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BLANKET COMPARISON AND SELECTION STUDY
FINAL REPORT

FUSION POWER PROGRAM

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ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue
Argonne, Illinois 60439

BLANKET COMPARISON AND SELECTION STUDY - FINAL REPORT

Contributors

D. L. Smith (ANL) G. D. Morgan (MDAC)

M. A. Abdou (UCLA)
C. C. Baker (ANL)
D. H. Berwald (TRW)
M. Billone (ANL)
W. D. Bjorndahl (TRW)
R. F. Bourque (GA)
D. A. Bowers (MDAC)
Y. S. Cha (ANL)
E. T. Cheng (GA)
R. Clemmer (ANL)
R. L. Creedon (GA)
C. F. Dahms (GA)
J. W. Davis (MDAC)
G. Deis (EG&G)[1]
J. DeVa (ORNL)
D. Diercks (ANL)
P. Finn (ANL)
A. Fischer (ANL)
J. Foley (ANL)[2]
K. Furuta (ANL)[3]
J. K. Garner (TRW)
Y. Gohar (ANL)
J. D. Gordon (TRW)
C. Gung (UCLA)
A. Hassanein (ANL)
S. Herring (EG&G)
G. Hollenberg (LLNL)
C. Johnson (ANL)
J. Jung (ANL)
T. Lechtenberg (GA)
J. D. Lee (LLNL)
R. Leonard (ANL)
Y. Y. Liu (ANL)
B. Loomis (ANL)
S. Majumdar (ANL)
R. C. Maninger (LLNL)
H. C. Mantz (MDAC)
R. F. Mattas (ANL)
I. Maya (GA)
T. J. McCarville (TRW)
R. Micich (Grumman)
B. Misra (ANL)
R. W. Moir (LLNL)
W. S. Neef, Jr. (LLNL)
B. Picologlou (ANL)
S. Piet (EG&G)
D. E. Ruester (MDAC)
R. Ryder (GA)
K. R. Schultz (GA)
A. E. Sherwood (LLNL)
K. Shin (UCLA)[4]
W. G. Steele (TRW)
D. K. Sze (ANL)
S.-W. Tam (ANL)
A. Tobin (Grumman)
P. Tortorelli (ORNL)
L. Turner (ANL)
D. Vasquez (GA)
S. Vogler (ANL)
E. Vold (UCLA)
L. M. Waganer (MDAC)
K. Wenzel (ANL)[5]
F. W. Wiffen (ORNL)
C. P. C. Wong (GA)
L. Yang (GA)
S. Yang (ANL)
M. Z. Youssef (UCLA)

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Office of Fusion Energy
U. S. Department of Energy

MASTER

[1]Presently with LLNL.
[2]Graduate Student, Princeton University.
[3]University of Tokyo.
[4]Kyoto University, Japan.
[5]Graduate Student, MIT.

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ABSTRACT

The Blanket Comparison and Selection Study (BCSS) was a two-year, multi-laboratory project initiated by the U.S. Department of Energy/Office of Fusion Energy (DOE/OFE) with the primary objectives of: (1) defining a limited number of blanket concepts that should provide the focus of the blanket R&D program, and (2) identifying and prioritizing critical issues for the leading blanket concepts. The BCSS focused on the mainline approach for fusion reactor (TMR) development, viz., the D-T-Li fuel cycle, tokamak and tandem mirror reactors 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 a nominal first wall neutron load of 5 MW/m^2, were used as reference designs for the study.

The study focused on:

- Development of reference design guidelines, evaluation criteria, and a methodology for evaluating and ranking candidate blanket concepts.

- Compilation of the required data base and development of a uniform systems analysis for comparison.

- Development of conceptual designs for the comparative evaluation.

- Evaluation of leading concepts for engineering feasibility, economic performance, and safety.

- Identification and prioritization of R&D requirements for the leading blanket concepts.

Sixteen concepts (nine TMR and seven tokamak) which 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)

Lithium/Lithium/Vanadium Alloy

Li_2O/Helium/Ferritic Steel

LiPb Alloy/LiPb Alloy/Vanadium Alloy

Lithium/Helium/Ferritic Steel

The primary R&D issues for the Li/Li/V concept are the development of an advanced structural alloy, resolution of MHD and corrosion problems, provision for an inert atmosphere (e.g., N_2) in the reactor building, and the development of non-water cooled near-plasma components, particularly for the tokamak. The main issues for the LiPb/LiPb/V concept are similar to the Li/Li/V blanket with the addition of resolving the tritium recovery issue. Furthermore, resolution of MHD and corrosion problems will be more severe for LