the leading candidate materials. The greatest uncertainties arise from possible variations in microstructure and the effects of radiation.

Table VIII presents a comparison of several important properties for the selected candidate materials. The $LiAlO_2$ exhibits the highest melting temperature and thus, the projected highest operating temperature limit. The thermal conductivities of all candidate alloys are quite low and sensitive to both microstructure and radiation. The tritium diffusivity of Li_2O is much greater than that for $LiAlO_2$.

Data on the mechanical properties, namely, elastic moduli, fracture strength, and creep properties, are nonexistent for the candidate materials. These properties are particularly important with regard to the accommodation of the differential thermal expansion and swelling of Li_2O. Significant uncertainties relative

TABLE VIII

Properties of Candidate Solid Breeder Materials*

Breeder	Melting Temperature (°C)	ρ Li[a] (g/cm^3)	k[b] (W/m·K)	T_{min}[c] (°C)	T_{max}[d] (°C)	ΔT (°C)	Grain[e] Diameter (μm)	Tritium[b] Diffusivity (cm^2/s)
Li_2O	1433	0.93	2.5[b] 1.27[i]	410[f]	800[g]	390	3.0[h]	10^{-7}
$LiAlO_2$	1610	0.28	1.6 1.1[i]	350	1000	650	0.2	10^{-14}
Li_8ZrO_6	1295	0.68	1.8	350	760	410	2.0	---
Li_2SiO_3	1200	0.36	1.5	410	700	290	---	---

*Estimates are based on limited unirradiated and irradiated data for candidate solid breeders and other ceramic materials.
[a]For 100% TD material.
[b]Estimated for 85% TD, sintered material at 1000 K.
[c]Values are estimated based on diffusive inventory considerations.
[d]Based on sintering at $0.66T_m$.
[e]Based on the smallest grain diameters with existing fabrication technology.
[f]Based on solubility consideration.
[g]Based on high-temperature mass transfer (LiOT/LiOH) considerations.
[h]Grain growth has been observed after irradiation of Li_2O.
[i]Estimated for 87% TD sphere-pac material.

to the mechanical response of solid breeder materials remain.

III.C.1.c. Tritium Recovery. Tritium recovery considerations impose perhaps the greatest restrictions on the solid breeder operating limits. All current designs provide for a helium purge stream to flow throughout the blanket for tritium recovery. Tritium generated within the solid must diffuse to the surface, desorb from the surface, migrate to the helium purge, and be carried in the purge stream to the tritium processing system. Results from in-reactor purge flow experiments indicate that the tritium inventory can be maintained at relatively low levels provided the grain size, porosity, and temperature of the breeder material and the purge gas flow rate and chemistry are adequately controlled. Addition of hydrogen to the purge stream has been shown to have a dramatic effect on the tritium inventory. Projected temperature and grain size limits for acceptable tritium release are listed in Table VIII.

Effects of high radiation fluence and thermal cycling on the tritium release characteristics of solid breeders are not well defined. Significant swelling and grain growth have been observed after irradiation of Li_2O at temperatures of 500 to 700°C. The $LiAlO_2$ is much more resistant to swelling and grain growth; however, significant retention of tritium was observed after capsule irradiations.

III.C.2. Liquid Breeder Materials

Three liquid breeder materials, namely, lithium, 17Li-83Pb eutectic alloy, and Flibe (LiF-BeF_2), have been considered for the liquid breeder blankets. Table IX summarizes several properties of these materials. The data base for lithium is fairly well established. Important issues relate to reactivity with water, air, and concrete. Lithium has a significant solubility for hydrogen (tritium), which is an advantage for tritium containment.

Several properties of the LiPb alloy have not been measured, e.g., thermal conductivity and solubility of pertinent structural material elements. Key features of LiPb include: high density, reduced reactivity with air and water compared to lithium, and low tritium solubility that results in low inventories and relatively high tritium pressures.

TABLE IX

Properties of Liquid Breeder Materials*

	Lithium	LiPb	Flibe
Melting temperature (°C)	180	235	363
Density (g/cm^3)	0.49	9.4	2.0
Heat capacity (J/g·K)	4.2	~0.15	2.3
Thermal conductivity (W/m·K)	50	---	0.8

*At ~500°C.

Various compositions of Flibe have been considered. The eutectic composition (47% LiF-53% BeF_2) is characterized by a relatively high melting temperature, low thermal conductivity, and low tritium solubility. Tritium can be contained in Flibe in both the reduced form, T_2, and in the oxidized state, TF. Since the solubility of tritium is very low, the tritium pressures will be quite high.

III.D. Special Materials

Neutron multipliers, electrical insulators, and NSs were evaluated for special applications in the BCSS.

III.D.1. Beryllium

Based on the phase 1 BCSS evaluation, beryllium was chosen as the reference neutron multiplier for all the $LiAlO_2$ and Flibe blankets. The main concerns for beryllium are the resource limitation, irradiation swelling, tritium release, and salt compatibility. An assessment of the resource issue has concluded that it is reasonable to consider beryllium as a neutron multiplier for the first and second generations [~1800 and 3000 GW(electric)·yr, respectively] of fusion reactor service. Recycle of beryllium is required and close attention to beryllium recycle losses is important. Since beryllium becomes radioactive in the fusion environment (due to impurities), a remote fabrication technology is required. A process for fabricating and recycling beryllium pebbles has been proposed; the remoting requirement adds substantially to the total cost. For the water- and NS-cooled blankets, an efficient method for separating beryllium from $LiAlO_2$ microspheres prior to recycling also needs to be developed.

Swelling in beryllium is caused by helium bubbles generated during irradiation. Depending on the fluence and particularly the temperature histories, volumetric swelling of beryllium can vary from 5 to 33%. Both the inter- and intragranular helium bubble swelling weaken the beryllium so that its mechanical integrity cannot be assured during blanket operation. If the beryllium is not contained by a structural material (as is the case for all the $LiAlO_2$ and Flibe designs that use bare beryllium rods, spheres, and pebbles), the consequence of beryllium losing its mechanical integrity must be considered. Potential impacts on the blankets include material relocation, coolant blockage, and temperature hotspots.

Tritium release and salt compatibility of beryllium are potential safety concerns for specific blanket concepts.

III.D.2. Electrical Insulators

Electrical insulators are important to liquid-metal blankets because they can significantly reduce the

MHD pressure losses. Both MHD experiment and theory indicate that the pressure losses would be significantly reduced if high (electrical) resistance structural walls were used in the design. Two possible methods of achieving this benefit have been considered. The first utilizes a thin insulator film on the surface of the conducting wall. Compatibility and stability of the insulator in contact with the liquid metal are major concerns. The second consists of a laminated structure with a thin metallic layer over an insulator layer on the wall. In this case, the insulator is protected from the corrosive effects of the liquid-metal coolant. However, the corrosion and mechanical integrity of the thin metal clad become more critical. In both cases radiation effects are critical, particularly for the insulator; however, only low-voltage (<1-V) insulators are required in these liquid-metal blanket applications.

Based on limited information, several oxides (Y_2O_3, Sc_2O_3, CaO) and a spinel ($MgAl_2O_4$) have been identified as potential candidates for the laminated concept. The Y_2O_3 is currently suggested as the reference for the coating concept.

The laminated insulator concept is considered sufficiently credible for use in current designs. Although the insulator coating exhibits several advantages, satisfactory performance is more questionable because of the added compatibility constraints. Further work on both concepts is recommended. Liquid-metal compatibility and radiation stability of the insulators are the primary development issues.

III.D.3. Nitrate Salts

Nitrate salts ($NaNO_3$-KNO_3 and $NaNO_3$-$NaNO_2$-KNO_3) have been used for many years in a nonnuclear environment.[1,11,13] Some thermophysical properties of the reference nitrate salt (50% $NaNO_3$–50% KNO_3), also called draw salt, are listed in Table X. Primary concerns related to the use of NS coolants include thermal and radiation stabilities, MHD effects, tritium chemistry, handling, and corrosion properties. In general, only limited information is available in these areas, and almost no information exists on either radiation and/or magnetic effects. The primary advantage of the salt coolant is the potential for low operating pressure. A dominant concern relates to activation of sodium, potassium, and nitrogen.

TABLE X

Selected Properties of NS

(50% $NaNO_3$–50% KNO_3)*

Melting temperature (°C)	220
Density (kg/m^3)	1840
Thermal conductivity (W/m·K)	0.52
Viscosity (mPa)	1.8
Heat capacity (J/kg·K)	1605
Electrical conductivity (1/Ω·cm)	1.04

*Properties except melting temperature are at 400°C.

III.E. Tritium Containment

The BCSS has concentrated on the issues of tritium containment in a D-T fusion reactor blanket and coolant system.[1,16] One of the most serious issues concerns tritium leakage in steam generators. To prevent tritium leakage to the steam side of a steam generator, either one or both of the following assumptions are necessary:

1. Tritium can be oxidized rapidly into the oxide form, which will significantly reduce its permeation rate.
2. Effective barriers, e.g., oxide films, will reduce the permeation rate by a factor of 100 to 1000.

Detailed calculations for the blanket tritium recovery systems for each blanket concept have been carried out including the tritium flow rates and inventories in each blanket subsystem. The blankets can be divided into four categories from tritium containment considerations.

1. *Self-cooled lithium blanket.* No major problem is anticipated due to the high solubility of tritium in lithium.

2. *Helium-cooled lithium blanket.* Some moderate problems may be encountered in the containment of the tritium that permeates through the first wall and into the blanket coolant.

3. *Solid breeders and 17Li-83Pb self-cooled blankets.* A major effort is required to provide adequate tritium containment. By using the combined effects of oxide barriers and isotope swamping, the tritium leakage rate can be limited to between 10 to 100 Ci/day.

4. *The Flibe blanket.* The problem here is critical. Special multiple diffusion barriers, each far more effective than those recommended by the task group, are required.

III.F. Tritium Breeding Requirements

Attaining fuel self-sufficiency is clearly a critical goal for fusion. Therefore, the tritium breeding potential has been evaluated as a figure of merit (FOM) for candidate blanket concepts. The required tritium breeding ratio (TBR) must exceed unity by a margin G_0 to supply inventory for startup of other fusion reactors, to compensate for losses and radioactive decay between production and use, and to compensate for holdup inventories in various components as well as reserve storage inventory. This margin G_0 is found to strongly depend on the desired doubling time and many of the reactor plasma and engineering parameters, such as,

1. tritium fractional burnup in the plasma
2. equilibrium tritium inventories in various components, particularly the blanket
3. time constants to reach equilibrium tritium inventories
4. frequency of failure and time to repair components in the tritium processing system
5. processing efficiencies of and nonradioactive (e.g., chemical) losses from various subsystems.[1,18]

In comparing blanket concepts, as well as plasma and technology choices, as to the potential for attaining D-T fuel self-sufficiency, one needs an FOM. One such FOM, F, which has been used in the BCSS final comparative evaluation, is

$$F = \frac{T_c - (1 + G_0)}{(\Delta_G^2 + \Delta_s^2 + \Delta_p^2)^{1/2}} , \tag{1}$$

where

T_c = calculated TBR for a reference reactor system

G_0 = tritium breeding margin required for startup inventory of other reactors, to compensate for holdup, losses, and decay, and to provide adequate reserve

Δ_G = uncertainties in breeding margin associated with variations in reference parameters

Δ_s = uncertainties in breeding margin associated with uncertainties in system definition

Δ_p = uncertainties in predicting the TBR in the reference system due to uncertainties in nuclear data, calculational methods, and geometrical representation.

Tables XI and XII show the results of T_c, $1 + G_0$, Δ_G^2, Δ_s^2, and Δ_p^2 for tokamaks and TMRs, respectively.

The general conclusions are as follows. The G_0 is relatively insensitive to the blanket concept with a value of ~0.07. Of the three uncertainty factors, namely, Δ_G^2, Δ_s^2, and Δ_p^2, only the Δ_p^2 term varies significantly with concept. Since Δ_p^2 is the smallest contributor to the uncertainties, the combined uncertainty term Δ_i^2 is relatively insensitive to concept. The

TABLE XI

Results of Tritium Breeding Requirements, Potential, and Uncertainties for Candidate Blanket Concepts in Tokamaks

Concept	T_c	$1 + G_0$	Δ_G^2	Δ_s^2	Δ_p^2	$\Sigma\Delta_i^2$	$T_c - (1 + G_0)/(\Sigma\Delta_i^2)^{1/2}$
A $LiAlO_2$/NS/FS/Be	1.24	1.073	0.05	0.0094	0.0009	0.0603	0.616
B Li/Li/FS	---	---	---	---	---	---	---
C LiPb/LiPb/V	---	---	---	---	---	---	---
D Li/Li/V	1.28	1.068	0.05	0.0094	0.0041	0.0635	0.734
E Li_2O/He/FS	1.11	1.067	0.05	0.0094	0.0029	0.0623	0.160
F $LiAlO_2$/He/FS/Be	1.04	1.067	0.05	0.0094	0.0009	0.0603	---
G Li/He/FS	1.16	1.068	0.05	0.0094	0.0030	0.0624	0.193
H Flibe/He/FS/Be	1.17	1.067	0.05	0.0094	0.0017	0.0611	0.384
I $LiAlO_2/H_2O$/FS/Be	1.16	1.071	0.05	0.0094	0.0009	0.0603	0.333

TABLE XII

Results of Tritium Breeding Requirements, Potential, and Uncertainties for Candidate Blanket Concepts in TMRs

Concept	T_c	$1 + G_0$	Δ_G^2	Δ_s^2	Δ_p^2	$\Sigma\Delta_i^2$	$T_c - (1 + G_0)/(\Sigma\Delta_i^2)^{1/2}$
A $LiAlO_2$/NS/FS/Be	1.29	1.069	0.05	0.0094	0.0009	0.0603	0.811
B Li/Li/FS	1.14	1.068	0.05	0.0094	0.0035	0.0629	0.265
C LiPb/LiPb/V	1.18	1.067	0.05	0.0094	0.0024	0.0618	0.417
D Li/Li/V	1.19	1.068	0.05	0.0094	0.0041	0.0635	0.445
E Li_2O/He/FS	1.14	1.067	0.05	0.0094	0.0029	0.0623	0.270
F $LiAlO_2$/He/FS/Be	1.16	1.067	0.05	0.0094	0.0009	0.0603	0.350
G Li/He/FS	1.17	1.067	0.05	0.0094	0.0030	0.0624	0.234
H Flibe/He/FS/Be	1.29	1.067	0.05	0.0094	0.0017	0.0611	0.812
I $LiAlO_2/H_2O$/FS/Be	1.22	1.070	0.05	0.0094	0.0009	0.0603	0.557

largest uncertainty is associated with Δ_G^2. This term is affected most by

1. tritium fractional burnup in the plasma
2. required doubling time
3. tritium processing efficiency.

III.G. Three-Dimensional Tritium Breeding Analysis

A three-dimensional tritium breeding analysis was performed for the nine TMR designs and seven tokamak designs rated as "$R = 1$" in the BCSS. These designs include the combinations of breeder, coolant, and structural materials presented in Tables XI and XII. All the ternary ceramic (TC) designs and the Flibe designs employ neutron multipliers in various forms and thicknesses.

The analysis was performed with a continuous energy Monte Carlo code, MCNP, and its associated cross-section libraries based on the latest ENDF/B-V data. For each design, 10 000 neutron histories were generated, resulting in a typical statistical error of ±1% or less in the estimate of total TBRs.

The basic geometrical configurations modeled for the study are based on the MARS design for the TMR concepts and on the STARFIRE design for the tokamak concepts. The reference limiter used for the tokamak analysis is taken from the FED/INTOR phase-2A study, i.e., a bottom limiter constructed of a Cu-2Be alloy with a water coolant and beryllium coating. An alternate limiter design that is used for the two liquid-lithium blanket concepts, Li/Li and Li/He, employs a lithium-cooled V-15 Cr-5 Ti heat sink along with a beryllium coating. The geometrical configuration of the radio-frequency waveguides that penetrate perpendicularly through the lower outboard sector is modeled after the STARFIRE design, i.e., HT-9 grid structure cooled by water.

To account for the D-T fusion taking place in the end plug regions of TMRs, the TBRs calculated by MCNP for the TMR blankets have been reduced by 2.5%. In addition, the blanket area lost for startup heating has been estimated to be ~0.5%. Thus, the overall breeding adjustment required for the TMR designs is −3% of the MCNP estimates.

III.H. Shielding Assessment

A shielding assessment was performed to determine shielding materials, compositions, arrangement, and thickness for each blanket concept. Two shielding criteria were adopted for this assessment:

1. Workers are permitted in the reactor hall 1 day after shutdown.
2. Superconducting coils are required to function for 150 MW·yr/m^2 D-T neutron exposure at the first wall.

The occupational exposure is limited to 0.5 mrem/h based on working 8 h/day and 40 h/week. The personnel exposure criteria were used to size the outboard bulk shield for tokamak reactors and the shield thickness between the central cell coils for TMRs. A shielding criterion of 10^{10} rad to the magnet thermal insulation was used to size the bulk shield in the inboard section of the tokamak reactors and the central cell sections under the coils for TMRs. As a result of this criterion, all other nuclear responses do not exceed any design limit for the superconductor materials or the copper stabilizer. Also, the nuclear heating in the winding material is ~0.1 mW/cm^3, which is very close to the optimum design conditions for TMRs and quite satisfactory for the design of the TF coils in tokamak reactors.

A steel-type shield is used for all designs to permit accurate comparison between the different blanket concepts. The shielding materials are type Fe-1422 steel as a bulk shielding material, B_4C as a neutron absorber, H_2O as a moderator and coolant, and lead as a gamma-ray absorber.

All calculations were performed with the discrete ordinates code ANISN (Refs. 2, 3, and 4) with S_8 symmetric angular quadrature set and P_3 Legendre expansion for the scattering cross sections. A 67-multigroup cross-section set (46 neutrons and 21 photons) collapsed from the controlled thermonuclear reactor library was used for ANISN calculations. The MACKLIB was employed to calculate the nuclear response functions (nuclear heating, radiation damage, gas production, etc.). The plasma and the first-wall radii were used from STARFIRE and MARS. Table XIII gives the shield thickness and the blanket energy multiplication factors (EMFs) for each concept based on the above criteria.

III.I. Activation/Waste Management

The activation of five structural materials and seven coolant/breeder/multiplier materials in a common reference neutron environment was calculated with the FORIG activation code.[1] The reference environment was the neutron flux and spectrum at the first wall of the MARS reactor. The structural materials were PCA, HT-9, modified HT-9, Tenelon, and V-15 Cr-5 Ti. The coolant/breeder/multiplier materials were $LiAlO_2$, 17Li-83Pb, beryllium, Li_2O, lithium, NS, and Flibe. Qualitative comparisons of these activated materials were made with respect to worker protection requirements for gamma radiation in handling the materials and with respect to their classifications for near surface disposal of radioactive waste.

The results of the comparisons follow:

1. All materials require remote handling and shielding during operations and in the first 10 years after removal from a reactor.

TABLE XIII

BCSS Blanket/Shield Dimensions and EMFs for the Reference Tokamak and TMR Blanket Concepts

Blanket Concept Breeder/Coolant/ Structure/Multiplier	Blanket/Shield/Total Thickness (cm)		Blanket EMF
	Inboard	Outboard	
Tokamak			
Li_2O/He/FS	41/73/114	85/102/187	1.223
$LiAlO_2$/He/FS/Be	41/74/115	70/116/186	1.280
$LiAlO_2$/H_2O/FS/Be	35/70/105	70/99/169	1.372
$LiAlO_2$/NS/FS/Be	51/60/111	51/112/163	1.323
Flibe/He/FS/Be	41/75/116	85/99/184	1.511
Li/He/FS	61/64/125	120/104/224	1.279
Li/Li/V	64/62/126	75/95/170	1.272
TMR			
Li_2O/He/FS	68/55/123	68/104/172	1.228
$LiAlO_2$/He/FS/Be	58/64/122	58/114/172	1.291
$LiAlO_2$/H_2O/FS/Be	70/45/115	70/95/165	1.386
$LiAlO_2$/NS/FS/Be	51/59/110	51/108/159	1.316
Flibe/He/FS/Be	85/47/132	85/95/180	1.549
Li/He/FS	108/52/160	108/101/209	1.270
Li/Li/V	80/48/128	80/80/160	1.259
Li/Li/FS	80/48/128	80/80/160	1.313
LiPb/LiPb/V	90/40/130	90/75/165	1.294

2. At 100 years after removal from a reactor, only Li_2O, lithium, and Flibe can be handled by workers without special protection.
3. Near surface disposal can be used for V-15 Cr-5 Ti, modified HT-9, Tenelon, beryllium, Li_2O, lithium, and Flibe.
4. Special processing is required before near surface disposal can be used for PCA, HT-9, $LiAlO_2$, 17Li-83Pb, and NS.
5. Current regulations for near surface disposal of radioactive wastes (10CFR61) will have to be amended to cover the basic performance requirements for waste disposal sites for fusion waste.

III.J. Electromagnetic Effects

Electromagnetic forces on the first wall and blanket of the TMR are small for blankets cooled by water or helium, moderate for liquid LiPb, and significant but manageable for liquid lithium. For a tokamak, the forces are significant but manageable for all the concepts.

IV. DESIGN CONCEPTS

The blanket concepts grouped by coolant type—liquid metal, helium, pressurized water, and NS—are summarized below.

IV.A. Self-Cooled, Liquid-Metal Blanket Concepts

The use of the same liquid metal as both tritium breeder and coolant greatly simplifies both design and materials considerations since the blanket requires only a structure material and a coolant breeder. Coolant breeder compatibility/reactivity is not a factor and structure compatibility considerations are less restrictive. Heat removal requirements are also less complex because most of the nuclear heating is deposited directly in the breeder coolant. Lithium and 17Li-83Pb both provide relatively high tritium breeding capability with LiPb having the advantage. Tritium recovery with relatively low tritium inventory is feasible. Lithium has an advantage with respect to tritium recovery while LiPb has potentially lower tritium inventories. Effects of radiation on breeder materials are not important considerations for liquid metals. There are important constraints related to the use of liquid metals in the blanket of a fusion reactor. For example, compatibility between the coolant and structural material limits the allowable coolant-to-structure interface temperature. The pressure drop of a liquid metal flowing through a transverse magnetic field is much higher than that in the absence of a magnetic field, leading to a requirement for relatively high-strength structural materials. Minimizing these pressure drops while providing for adequate removal of first-wall surface heat fluxes and bulk nuclear heating is a complex and challenging design task. The proposed design approach involves the incorporation of the manifold into the blanket. Reactivity of lithium with air and water is an important design consideration. Nonwater-cooled in-vessel components, e.g., limiters, are essential for acceptable safety in lithium blankets. A nitrogen reactor room environment is also suggested to provide improved safety ratings for the lithium-cooled concepts. Special tritium barriers and/or double-walled steam generators are necessary to adequately contain tritium in the LiPb blankets.

IV.A.1. Final Rankings for Self-Cooled, Liquid-Metal Concepts

A summary of final rankings for the various liquid and solid breeder blanket concepts is given in Tables XIV.A and XIV.B for tokamaks and TMRs. In some cases subgroupings, e.g., 2A or 2B, were used to further refine the rankings. In general, the blanket designs of a TMR are ranked higher than those of a tokamak reactor for the same coolant/structural material combination. This is the result of less stringent MHD design requirements for a TMR compared to that of a tokamak reactor. Liquid lithium, owing to its superior thermophysical properties, is a better coolant than LiPb. From an engineering design point of view, the vanadium alloy is a better structural material than either FS or PCA since the vanadium alloy has both

TABLE XIV.A

Ranking of Tokamak and TMR Blanket Concepts*
Liquid-Metal and Molten Salt Breeder Concepts

Concept	PCA	FS	Vanadium
A. Outboard Blanket Same as Inboard			
Li/Li	2B/2A	2A/1	1/1
LiPb/LiPb	3	2B	2A/1
Li/H_2O	3	3	3
Li/He	2A	1	2B
Li/Na	3	3	3
Li/NS	3	3	3
LiPb/H_2O	2B	2B	2B
LiPb/He	2B	1B	2B
LiPb/Na	3	3	3
LiPb/NS	2B	2A	3
B. Liquid-Metal Outboard Blanket Different Inboard Blanket[a]			
Li/Li: –/He	2A	2A	1B
LiPb/LiPb: –/He	2B	2A	2A
LiPb/LiPb: –/H_2O	2B	2A	2A
C. Either A or B using more than one structural material in the same blanket (FS for liquid-metal containment)			
LiPb/He	2A	---	---
D. Molten Salt Breeder			
Flibe/He	3	3	3
Flibe/He/Be	1B	1	2B
Flibe/He/Pb	2B	2A	2B

*Same ranking for tokamaks and TMRs except where two numbers are listed; where different, the tokamak ranking is listed first.
[a]Concepts considered for tokamak only.

TABLE XIV.B

Ranking of Tokamak and TMR Blanket Concepts
Solid Breeder Concepts

Concept	PCA	FS	Vanadium
Li_2O/H_2O	2B	2B	2B
Li_2O/He	2A	1	3
Li_2O/NS	2A	2A	3
Li_8ZrO_6/H_2O	2B	2B	2B
Li_8ZrO_6/He	2B	2B	3
Li_8ZrO_6/NS	2B	2B	3
Li_2O/H_2O/Be	2A	2A	2A
Li_2O/He/Be	2A	2A	3
Li_2O/NS/Be	2B	2B	3
Li_8ZrO_6/H_2O/Be	2B	2B	2B
Li_8ZrO_6/He/Be	2B	2B	3
Li_8ZrO_6/NS/Be	2B	2B	3
Li_2O/H_2O/Pb	2B	2B	2B
Li_2O/He/Pb	2B	2B	3
Li_2O/NS/Pb	2B	2B	3
TC/H_2O	3	3	3
TC/He	3	3	3
TC/NS	3	3	3
TC/H_2O/Be	1B	1	1B
TC/He/Be	1B	1	3
TC/NS/Be	1B	1	3
TC/H_2O/Pb	2B	2B	2B
TC/He/Pb	2A	1B	3
TC/NS/Pb	2A	1B	3
SB/He/Be: –/H_2O[a]	2A	2A	2B

[a]Solid breeder (SB) with neutron multiplier outboard, nonbreeding inboard.

a higher allowable structural temperature and a higher allowable coolant-to-structure interface temperature.

IV.A.2. Reference Designs for Concepts Ranked R = 1 (Tokamak and TMR)

Three TMR concepts (LiPb/LiPb/V, Li/Li/V, and Li/Li/FS) and one tokamak concept (Li/Li/V) were ranked $R = 1$, and were given a full comparative evaluation with all other $R = 1$ concepts. Because of the major differences in the relevant parameters between a tokamak and a TMR (Table III), the MHD, heat transfer, and structural material requirements for a tokamak blanket are much more stringent than for a TMR blanket. This had a very strong impact on the design configurations for the blankets.

The reference design for the tokamak reactor is the poloidal/toroidal flow module shown in Fig. 2. This reference design is composed of slightly slanted poloidal manifolds and relatively small toroidal channels. Each manifold supplies a number of toroidal channels. The coolant velocity in the toroidal channels is relatively high whereas that in the poloidal manifolds can be maintained at low values. Consequently, sufficient cooling of the first wall can be achieved without significantly increasing the total pressure drop through the blanket since the single largest pressure drop is due to the poloidal flow through the manifold, which is perpendicular to the toroidal magnetic field.

The reference design for the TMR is similar to that of the MARS design.[4] This design is chosen primarily because of its simplicity, which outweighs some of its drawbacks such as large void fractions in the blanket and a relatively poor heat transfer capability near the first wall. However, adequate cooling of the first wall is still achieved with moderate pressure drops

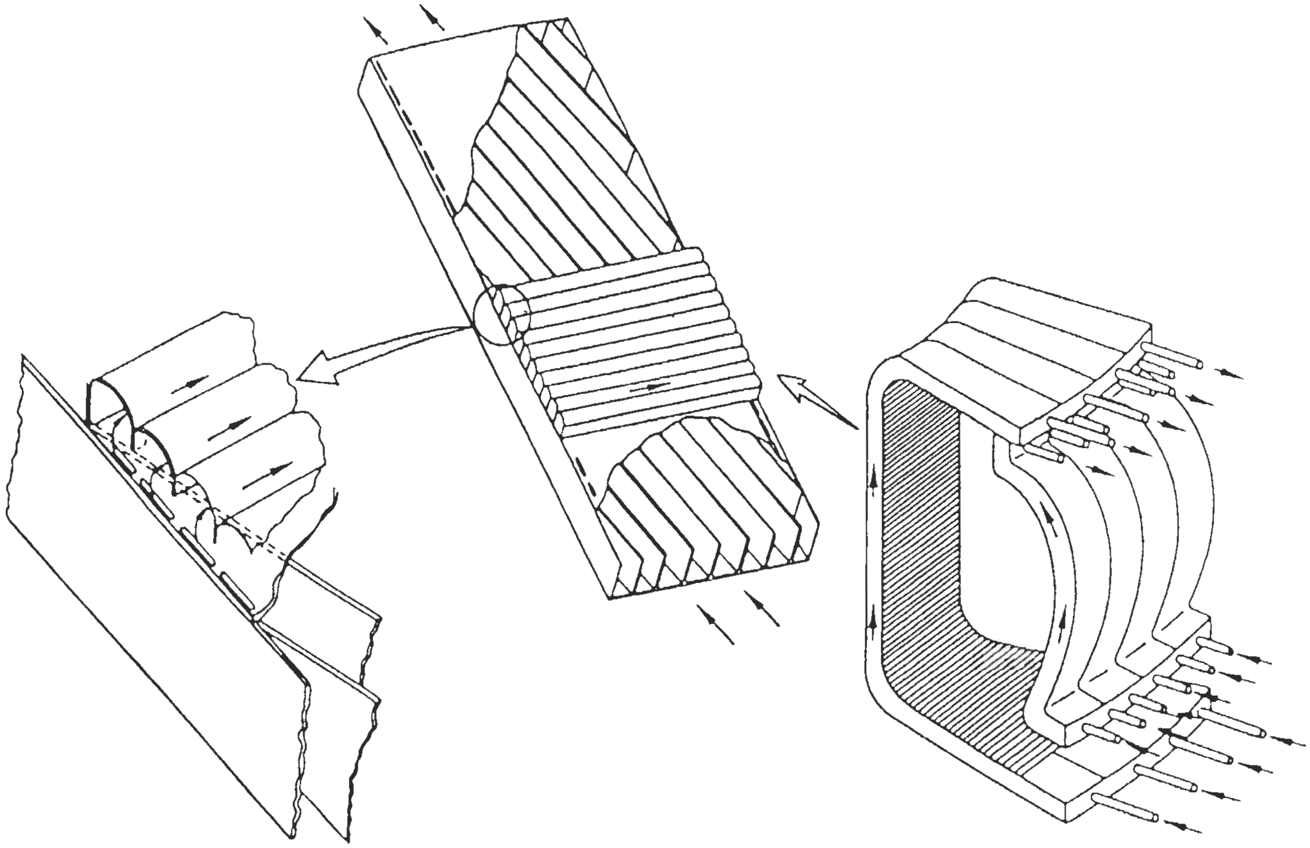


Fig. 2. Schematic of the reference design for the self-cooled, liquid-metal blanket (poloidal/toroidal flow) of a tokamak reactor.

since the surface heat flux in the TMR first wall is relatively small.

Critical issues and design constraints associated with the self-cooled liquid-metal concepts include

1. liquid-metal MHD constraints (pressure drop and heat removal)
2. corrosion limitations
3. reactivity of lithium
4. tritium recovery and containment for LiPb
5. high mass and cleanup of LiPb.

IV.B. Helium-Cooled Blanket Concepts

The principal advantages of helium as a blanket coolant derive from its chemical inertness and virtual transparency to neutrons. Helium is a gas and there are no phase changes in the temperature range of interest. Helium is also nonmagnetic and nonconductive, an additional advantage for magnetically confined systems. It is used as a heat transfer medium for fission reactors; thus, systems for purity control, including tritium recovery, have been developed. There are also advantages in reactor maintenance.

The principal disadvantage for all gas coolants is their low volumetric heat capacity, which leads to the need to operate the helium pressure in the range of 4 to 8 MPa. The pumping powers for the helium-cooled designs in the BCSS are high (~2 to 5% of blanket thermal power) compared to the other coolants. The heat transfer coefficient obtainable at reasonable velocities in helium can be relatively low, leading to relatively high film drops and thus high material temperatures. Despite some commercial usage of helium for reactors, the relative experience in commercial deployment is much less than that of water-cooled technology, particularly in the United States.

IV.B.1. Final Rankings for Helium-Cooled Concepts

A summary of the final ranking for all the helium-cooled concepts examined in the BCSS is presented in Tables XIV.A and XIV.B. Rationale for the rankings is summarized in this section.

In general, the rankings of helium-cooled concepts reflect the relative safety advantages of helium coolant, its neutronics advantages, and the relatively good thermal conversion efficiencies obtainable with the helium outlet temperature achievable with radial coolant flow through the module and the 550°C temperature limits for PCA and FS structure. In most cases, lower rankings for concepts relate primarily to relative disadvantages in other materials, narrow temperature windows, or structural materials limitations.

Lithium zirconate (Li_8ZrO_6) breeder concepts rank considerably lower than concepts with Li_2O or $LiAlO_2$ because of waste management concerns, lower thermal conductivity, and/or lower TBRs. Concepts with PCA structure generally rank lower than those with HT-9 FS because of greater thermal stress constraints for PCA. For concepts with liquid-metal breeders, allowable temperatures for the liquid-metal-to-structure interface are also generally lower for PCA than for FS, which can restrict the allowable system ΔT. Concepts with lead neutron multipliers were generally ranked lower than those with beryllium; the relatively high melting point of lead (327°C) sharply restricts the allowable helium coolant ΔT by raising the required inlet temperature to provide adequate margin against freezing of the lead.

Concepts with vanadium alloy structure were ranked $R = 2B$ for liquid-metal concepts and $R = 3$ for solid breeder concepts because of concerns for oxidation of the structure by oxygen contaminants in the helium coolant stream or from the oxidizing environment associated with the solid breeders.

IV.B.2. Blanket Configuration for Reference Helium-Cooled Concepts

The lobular pressurized-module concept was selected for the reference design for all helium-cooled concepts in the BCSS. All the helium-cooled blanket concepts appear to be equally applicable to the TMRs and tokamak reactors. The blanket internals and pressure boundary configuration would be essentially identical for a given concept, and only the overall mechanical structure would change. The first-wall design for a TMR is simplified by the absence of any significant level of particle erosion or surface heat flux. An integral first wall is used for all of the helium-cooled designs with full flow of the inlet helium directed to the first wall. An internally finned first wall is required for the tokamak, but a simple channel suffices for the TMR versions.

IV.B.2.a. Li_2O/He/FS Concept (Tokamak and TMR). The Li_2O/He/FS concept is shown schematically in Fig. 3. To achieve the maximum volume of solid breeder in the Li_2O/He blanket, a flat plate fuel element geometry was adopted. Solid breeder pellets are clad in HT-9 sheets to form plates. The coolant flows through the 1-mm coolant gaps between breeder plates and maintains the solid breeder temperature distribution within its specified temperature limits. The breeder plates are purged with a separate helium stream with 1% H_2 added, for positive control of tritium extraction. The purge stream operates at 0.1-MPa pressure. This allows the 5-MPa coolant pressure to clamp the cladding onto the Li_2O pellets, giving good thermal contact. It also reduces the purge mass flow rate and avoids concerns about cladding deformation in case of a coolant depressurization accident.

A number of potentially critical issues need to be addressed for the Li_2O/He concepts:

1. irradiation-induced swelling of Li_2O
2. tritium recovery and containment
3. LiOH mass transfer
4. first-wall cooling/stress limits
5. marginal TBR without neutron multiplier
6. helium leakage into plasma.

IV.B.2.b. $LiAlO_2$/He/FS/Be Concept (Tokamak and TMR). The $LiAlO_2$/Be/FS blanket is very similar in configuration to the Li_2O design shown in Fig. 3. The rectangular fuel plate approach for the Li_2O blanket is also used for the $LiAlO_2$. The beryllium needed for adequate tritium breeding design is placed in front of the $LiAlO_2$ plates, in the form of 2-cm-diam cylindrical rods cooled by crossflowing helium. This configuration allows for easy manufacturing and assembly. Furthermore, the rod arrangement provides accommodation of radiation-induced small dimension changes in the beryllium without allowing high stresses to develop. The TBR of the $LiAlO_2$/Be design for a tokamak is only 1.04. Beryllium was used only on the outboard blanket to make the inboard blanket thinner in order to improve its economic performance. In retrospect, it would have been better to include ~10 cm of beryllium on the inboard blanket to achieve a higher TBR, even though economic and safety penalties might have resulted.

Some of the critical issues for this concept that relate specifically to the use of $LiAlO_2$ and beryllium are

1. tritium recovery and containment
2. temperature control of breeder
3. control of tritium from beryllium
4. first-wall cooling/stress limits
5. irradiation damage of beryllium
6. helium leakage into plasma.

IV.B.2.c. Li/He/FS Concept (Tokamak and TMR). For the helium-cooled, liquid-lithium breeder concept, an overall configuration similar to that of the solid breeder designs was used, as shown in Fig. 4. A tubular array of breeder elements was used. Liquid lithium flows slowly through the tubes, allowing tritium recovery external to the blanket. The slow flow

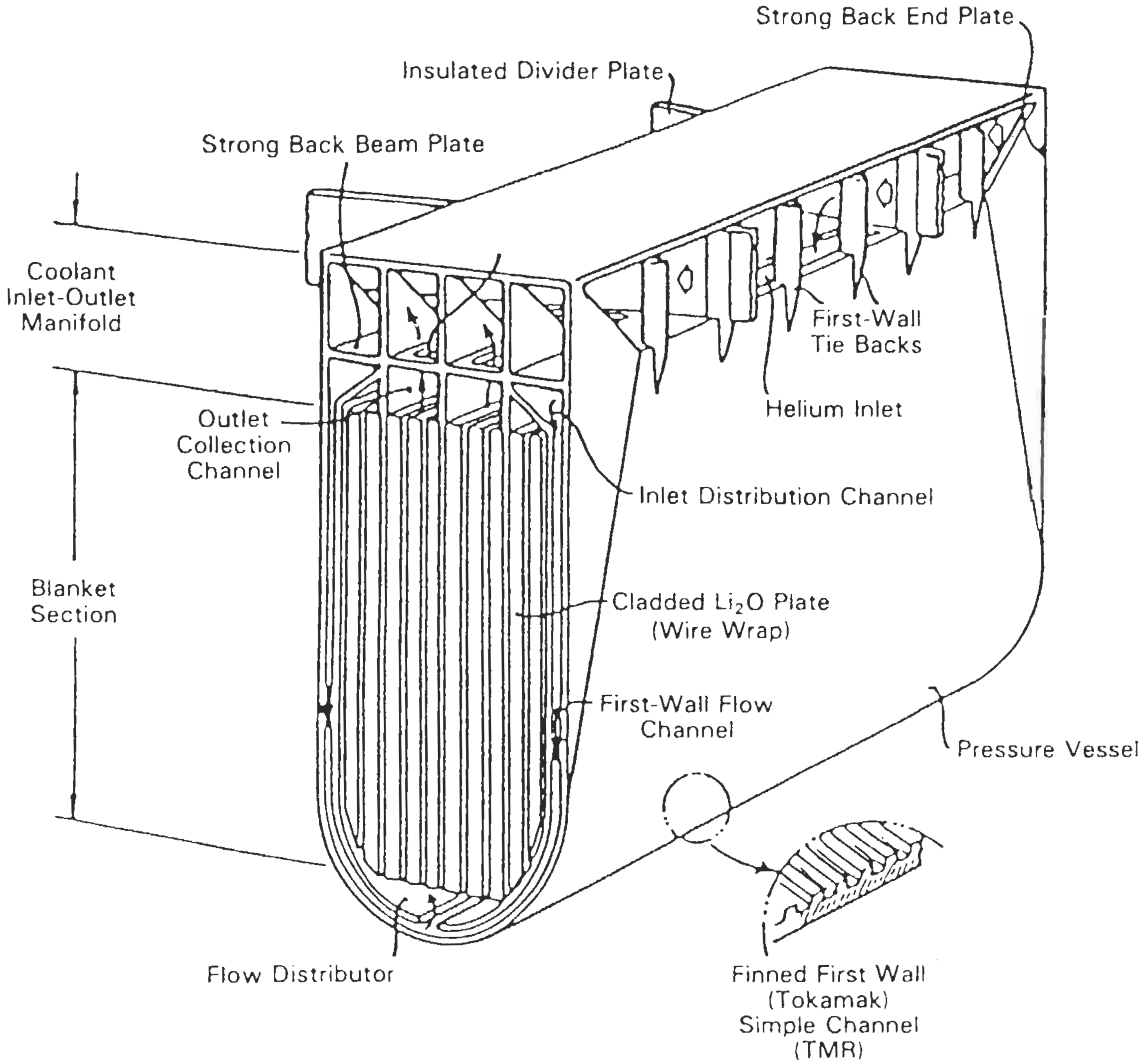


Fig. 3. Li_2O/helium blanket design.

velocity of lithium minimizes MHD effects. Tritium permeation through the breeder tubes into the helium coolant is negligible. The primary source of tritium permeation into the helium is through the first wall.

Critical issues associated with the Li/He/FS concept include

1. corrosion temperature limitations
2. reactivity of lithium
3. first-wall cooling/stress limits
4. tritium containment/recovery
5. coolant leakage into plasma.

IV.B.2.d. Flibe/He/FS/Be Concept (Tokamak and TMR). The breeder in this concept is $2LiF \cdot BeF_2$, the low-melting temperature (363°C) version of the compound commonly known as Flibe. The blanket reference design uses a lobe-shaped module essentially identical to the other helium-cooled designs to contain the 5-MPa helium gas. Helium cools the first-wall and blanket internals. The internals consist of a bed of beryllium balls, nominally 1 cm in diameter, in which neutrons are multiplied and later captured, breeding tritium and releasing energy in exothermic nuclear reactions. Tritium is bred in the molten Flibe salt, which flows slowly (0.1 m/s) in FS tubes. The salt is kept in a reducing form by periodic reaction with beryllium so the tritium will be in the T_2 form. To prevent the tritium from permeating into the helium stream at too high a rate, a tungsten coating on the inside of the tubes is proposed. Tritium is removed from the salt and helium by processing both. Because the solubility of tritium in Flibe is so low, there is a strong driving force for tritium permeation. This requires a high-integrity tungsten permeation barrier. The tritium in the helium is prevented from permeating excessively into the steam system by jacketing the

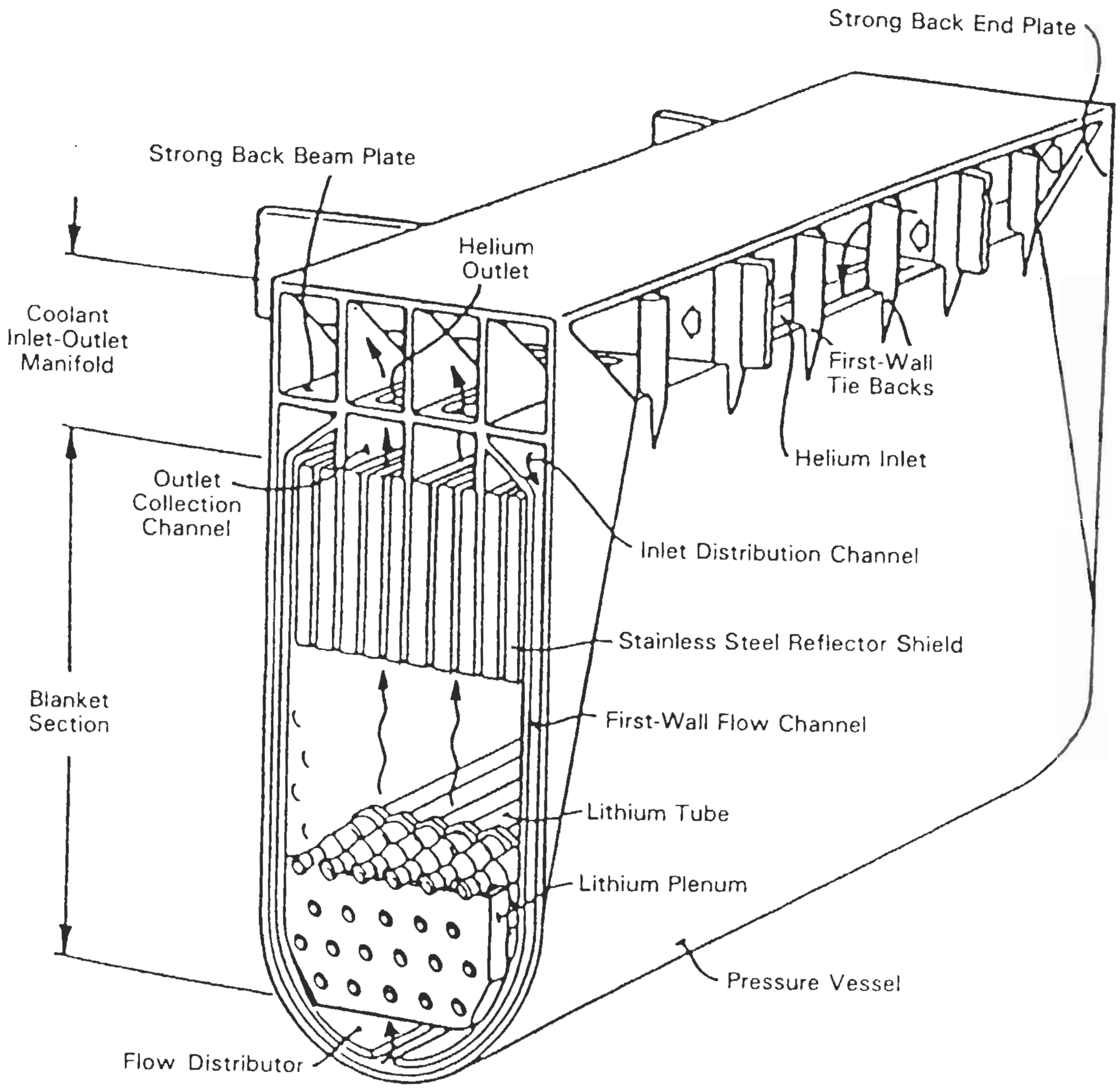


Fig. 4. Liquid-lithium/helium blanket design.

steel steam generator tubes with a 1-mm aluminum jacket.

Beryllium in the form of pebbles was chosen because, by fluidizing, the beryllium can be loaded into the blanket after manufacturing and the beryllium balls can be replaced periodically (~1 to 2 yr) to accommodate radiation-induced swelling. Once the balls have reached their radiation damage lifetime, they can be removed by flowing the blanket for refabrication and recycle.

Critical issues for the Flibe/He concept include

1. tritium containment
2. first-wall cooling/stress limits
3. beryllium resources and recycling losses
4. beryllium pellet radiation damage
5. cleanup of Flibe in event of leak.

IV.C. Water-Cooled Concepts

Blankets with pressurized water coolant have been examined in numerous studies such as STARFIRE. Water has a good materials compatibility data base and excellent heat transfer characteristics and is very low in cost. Power conversion technology for water is well established. The thermal energy conversion efficiency, however, is only moderate, and high-pressure containment is required. In addition, tritium removal is costly and careful chemistry control is required.

IV.C.1. Final Rankings for Water-Cooled Concepts

The rankings for all water-cooled blanket concepts considered in the BCSS are presented in Tables XIV.A and XIV.B. The rationale for those rankings is summarized in this section.

The $LiAlO_2/H_2O/FS/Be$ concept was ranked $R = 1$ and given a comparative evaluation against all other $R = 1$ concepts. The concept gives adequate tritium breeding, and appears to give reasonable performance with no unacceptable safety risks. Ferritic steel is superior to austenitic stainless steel (PCA) for this concept; vanadium alloy is less attractive, and might ultimately not be acceptable because of high tritium permeation rates. The use of sphere-pac fabrication for the breeder should give acceptable breeder temperature predictability. The $LiAlO_2$ breeder appears to be very stable under irradiation within the specified allowable temperature range.

The $Li_2O/H_2O/FS$ concept was given a ranking of 2B. There are major uncertainties in the viability of Li_2O because of radiation-induced swelling. Reactivity of H_2O with Li_2O is a major concern as is reliable containment of pressurized water. In addition, unless a neutron multiplier is included, Li_2O does not appear to be capable of breeding with an adequate margin. If a neutron multiplier has to be introduced in a solid breeder blanket, then $LiAlO_2$ appears to be a better choice overall than Li_2O.

Concepts using molten lead as a neutron multiplier were also ranked $R = 2B$. The use of molten lead in water-cooled concepts leads to a large number of serious design problems that relate to the proximity of lead's solidus temperature (327°C) and the desired operating temperature of the water coolant (280 to 320°C).

All concepts with Li_8ZrO_6 solid breeder were ranked $R = 3$. This concept will not produce sufficient tritium without the addition of a neutron multiplier. Phase transformation at ~660°C, serious waste management problems, and very low thermal conductivity make this breeder even less attractive than other solid breeders for water-cooled blankets.

IV.C.2. $LiAlO_2/H_2O/FS/Be$ Concept Reference Design (Tokamak and TMR)

The blanket configuration is modular in nature, with a lobe-shaped semicylindrical actively cooled first wall. Nominal dimensions are 30 cm wide poloidally (15-cm radius for the first wall) and 70 cm deep, measured radially away from the plasma. The first 20 cm of the breeding zone is a 90:10 volume mixture ratio of beryllium and the $LiAlO_2$ breeder. Both materials are fabricated in sphere-pac form; the individual beryllium and TC spheres are ~100% TD. Packing density for the sphere-pac is 86%. The remaining 32 cm of the breeding zone is $LiAlO_2$, again in sphere-pac form. The remaining 18 cm of the nominal module depth is coolant inlet and outlet manifolds that extend around all the blanket modules to form the blanket sector, with the manifold acting as sector structure. A schematic diagram of the water-cooled concept is given in Fig. 5.

The individual blanket modules contact each other along their side walls from the juncture of adjacent lobes radially back to the manifold zone. The side walls bear against each other, providing mutual support to reduce structural requirements for reacting loads due to the 0.6-MPa maximum internal pressure of the helium purge gas.

The tokamak inboard blanket modules are very similar to the TMR and tokamak outboard modules except for depth. The breeding zone plus the first wall is 28 cm, and the manifold depth is 7 cm. The Be/TC mixture depth is 20 cm as in the outboard modules, with the last 8 cm of the breeding zone being breeder only.

Critical issues associated with the $LiAlO_2/H_2O$ concept include

1. reliability of coolant tubes
2. tritium recovery/containment
3. safety related to pressurized water
4. fabrication/refabrication of breeder
5. limited power variation capability
6. beryllium reprocessing.

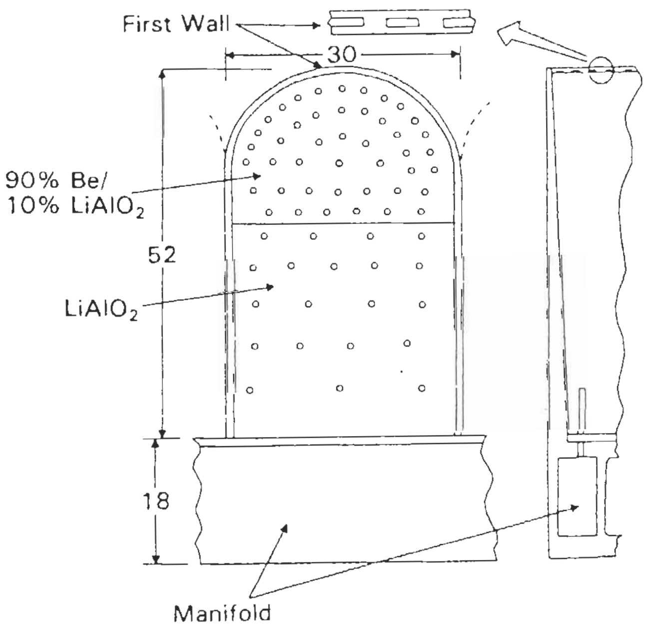


Fig. 5. Reference design configuration for $LiAlO_2/H_2O/FS/Be$ concept for a tokamak. (Dimensions are in centimetres.)

IV.D. Molten-Salt-Cooled Blanket Concept

The two characteristics of blanket coolant that are highly desirable in a fusion reactor are the ability to operate at low pressure with high temperature and a high-heat transfer coefficient. These characteristics are best met by molten salt coolants. The families of NS and nitrate/nitrite salts were specifically considered. The many desirable features of molten salt coolants are offset by some undesirable features and by several uncertainties that cannot be resolved without experiments. The salt selected was an equimolar mixture of $NaNO_3$ and KNO_3 known as draw salt. The reasons for its selection are the data base established from its use in the solar program, its high-temperature stability, and the hope that thermal stability would also result in radiation stability.

IV.D.1. Final Rankings for NS-Cooled Concepts

The NS-cooled concepts for the tokamak and TMR ($LiAlO_2$/NS/FS/Be) were ranked $R = 1$ and underwent comparative evaluation with all other $R = 1$ concepts. The rankings for all other NS-cooled concepts are given in Table XIV.B. A neutron multiplier is required to provide adequate tritium breeding with Flibe. Beryllium is preferrable to lead as the neutron multiplier and FS was selected as the most appropriate structural material.

IV.D.2. $LiAlO_2$/NS/FS/Be Concept Reference Design—Tokamak and TMR

A pod concept was chosen for the tokamak blanket to reduce thermal and swelling stresses and to contain the pressure with the minimum amount of structure. The NS is contained in tubes to minimize its volume fraction and to minimize voltage-enhanced corrosion. This concept is similar in many respects to the water-cooled concept (Fig. 5).

Flow through the coolant tubes and first wall in the pods is toroidal for design simplicity. Thermal-hydraulics considerations result in desirable cooling tube lengths of ~6 m or two average pod lengths. Tubes could be routed back and forth within the pods to achieve this length; however, temperature control and manufacturing simplicity suggest that axial flow through two adjacent pods in series is a better choice. Two independent coolant loops are provided by manifolding and crossing over tubes at the back of the blanket such that alternate tubes are supplied by one coolant loop. This allows removal of afterheat in the event of failure of one of the loops.

The first-wall and pod side walls are actively cooled. This is accomplished by making one side and the first wall of each pod a coolant panel. The first wall is a composite structure with a grooved sacrificial erosion layer. The tritium purge system has an inlet plenum at the front and an outlet plenum at the back of the blanket that supply the sphere-pac breeder with a 1 cm/s flow of helium.

The TMR version of the blanket is very similar with a composite cylindrical first wall loaded in compression to contain the sphere-pac breeder and multiplier and the helium purge gas. The first wall is connected to the back of the blanket at the module ends by semielliptical toroidal end caps. Coolant tubes are routed axially; the 6.32-m-long TMR module does not require that the coolant pass through more than one module. One of the dual coolant loops also supplies the first-wall channels. The tritium purge system is essentially the same as for the tokamak blanket, but with simpler cylindrical geometry.

Critical issues associated with the NS-cooled concept include

1. salt stability
2. activation product control
3. tritium recovery/containment
4. voltage-enhanced corrosion
5. coolant compatibility with beryllium.

V. EVALUATION OF LEADING BLANKET CONCEPTS

An evaluation methodology was developed as part of the BCSS project to compare the leading blanket concepts. Detailed evaluations were performed for the nine leading concepts [seven for the tokamak (see Table IV)] in four areas:

1. engineering feasibility
2. economics
3. safety
4. R&D requirements.

Based on the results of these evaluations, an overall ranking of the blankets was performed to identify those concepts that should provide the focus for the R&D program.

V.A. Engineering Feasibility

The items included under engineering feasibility, listed in Table XV, include important blanket criteria that either deserve separate consideration or do not readily fit under the categories of safety and economics.

V.A.1. Methodology for Engineering Evaluation

The evaluation approach was to determine an overall engineering figure of merit (EFM), defined as the weighted W_i sum of an index I_i for each item listed in Table XV:

TABLE XV

Engineering Evaluation Indexes

Index Name		Weight
1. Tritium breeding and inventory		25
2. Engineering complexity and fabrication		25
3. Maintenance and repair		15
4. Resources		5[a]
5. Power swings		10
6. Increased capability		10
6.1 Increased neutron wall loading	5	
6.2 Higher surface heat flux, higher erosion	5	
7. SU/SD requirements		10

[a]Assumes "go/no-go" material shortage does not exist.

$$\mathrm{EFM} = \sum_i I_i W_i \ , \tag{2}$$

where I_i has a value of 0 to 1 for each item listed. The maximum score is 100. Separate scores for EFM were developed for tokamaks and TMRs. Each of the seven indexes is briefly described.

1. *Tritium breeding and inventory* I_1. This is considered a major feasibility issue and given a weighting of 25 points. An FOM ($= I_1$) was calculated for each concept based on its three-dimensional TBR and steady-state tritium inventory. Uncertainties in estimating actual breeding requirements, in reactor design definition, and in calculation and modeling accuracies were considered.

2. *Engineering complexity and fabrication* I_2. Eight important features of blanket designs were identified that affect complexity and fabrication.[1,5] Concepts were judged as to how well they rated in each area.

3. *Maintenance and repair* I_3. Four general blanket features were identified that impact the reactor operator's ability to maintain the reactor.[1,5]

4. *Resources* I_4. Concepts were rated on their consumption of scarce resources, based on a 1000-GW(electric) fusion plant capacity over a 40-yr span.

5. *Power variation* I_5. The concept's capabilities to permit reactor operation below or above the nominal design points were measured.

6. *Increased capability* I_6. This index measured each concept's ability to (a) operate at higher neutron wall loads (P_{NW}), and (b) accommodate higher first-wall surface heat loads and/or higher particle fluxes; this is particularly important for tokamaks.

7. *Startup/shutdown (SU/SD) requirements* I_7. The SU/SD times required were determined, and any reactor subsystems necessary to permit SU/SD were identified and their added complexity was evaluated.

V.A.2. Results and Conclusions of Engineering Evaluation

The scores for all engineering evaluation indexes are compared in Table XVI for TMR and tokamak concepts.

V.A.2.a. TMR Concepts. The scores for all but the water-cooled concept fall within a relatively narrow range. It seems fairly clear that the water-cooled concept is inferior to the other groups in this evaluation area. The concept scores poorly in the complexity and power variation indexes, and is not outstanding in any of the categories.

The self-cooled, liquid-metal concepts score quite well individually and as a group. They score very well in the complexity category and do relatively well in all other categories except maintenance.

The helium-cooled concepts (Flibe excepted) as a class do only slightly less well than the self-cooled, liquid-metal concepts. The $LiAlO_2$ concept scores below the other two helium-cooled concepts, primarily due to lower scores in the complexity and resources categories that result from the need to add beryllium to the blanket. The Flibe blanket scores considerably higher than the other helium-cooled concepts primarily because of the higher TBR for its reference design; but it scores lower relative to the others in maintenance because of the presence of Flibe and for resources because of the beryllium neutron multiplier.

V.A.2.b. Tokamak Concepts. The distinctions among concept groups become more evident for the tokamak versions, and the spread among scores is much wider than for the TMR concepts.

The self-cooled, liquid-metal blanket (Li/Li/V) is clearly at the top of this group. It does well in the breeding, complexity, resources, and power variation categories, and scores reasonably well in the other three categories.

The helium-cooled concepts do not score as well as their TMR counterparts, and are well below the Li/Li/V concept. This reflects the effects of the economics-motivated need for thin inboard blankets, which in turn affects TBR, and the relative difficulties in handling tokamak first-wall surface heating and particle fluxes with helium coolant. The $LiAlO_2$/He/FS/Be concept would likely have scored significantly higher if the decision had been made to incorporate a beryllium neutron multiplier into the inboard blanket. This would have given it ~4 to 5 points for the breeding category. The Flibe concept does not score as well as its TMR counterpart, because the reference design does not breed as well and the helium coolant has a limited capability for handling increases in surface heat flux q and first-wall erosion thicknesses t_e.

The salt-cooled concept scores relatively well—second in this group—primarily because of its good scores in the breeding, power variation, and heat load increase categories relative to the helium- and water-cooled concepts. The water-cooled concept is at the bottom of the group together with the Flibe concept (if the $LiAlO_2$/He/FS/Be concept is mentally granted ~5 points for breeding). It fares poorly in the complexity, resources, and power variation categories, and stands out only in the $P_{NW}/q/t_e$ increase category.

V.A.2.c. General Conclusions. From the results discussed above, several general conclusions can be drawn from the engineering feasibility evaluation.

1. The overall differences among ranked blanket concepts are considerably larger for tokamaks than for TMRs. These distinctions are brought out by the more difficult problems for blankets in tokamaks due to higher magnetic field strengths, higher surface heating and particle fluxes, and the more complex geometry of the fusion core.

2. Self-cooled, liquid-metal concepts have a slight overall advantage over helium-cooled concepts for TMRs, and where they satisfy design guidelines, have substantial advantages for tokamaks. The helium-cooled concepts do less well in tokamaks primarily because of the need for a thin inboard blanket and the much higher surface heat loads and erosion allowance requirements.

3. The water-cooled solid breeder concept is clearly the least favored for TMRs where its relatively good cooling capabilities do not give it an advantage, and it is in the lowest group for tokamaks.

4. The salt-cooled solid breeder concept does well for both reactor types, which largely reflects the salt's perceived engineering advantages relative to water coolant of very low pressure, higher temperature capability, and better neutronics.

V.B. Economic Evaluation

One of the major evaluation categories for the selection of promising blanket concepts is how each blanket concept affects the overall plant economics, namely, cost of electricity (COE). The COE was chosen as the sole economic parameter because it incorporates direct cost influences, annual cost influences, and technical performance (e.g., power conversion, thermal efficiency, pumping power, and thickness of blanket).

The $R = 1$ ranked blanket concepts were evaluated in the context of both TMR and tokamak reactor power plants. A system performance and economic computer code was written to compare the blanket options. The costs categories that may be affected by the blanket design parameters were the costs of the

TABLE XVI

Engineering Evaluation—Summary of Scores

Reactor		Concept	Breeding and Inventory (25)	Complexity and Fabrication (25)	Maintenance (15)	Resources (5)	Power Variation (10)	P_{NW}, q, t_e, Increased Capacity (10)	SU/SD (10)	Total Score (of 100)	(Score ÷ Highest Score)
TMR	A	$LiAlO_2$/NS/FS/Be	20.3	13.7	9.8	1.5	10	8	5	68.3	1.000
	B	Li/Li/FS	6.6	22.0	7.5	5	9	7.5	5	62.6	0.917
	C	LiPb/LiPb/V	10.4	21.6	6.0	5	10	7.5	5	65.5	0.959
	D	Li/Li/V	11.1	21.6	7.5	5	10	8	5	68.2	0.999
	E	Li_2O/He/FS	6.8	12.5	13.5	5	8.5	10	7	63.3	0.927
	F	$LiAlO_2$/He/FS/Be	8.8	10.6	13.5	2.5	8	10	7	60.4	0.884
	G	Li/He/FS	5.9	16.7	10.5	5	10	10	6	64.1	0.939
	H	Flibe/He/FS/Be	20.3	15.3	9.8	1.5	6	10	5	67.9	0.994
	I	$LiAlO_2$/H_2O/FS/Be	13.9	10.7	11.3	1.5	3	8	7	55.4	0.811
Tokamak	A	$LiAlO_2$/NS/FS/Be	15.4	13.7	9.8	1.5	10	6	5	61.4	0.849
	B	---	---	---	---	---	---	---	---	---	---
	C	---	---	---	---	---	---	---	---	---	---
	D	Li/Li/V	18.4	19.4	10.5	5	10	4	5	72.3	1.000
	E	Li_2O/He/FS	4.0	12.5	13.5	5	8	2	7	52.0	0.719
	F	$LiAlO_2$/He/FS/Be	0	10.7	13.5	2.5	8	2.5	7	44.2	0.611
	G	Li/He/FS	4.8	16.7	10.5	5	9	2.2	6	54.2	0.750
	H	Flibe/He/FS/Be	9.6	15.3	9.8	1.5	6	2.3	5	49.5	0.685
	I	$LiAlO_2$/H_2O/FS/Be	8.3	10.7	11.3	1.5	3	7.5	7	49.3	0.682

first wall, blanket, shield, primary coolant loop, intermediate coolant loop (if required), fuel handling and storage, magnets, turbine plant equipment, and electric plant equipment. Annual costs include capital costs, fuel, operation and maintenance, and scheduled component replacement. The main performance parameters that influence the economic evaluation are the neutron energy multiplication, coolant temperatures, gross thermal energy conversion efficiency, primary coolant pumping power, and thickness of blanket and shield.

This study was structured to evaluate the merits of the blankets when incorporated in a tokamak or TMR. Although the costs are shown for both reactor types, many of the underlying technical performance and economic assumptions and ground rules inherent in the reference STARFIRE and MARS conceptual designs preclude a meaningful cross-comparison of the relative merits of the two reactor types. Valid conclusions should only be drawn for the blanket concepts within a specific reactor type. In both reactor concepts the COE is based on a fixed 80% availability factor and a 6-yr construction period.

V.B.1. Results of Economic Evaluation

The COE is composed of several factors, namely,

$$\mathrm{COE} \sim \frac{\begin{array}{c}\text{total capital cost} \times \\ \text{fixed charge rate} + \text{annual costs}\end{array}}{\begin{array}{c}(\text{thermal power} \times \text{gross efficiency} - \\ \text{recirculating power}) \times \text{availability}\end{array}} .$$

The cost of the blanket components (first wall, blanket structure, breeder, multiplier, reflector, and plenum) is shown in Fig. 6. The overall highest cost blankets were the ones with vanadium structure (C and D), with highly enriched breeders (I), and with high usage of beryllium multipliers (A, H, F, and I). The

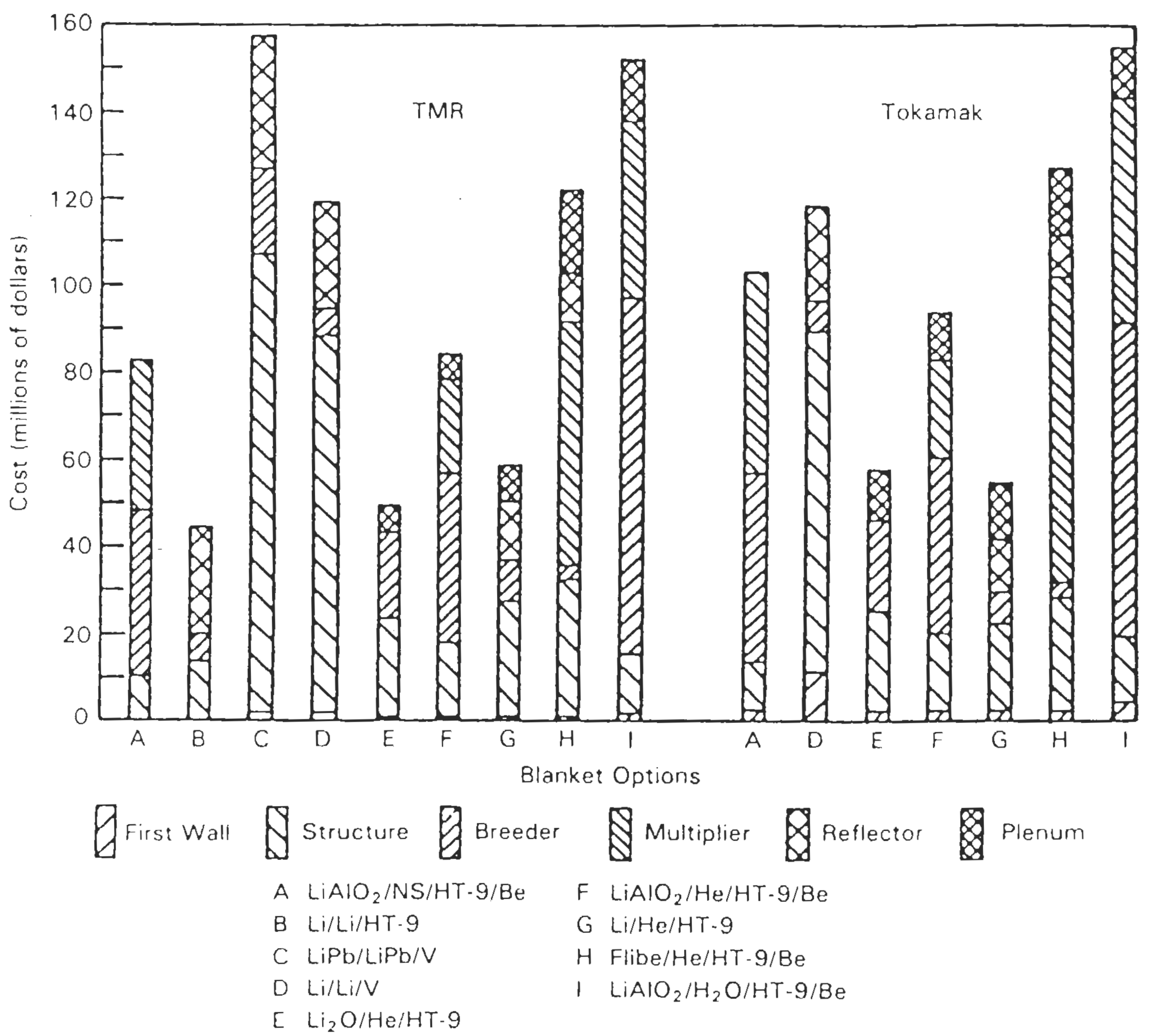


Fig. 6. Blanket cost elements.

total cost of the blankets ranged from $44 to $157 million.

The shield surrounding the blanket was composed of the same materials for all designs, but varied in thickness to achieve the same shielding effectiveness. The shielding costs ranged from $80 to $122 million. The specific cost values can be found in Refs. 1 and 6.

The use of the reference reactors, STARFIRE and MARS, for the BCSS required that specific ground rules and methodology be adopted regarding how the fusion power scaled with respect to the blanket and shield thicknesses. References 1 and 6 define in detail those ground rules and methodologies. Briefly, the fusion power for the TMR is held constant and the blanket and shield thickness variations then influence only the central cell magnet costs. In the tokamak, the magnet costs also changed but variations in the inner blanket and shield thickness influenced the on-axis magnetic field, which increased or decreased the fusion power. Note also that a portion of the TMR plasma energy is deposited on the direct convertor and the halo scraper, which converted the energy to electrical energy at an efficiency different from that of the main heat transport system.

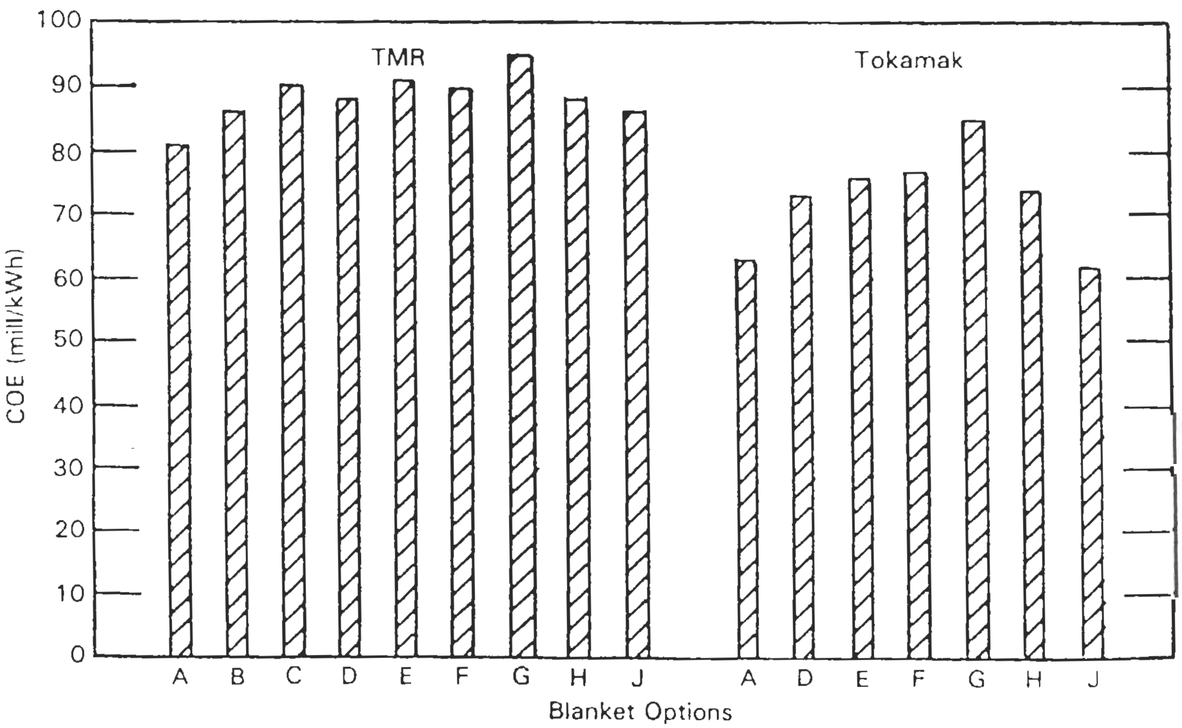
Although there are other costs affected by the blanket choice, the major costs are in the reactor plant equipment, which include the first wall and blanket, shield, magnets, heat transport system, and fuel handling and storage. There are significant variations in the individual elements; however, the variation in the summation of these elements is not that large. The highest cost is only 30% more than the lowest cost. The helium-cooled, lithium breeder (option G) is the highest cost option in both reactor types while NS- and water-cooled designs are the lowest cost designs.

Figure 7 presents the COE values for the $R = 1$ blanket concepts in both reactor types. These results can be summarized as follows:

1. The helium-cooled lithium breeder is the least economically attractive concept in either reactor type.

2. The NS-cooled design is a good candidate for either reactor type.

3. The water-cooled design, because of its thinner inner blanket and shield design, is attractive for the tokamak.



A $LiAlO_2$/NS/HT-9/Be
B Li/Li/HT-9
C LiPb/LiPb/V
D Li/Li/V
E Li_2O/He/HT-9
F $LiAlO_2$/He/HT-9/Be
G Li/He/HT-9
H Flibe/He/HT-9/Be
J $LiAlO_2$/H_2O/HT-9/Be

Fig. 7. Cost of electricity.

4. The remaining self-cooled, liquid-metal, and helium-cooled designs represent moderate cost approaches and are roughly equivalent from an economic performance standpoint.

Several sensitivity studies were performed to assess the impact of variations in materials costs or blanket performance on the economic evaluation. Those factors investigated include

1. material cost for lithium, beryllium, and vanadium
2. blanket lifetime
3. blanket energy multiplication
4. gross energy conversion efficiency
5. blanket/shield thickness
6. coolant pumping power.

The results of the analyses indicate that factors 3 through 6 provide the greatest leverage for improved performance.

V.C. Safety Evaluation

The safety evaluation is intended to measure the relative safety and environmental attractiveness of the various blanket concepts. One possible comparison approach would be to conduct a complete probabilistic risk assessment comprising the entire fuel and facility cycle for each blanket concept; however, restrictions on study resources and knowledge necessitate a more modest approach. Thus, 11 specific evaluation indexes have been established to compare blanket designs. Individual indexes are mixtures of quantitative and/or qualitative information. The comparison is intended to approximate a relative risk assessment comparison to the extent possible by focusing on various specific areas of possible differences among blanket designs.

Each design received a score for each index I_i between 0.0 and 1.0, listed in Table XVII. Each index also has a weighting value W_i, indicating its judged relative importance. The sum of weighting values equals 100. An overall safety figure of merit (SFM) is defined as the weighted sum of index scores

$$\mathrm{SFM} = \sum_{i=1}^{11} I_i W_i \ . \tag{3}$$

Most of the index scores are directly related to specific FOMs by utility functions. The range of these FOMs among designs for specific issues vary from a factor of 3 to a factor of 7 orders of magnitude. Therefore, a 1% change in SFM would translate to much more than a 1% change in safety and environmental risk.

The safety evaluation indexes can be grouped into four major evaluation categories: accident source term characterization (30% of SFM), accident fault tolerance (30%), effluent control (20%), and maintenance and waste management (20%). The balance (60 to 40%) between accidents and nonaccident issues was a compromise between the general public perception that accidents should be weighted high as compared with the actual low weighting for accidents that result from total fuel and facility cycle risk studies for other energy technologies.

The resulting blanket SFM scores and their rank orderings are shown in Figs. 8 and 9. The key trade-off was between tritium effluent control and chemical reaction control. In particular, elemental lithium-bearing designs are generally favorably ranked because they appear much more attractive in the area of tritium effluent control while their chemical reaction problems are generally assumed to be solved by design, e.g., use of an inert building atmosphere. The figures demonstrate that the contribution to the total SFM score from the four major evaluation categories differs substantially among blanket designs. The top-ranked designs, LiPb/LiPb/V (TMR only), Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9, do better overall because they do reasonably well in each category. The bottom-ranked designs, $LiAlO_2/H_2O$/HT-9/Be and $LiAlO_2$/NS/HT-9/Be, do poorly overall because they do reasonably well only in the category of fault tolerance.

The two key areas of uncertainties and assumptions are tritium control and chemical reaction control. The reference results depend on

TABLE XVII

Safety Evaluation Indexes

Index	Index Name	Weighting Value
1	Structure source term characterization	10
2	Breeder/multiplier source term characterization	10
3	Coolant source term characterization	10
4	Fault tolerance to breeder-coolant mixing	6
5	Fault tolerance to cooling transients	6
6	Fault tolerance to external forces	6
7	Fault tolerance to near blanket systems interactions	6
8	Fault tolerance of the reactor building to blanket transients	6
9	Normal radioactive effluents	20
10	Maintenance occupational exposure	10
11	Waste management	10

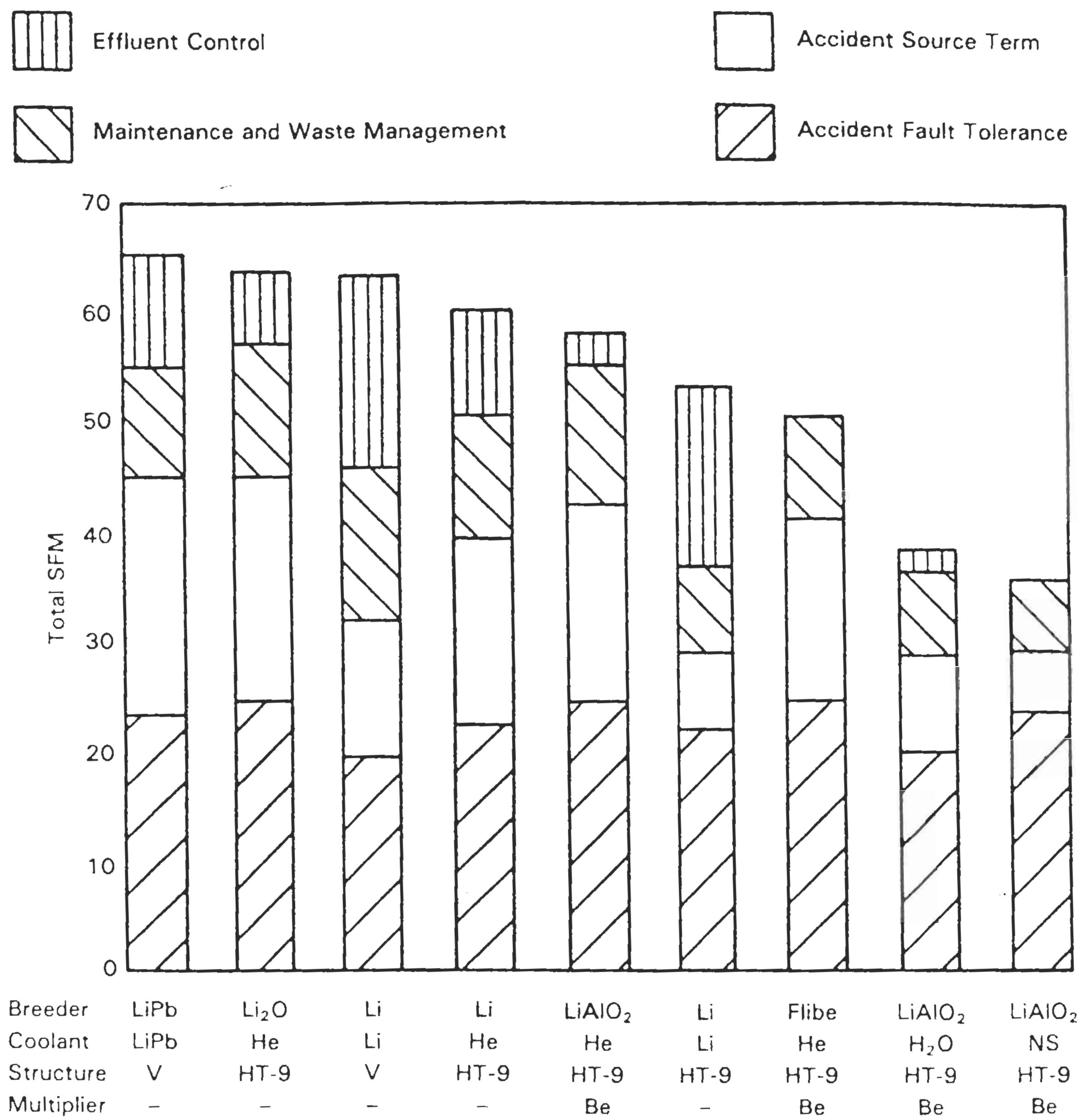


Fig. 8. Safety evaluation results for TMR blankets.

1. use of nitrogen cover gas to reduce the severity of air/metal chemical reactions, e.g., lithium fires and air-vanadium oxidation
2. use of nonwater-cooled limiters to reduce water/metal chemical reactions
3. assumption that water/metal separation is adequate to keep water/metal chemical reaction risk to a low level
4. assumptions and data indicating that tritium control is exceedingly difficult
5. assumptions that some tritium control techniques work.

Given these conditions, the favorable ranking of some of the liquid-lithium designs is less surprising.

The use of beryllium was found to have only a modest penalty because the addition of beryllium toxicity generally has a small impact on breeder/multiplier biological health potential.

The influence of two proposed low-activation steels was examined. Modified HT-9 and Tenelon were found to basically meet the goal of near surface waste disposal, whereas the reference steels, HT-9 and PCA, do not. Basically, the low-activation steels solve the waste disposal problem by eliminating elements that give rise to long-term (>10-yr) isotopes but replace them with elements (tungsten and manganese) that give

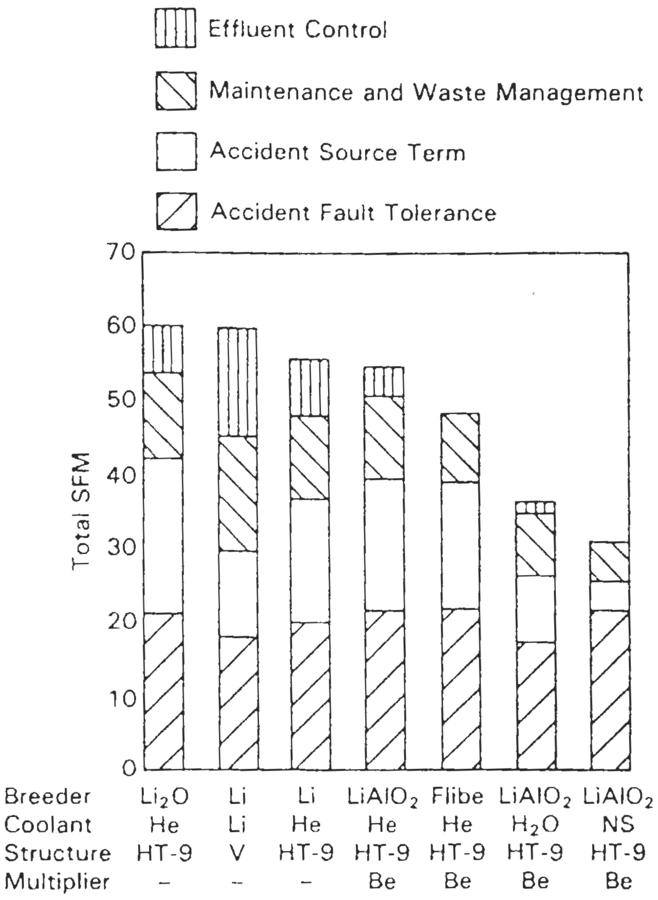


Fig. 9. Safety evaluation results for tokamak blankets.

rise to shorter term isotopes. However, in other activation-relevant areas—accident source term, afterheat, maintenance of structure, and maintenance of cooling systems—the use of these low-activation steels was found to have either an insignificant impact or even a negative one. It appears that modified HT-9 is a net improvement over HT-9 even though the tungsten in modified HT-9 gives rise to substantial amounts of tantalum, tungsten, and rhenium isotopes. It appears that Tenelon is not a net improvement over PCA because the high-manganese content gives rise to high amounts of ^{54}Mn and ^{56}Mn. Future work would be needed to clarify these trade-offs. The V-15 Cr-5 Ti alloy is better from the activation standpoint.

Overall, the most attractive blankets from the safety standpoint are LiPb/LiPb/V, Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9. Future research will either confirm or change the present results with the areas of tritium control and chemical reaction control being paramount. Further details may be found in Refs. 1 and 7.

V.D. Summary of R&D Concept Evaluation Results

The evaluation methodology for R&D provides for an overall figure of merit (RDFM) given by

$$\mathrm{RDFM} = \frac{30}{\mathrm{RDR}} + \frac{1}{\mathrm{RDI}} , \tag{4}$$

where

RDR = parameter that assesses the "risk" in carrying out the R&D for a particular blanket option

RDI = parameter that assesses the R&D "investment" cost or resource requirements for that option.

The factor 30 provides for an approximately equal weighting between the two terms.

V.D.1. R&D Investment Evaluation Results

The RDI parameter is a score made up of a sum of three numbers dealing with schedule X_1, operating cost X_2, and facility needs X_3 such that

$$\mathrm{RDI} = \frac{X_1 + X_2 + X_3}{3} . \tag{5}$$

Table XVIII defines the score for X_1, X_2, and X_3. The results of the RDI evaluation are shown in Table XIX. No significant differences for R&D resource requirements were identified for a given blanket concept between tokamaks and TMRs.

In general, schedule considerations X_1 were dominated by the time to obtain data on neutron irradiation effects on structural materials. Noting that the guideline for this evaluation was to obtain sufficient information to be able to select a blanket for a fusion demonstration reactor, it was the BCSS project judgment that the blankets using FS (i.e., HT-9) could be developed to that point in <10 yr; thus, $X_1 = 1$. Vanadium alloy blankets would require 10 to 20 yr; thus, $X_1 = 2$. The low-activation FSs were also judged to require 10 to 20 yr for development. No blanket option was judged to take longer than 20 yr, given adequate funding.

TABLE XVIII

RDI Categories

Time Scale (yr)	Score X_1	Average Annual Operating Cost (millions of dollars)	Score X_2	Required Facilities, Cost (millions of dollars)	Score X_3
<10	1	<5	1	No new facilities >10	1
10 → 20	2	5 to 20	2	New facilities 10 to 50	2
>20	3	>20	3	New facilities >50	3

TABLE XIX

RDI Evaluation Results

Concept	X_1	X_2	X_3	RDI
Li/Li/V	2	2	2	2.0
Li/Li/FS	1	2	2	1.7
LiPb/LiPb/V	2	3	3	2.7
Li/He/FS	1	2	2	1.7
Li_2O/He/FS	1	2	2	1.7
TC/He/FS/Be	1	3	3	2.3
TC/H_2O/FS/Be	1	3	3	2.3
TC/NS/FS/Be	1	3	3	2.3
Flibe/He/FS/Be	1	3	2	2.0

Note: See Table XVIII for definition of categories for X_1, X_2, and X_3.

All blankets were judged to require annual R&D operating costs of more than $5 million. The blanket concepts employing lithium and Li_2O without neutron multipliers were judged to be in the range of $5 to $20 million/yr; thus, $X_2 = 2$. The LiPb was judged to require more resources than lithium. Similarly, TC blankets with beryllium neutron multipliers would require more resources than Li_2O. Thus, these blankets were rated $X_2 = 3$. Vanadium alloy blankets were judged to require annual expenses similar to FS blankets but to require a longer time as indicated on the X_1 scores.

The facility scores X_3 are similar to the scores for X_2. The BCSS did not consider facility costs related to integrated testing in some type of fusion-based test reactor, except to note that integrated testing of blankets is needed. It was further assumed that a 14-MeV neutron source like the Fusion Materials Irradiation Test (FMIT) Facility would not be available and that maximum use would be made of fissison reactor and ion irradition techniques. While it is clear that the absence of a facility like FMIT would add risk to the R&D program, it appears that it would still be possible to develop the various blanket concepts to the point where a decision could be made on their selection for a demonstration reactor.

The overall score (RDI) indicates that the FS blankets using lithium as a coolant or static lithium with helium for a coolant, and Li_2O with helium as the coolant, have the lowest R&D resource requirement. The blanket with the largest resource requirement is the self-cooled LiPb concept with vanadium alloy as the structure. The blankets employing TC with a beryllium neutron multiplier have somewhat less resource requirements than LiPb.

V.D.2. R&D Risk Evaluation Results

The R&D risk parameter is a summation of key issues for each blanket option where each key issue is rated by the product $C_i \times P_i$ of the consequence C_i of the issue and the probability P_i that the issue will arise for that blanket option. The consequence is rated 1, 2, or 3 (low to severe impact) and the probability is also rated 1, 2, or 3 (low to likely).

Twenty-nine issues were identified for all top blanket concepts developed during the first year of the BCSS. The issues were then combined into a single table (see Table XX) and each issue was given the $C_i \times P_i$ rating for each blanket concept to which it applies. The issues are described in detail in Ref. 1.

In general, the issues are grouped in Table XX into items dealing with structural materials (1 to 5), liquid metals (6 to 11), solid breeders (13 to 17), and neutron multipliers (19 to 22). Several key issues [tritium breeding (12), tritium recovery/leakage/control (18), tokamak first wall (24), coolant leakage (25), and electromagnetic effects (29)] apply to almost all blanket concepts.

There are some important differences between tokamaks and TMRs; thus, an entry is made in Table XX for both concepts (the top entry is for tokamaks). The overall score (RDR) for each blanket concept is then a sum down the column of all the $C_i \times P_i$ values separately for tokamaks and TMRs. The results are shown at the bottom of Table XX.

In summary, the Li/He/FS blanket represents the minimum R&D risk (lowest total of $\Sigma C \times P$) for both tokamaks and TMRs. The Li/Li/FS and Li/Li/V designs are the next lowest risk designs for TMRs, while Li/Li/V is the second lowest risk design for tokamaks. The highest risk designs are the TC concepts with beryllium multipliers, particularly with an NS coolant, for both TMRs and tokamaks.

V.D.3. Composite R&D Evaluation Results

The composite score for the R&D evaluations is shown in Table XXI. The relative ranking of the blanket concepts is also indicated.

V.E. Overall Evaluation

The previous sections presented the results of the engineering feasibility, economic, safety, and R&D evaluations. Tables XXII and XXIII summarize the relative ratings and the rankings of the tokamak and TMR blankets, respectively, in each of the four categories. The rating of the top blanket concept in each category has been normalized to unity.

A list of the overall top-rated blankets is not readily apparent from the individual evaluations. In general, each concept rates high in one or two categories and low in the other categories. For example, the water-cooled lithium aluminate blanket for the tokamak rates highest in economics but near the bottom in both engineering feasibility and safety. Only the lithium breeder/coolant concept with the vanadium alloy structure rates high in all evaluations for both reactor configurations. Because performance cannot be guaranteed for any of the blankets, however, it is important to identify three or four options that should provide the focus for blanket R&D.

Several methods have been considered for utilizing the information in Tables XXII and XXIII to identify the other top-rated blanket concepts that should provide a focus for the R&D effort.

1. Numerical averaging of the normalized ratings in the four areas. This method implies that the relative importance of each category is similar and that the rating in each category provides an appropriate comparative evaluation.

2. Numerical averaging of the normalized ratings for engineering feasibility, economics, and safety. This method implies that the attractiveness of each concept is defined primarily by these three evaluations and that unless resolution of some key issue is prohibitive, the R&D should not be a discriminator.

3. Numerical averaging of the individual evaluations with either the three- or four-factor approaches above, but with nonuniform weighting factors. Two specific weighting factors proposed and considered were 30-30-30-10 and 25-50-25-0 for the engineering feasibility, economic, safety, and R&D evaluations, respectively.

4. A more qualitative comparison of the evaluation results including either the three- or four-factor approach discussed above. In this case the high-rated blankets in each category were given a point and the bottom-rated concepts were given a negative point. This approach penalized those concepts that rate very low in any category.

The final results of all of these methods are quite similar. The three-factor approach that provides equal weighting to the engineering feasibility, economic, and safety evaluations is generally favored. Results obtained by this method with individual ratings normalized to unity for the highest rated concept are summarized in Tables XXII and XXIII. The results obtained by this method of comparison with some additional judgmental considerations have been used to identify the four leading blanket concepts that should provide the focus for the R&D effort.

Clearly, the Li/Li/V concept is the top-rated blanket in this study. The Li/Li/V concept rates superior to all other concepts in the tokamak case, and it rates marginally superior to other concepts for the TMR. The key issues associated with this concept relate to MHD and corrosion problems, the use of a nonreactive reactor room environment (nitrogen), and the feasibility of a nonwater-cooled limiter/divertor.

The Li_2O/He/FS is the top-rated solid breeder concept, ranking considerably below the Li/Li/V concept for the tokamak and relatively close to the Li/Li/V concept for the TMR. In general, the key feasibility issues associated with the Li_2O/He/FS concept are fundamentally different from those for the liquid-metal systems. Major concerns relate primarily to tritium recovery/containment, solid breeder integrity (swelling), and tritium breeding capability. This concept avoids the MHD and corrosion problems associated with liquid-metal systems.

The LiPb/LiPb/V blanket, which was rated high and thus evaluated in detail only for the TMR, rates only marginally below the Li/Li/V concept. This concept consistently ranks high by all methods considered except when R&D is given a high weighting factor. Although these two liquid-metal systems are quite similar with respect to key issues, important differences related to the lower chemical reactivity with the environment and the more difficult tritium containment for LiPb.

The Li/He/FS system is included in the list of concepts partially for judgmental reasons and partially on the basis of the combined quantitative evaluation. This concept ranks third for the tokamak and fourth for the TMR, although it rates only marginally better than several other concepts. The primary justification for including this system relates to the fact that the key feasibility issues are fundamentally different from both the self-cooled, liquid-metal concepts and the solid breeder concepts. The Li/He/FS concept avoids the tritium containment/recovery and breeder stability problems associated with the solid breeder concepts and is less susceptible to the MHD problems associated with the self-cooled, liquid-metal concepts. The primary problem associated with this concept relates to poor economic performance. For this reason, R&D issues specific to this concept should receive high priority only if the feasibility issues or performance characteristics of the other three concepts become less favorable.

The following important observations can be made:

1. Each of these four concepts has a unique set of key issues. Therefore, serious negative results associated with the key feasibility issues will most likely apply to no more than two concepts, leaving two potentially viable options.

2. The evaluation indicates a relative high importance factor for design simplicity. None of the four concepts requires a neutron multiplier.

3. Emphasis is placed on those concepts that appear to provide superior performance. In general, improvements to the lower rated blankets are likely to be applicable to at least one of these higher ranked concepts.

4. Other concepts rated $R = 1$ and $R = 1B$ should be considered backup options. The priority of R&D for these systems will depend on results obtained for the four top-rated concepts.

TABLE

Blanket Concept Key

($R = C \times P$,

Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS
1. Unsatisfactory weld/fabrication of structural materials	3 × 2 3 × 2	--- 3 × 2	--- 3 × 2	3 × 2 3 × 2
2. Excessive embrittlement of structure by hydrogen	---	---	--- 2 × 1	---
3. Unacceptable radiation-induced embrittlement of structure including DBTT concerns	3 × 1 3 × 1	--- 3 × 1	--- 3 × 1	3 × 2 3 × 2
4. Vanadium blanket requires nonvanadium balance of plant (BOP)	2 × 1 2 × 1	---	--- 2 × 1	---
5. Risk from reactivity of structure with environment (inability to use inert atmosphere)	3 × 1 3 × 1	---	--- 3 × 1	---
6. Risk from reactivity of coolant and breeder with environment (inability to use inert atmosphere)	3 × 1 3 × 1	--- 3 × 1	--- 2 × 1	2 × 1 2 × 1
7. Corrosion worse than expected (includes nonvanadium BOP in vanadium designs)	2 × 2 2 × 2	--- 2 × 2	--- 2 × 2	2 × 2 2 × 2
8. MHD effects substantially worse	3 × 2 2 × 1	--- 2 × 1	--- 2 × 1	2 × 1
9. Insulators not developed for liquid-metal blanket	3 × 1 2 × 1	--- 2 × 1	--- 2 × 1	2 × 1
10. Inability to develop nonwater-cooled near plasma components	3 × 2 3 × 1	--- 3 × 1	--- 2 × 1	2 × 2 2 × 1
11. Difficult to meet seismic requirements	---	---	--- 2 × 2	---
12. Inadequate tritium breeding	2 × 1 2 × 1	--- 2 × 1	---	2 × 1 2 × 1
13. Inability to accommodate breeder swelling	---	---	---	---
14. Temperature range for tritium release much less than predicted	---	---	---	---
15. Unacceptable temperature predictability of breeder (e.g., breeder-to-structure gaps are created)	---	---	---	---
16. Unacceptable power variation capability	---	---	---	---
17. Fabrication/refabrication of solid breeder	---	---	---	---
18. Tritium recovery/leakage/control worse than predicted	---	---	--- 3 × 2	3 × 1 3 × 1
19. Loss of beryllium integrity a major problem	---	---	---	---
20. Inability to reprocess beryllium in efficient manner	---	---	---	---
21. Tritium release from beryllium to primary coolant	---	---	---	---
22. Excessive chemical reactivity of beryllium with salt	---	---	---	---
23. Salt stability/decomposition worse than predicted	---	---	---	---

(Continued)

XX

Issues Rating

Tokamak TMR)

	Li_2/He/FS	$LiAlO_2$/He/FS/Be	$LiAlO_2$/H_2O/FS/Be	$LiAlO_2$/NS/FS/Be	Flibe/He/FS/Be
	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2
	2 × 1 2 × 1	2 × 1 2 × 1	2 × 1 2 × 1	2 × 1 2 × 1	2 × 2 2 × 2
	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2	3 × 1 3 × 1	3 × 2 3 × 2
	---	---	---	---	---
	---	---	---	---	---
	---	---	---	---	---
	1 × 2 1 × 2	---	---	2 × 2 2 × 2	2 × 1 2 × 1
	---	---	---	---	---
	---	---	---	---	---
	---	---	---	---	---
	---	---	---	---	---
	3 × 2 3 × 2	3 × 1 2 × 1	3 × 1 2 × 1	2 × 1	3 × 1
	3 × 2 3 × 2	2 × 1 2 × 1	2 × 1 2 × 1	2 × 1 2 × 1	---
	3 × 2 3 × 2	2 × 2 2 × 2	2 × 2 2 × 2	2 × 2 2 × 2	---
	3 × 1 3 × 1	3 × 1 3 × 1	3 × 2 3 × 2	3 × 1 3 × 1	---
	2 × 1 2 × 1	2 × 1 2 × 1	2 × 2 2 × 2	2 × 1 2 × 1	---
	2 × 2 2 × 2	2 × 1 2 × 1	2 × 2 2 × 2	2 × 2 2 × 2	---
	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2	3 × 2 3 × 2	3 × 3 3 × 3
	---	1 × 3 1 × 3	---	---	1 × 2 1 × 2
	---	2 × 1 2 × 1	2 × 2 2 × 2	2 × 2 2 × 2	2 × 1 2 × 1
	---	2 × 2 2 × 2	---	---	2 × 2 2 × 2
	---	---	---	3 × 2 3 × 2	---
	---	---	---	3 × 2 3 × 2	---

(Continued)

TABLE XX

Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS
24. Inadequate performance of grooved first wall (tokamaks only)	2 × 2	---	---	3 × 2
25. Excessive coolant leakage to plasma	2 × 1 2 × 1	--- 2 × 1	--- 2 × 1	2 × 2 2 × 2
26. Difficult cleanup after spill	---	---	--- 2 × 2	---
27. Coolant containment reliability of double-tubed wall less than predicted	---	---	---	---
28. Excessive activation products, difficult to control	---	---	--- 2 × 2	---
29. Electromagnetic effects worse than assumed	3 × 1 2 × 1	--- 2 × 1	---	2 × 1
Totals – RDR				
Tokamak	47	---	---	43
TMR	34	32	48	29

VI. SUMMARY

A 2-yr multilaboratory BCSS initiated by the DOE/OFE and led by ANL was conducted to (a) define a limited number of blanket concepts that should provide the focus for the blanket R&D program, and (b) identify and prioritize critical issues for the leading blanket concepts. The BCSS focused on the mainline approach for fusion reactor development, namely, the D-T-Li fuel cycle, tokamaks and TMRs for electrical energy production, and a reactor parameter space that is generally considered achievable with modest extrapolations from the current data base. The STARFIRE and MARS reactor and plant designs, with nominal first-wall neutron load of 5 MW/m^2, were used as reference designs for the study.

The study focused on

1. development of reference design guidelines, evaluation criteria, and a methodology for evaluating and ranking candidate blanket concepts
2. compilation of the required data base and development of a uniform systems analysis for comparison
3. development of conceptual designs for the comparative evaluation
4. evaluation of leading concepts for engineering feasibility, economic performance, and safety
5. identification and prioritization of R&D requirements for the leading blanket concepts.

In the first phase of the study, nine TMR blanket concepts and seven tokamak blanket concepts were selected for detailed evaluation using the methodologies developed as part of the study (see Table IV).

A detailed methodology was developed for evaluation of these 16 concepts in each of four areas:

1. engineering
2. economics
3. safety
4. R&D requirements.

An overall evaluation was obtained from an equal weighting of the first three evaluations. The study concluded that the R&D should not be a primary discriminator in the selection of the leading concepts.

TABLE XXI

R&D Evaluation for Tokamaks

Blanket Concept	Tokamak		TMR	
	RDFM	Rank	RDFM	Rank
Li/Li/V	1.14	2	1.38	3
Li/Li/FS	---	---	1.52	2
LiPb/LiPb/V	---	---	1.00	8
Li/He/FS	1.29	1	1.62	1
Li_2O/He/FS	1.08	3	1.16	5
TC/He/FS/Be	0.97	5	1.06	6
TC/H_2O/FS/Be	0.93	6	1.00	7
TC/NS/FS/Be	0.89	7	0.95	9
Flibe/He/FS/Be	1.06	4	1.20	4

(Continued)

Li_2/He/FS	$LiAlO_2$/He/FS/Be	$LiAlO_2$/H_2O/FS/Be	$LiAlO_2$/NS/FS/Be	Flibe/He/FS/Be
3 × 2	3 × 2	2 × 2	2 × 2	3 × 2
2 × 2 2 × 2	2 × 2 2 × 2	2 × 2 2 × 2	---	2 × 2 2 × 2
---	---	---	---	2 × 2 2 × 2
---	---	3 × 1 3 × 1	---	---
---	---	---	2 × 3 2 × 3	---
2 × 1	2 × 1	2 × 1	2 × 1	2 × 1
61 53	57 48	60 53	66 58	54 43

TABLE XXII

Tokamak Blanket Ranking

	Engineering	Economics	Safety	R&D	Overall[a]
Li/Li/V	1.000 (1)	0.85 (3)	0.998 (2)	0.886 (2)	1.000 (1)
Li/Li/FS	---	---	---	---	---
LiPb/LiPb/V	---	---	---	---	---
Li/He/FS	0.750 (3)	0.73 (7)	0.925 (3)	1.000 (1)	0.842 (3)
Li_2O/He/FS	0.719 (4)	0.79 (5)[b]	1.000 (1)	0.840 (3)	0.878 (2)
$LiAlO_2$/He/FS/Be	0.611 (7)	0.79 (5)[b]	0.904 (4)	0.754 (5)	0.806 (6)
$LiAlO_2$/H_2O/FS/Be	0.682 (5)	1.00 (1)	0.597 (6)	0.723 (6)	0.805 (7)
$LiAlO_2$/NS/FS/Be	0.849 (2)	0.98 (2)	0.515 (7)	0.692 (7)	0.831 (4)
Flibe/He/FS/Be	0.658 (6)	0.84 (4)	0.807 (5)	0.824 (4)	0.809 (5)

[a]Based on equal weighting for engineering, economic, and safety evaluation results.
[b]Tie in ranking.

TABLE XXIII

TMR Blanket Ranking

	Engineering	Economics	Safety	R&D	Overall[a]
Li/Li/V	0.999 (2)	0.92 (4)[b]	0.974 (3)	0.852 (3)	1.00 (1)
Li/Li/FS	0.917 (7)	0.94 (2)[b]	0.832 (6)	0.944 (3)	0.922 (7)
LiPb/LiPb/V	0.959 (4)	0.90 (6)[b]	1.000 (1)	0.617 (2)[b]	0.982 (2)
Li/He/FS	0.939 (5)	0.85 (9)	0.936 (4)	1.000 (1)	0.943 (4)
Li_2O/He/FS	0.927 (6)	0.89 (8)	0.987 (2)	0.716 (5)	0.970 (3)
$LiAlO_2$/He/FS/Be	0.884 (8)	0.90 (6)[b]	0.905 (5)	0.654 (6)	0.927 (5)
$LiAlO_2$/H_2O/FS/Be	0.811 (9)	0.94 (2)[b]	0.595 (8)	0.617 (7)[a]	0.793 (9)
$LiAlO_2$/NS/FS/Be	1.000 (1)	1.00 (1)	0.552 (9)	0.586 (9)	0.863 (8)
Flibe/He/FS/Be	0.994 (3)	0.92 (4)[b]	0.782 (7)	0.741 (4)	0.924 (6)

[a]Based on equal weighting for engineering, economic, and safety evaluation results.
[b]Tie in ranking.

The results of the study indicate that the self-cooled lithium blanket with a vanadium alloy structure (Li/Li/V) is the top-rated blanket in the study. The Li/Li/V concept rates superior to all other concepts in the tokamak and marginally superior to other concepts for the TMR. The Li_2O breeder with helium coolant and FS structure is the top-rated solid breeder concept, ranking considerably below the Li/Li/V concept for the tokamak and relatively close to the Li/Li/V concept for the TMR. The LiPb/LiPb/V concept, which was rated high and thus evaluated in detail only for the TMR, rates only marginally below the Li/Li/V for this configuration. The Li/He/FS concept is included in the top list partially for judgmental reasons and partially on the basis of the quantitative evaluation. Since this concept rates only marginally better than several other concepts, the primary justification for including this concept in the top grouping relates to the fact that the key feasibility issues are fundamentally different from both the self-cooled, liquid-metal concepts and the solid breeder concepts.

Key issues for each of the leading concepts have been identified and prioritized.

ACKNOWLEDGMENT

This work was supported by the DOE/OFE.

REFERENCES

1. D. L. SMITH et al., "Blanket Comparison and Selection Study—Final Report," ANL/FPP-84-1, Argonne National Laboratory (Sep. 1984).

2. M. A. ABDOU et al., "Blanket Comparison and Selection Study—Interim Report," ANL/FPP-83-1, Argonne National Laboratory (Oct. 1983).

3. C. C. BAKER et al., "STARFIRE—A Commercial Tokamak Fusion Power Plant Study," ANL/FPP-80-1, Argonne National Laboratory (Sep. 1980).

4. G. LOGAN et al., "Mirror Advanced Reactor Study," UCRL-53480, Lawrence Livermore National Laboratory (July 1984).

5. G. D. MORGAN, D. A. BOWERS, and D. E. RUESTER, "Engineering Feasibility Evaluation of Blanket Concepts for Blanket Comparison and Selection Study," *Fusion Technol.*, **8**, 45 (1985).

6. L. M. WAGANER, "Economic Evaluation of the Blanket Comparison and Selection Study," *Fusion Technol.*, **8**, 55 (1985).

7. S. J. PIET, "Safety Evaluation of the Blanket Comparison and Selection Study," *Fusion Technol.*, **8**, 77 (1985).

8. Y. CHA, Y. GOHAR, A. M. HASSANEIN, S. MAJUMDAR, B. PICOLOGLOU, D.-K. SZE, and D. L. SMITH, "Design of Self-Cooled, Liquid-Metal Blankets for Tokamak and Tandem Mirror Reactors," *Fusion Technol.*, **8**, 90 (1985).

9. C. P. C. WONG, R. F. BOURQUE, E. T. CHENG, R. L. CREEDON, I. MAYA, R. RYDER, and K. R. SCHULTZ, "Helium-Cooled Blanket Designs," *Fusion Technol.*, **8**, 114 (1985); see also R. W. MOIR, J. D. LEE, R. C. MANINGER, W. S. NEEF, Jr., A. E. SHERWOOD, D. H. BERWALD, J. H. DeVAN, and J. JUNG, "Helium-Cooled, Flibe Breeder, Beryllium Multiplier Blanket," *Fusion Technol.*, **8**, 133 (1985).

10. G. D. MORGAN, D. A. BOWERS, D. E. RUESTER, J. JUNG, and B. MISRA, "Water-Cooled Blanket Concepts for the Blanket Comparison and Selection Study," *Fusion Technol.*, **8**, 149 (1985).

11. J. D. GORDON, J. K. GARNER, W. G. STEELE, and W. D. BJORNDAHL, "Nitrate-Salt-Cooled Blanket Concepts," *Fusion Technol.*, **8**, 163 (1985).

12. J. W. DAVIS, T. LECHTENBERG, D. SMITH, and F. WIFFEN, "Structural Materials Data Base Assessment for the Blanket Comparison and Selection Study," *Fusion Technol.* (to be published in Sep. 1985).

13. O. CHOPRA, D. L. SMITH, P. F. PORTORELLI, J. H. DeVAN, and D. K. SZE, "Liquid-Metal Corrosion," *Fusion Technol.* (to be published in Sep. 1985).

14. Y. Y. LIU, M. C. BILLONE, A. K. FISCHER, S. W. TAM, and R. G. CLEMMER, "Solid Tritium Breeder Materials," *Fusion Technol.* (to be published in Sep. 1985).

15. S. MAJUMDAR, "Structural Analysis Under the Blanket Comparison and Selection Study," *Fusion Technol.* (to be published in Sep. 1985).

16. D.-K. SZE, A. HASSANEIN, S. PIET, C. WONG, and W. BJORNDAHL, "An Assessment of Problems Associated with Tritium Containment," *Fusion Technol.* (to be published in Sep. 1985).

17. J. JUNG and J. V. FOLEY, "A Comparative Multidimensional Nuclear Analysis of Candidate Blanket Designs for Tokamak and Tandem Mirror Reactor Concepts," *Fusion Technol.* (to be published in Sep. 1985).

18. Y. GOHAR and S. YANG, "Energy Deposition and Shielding Analysis for All Concepts of the Blanket Comparison and Selection Study," *Fusion Technol.* (to be published in Sep. 1985).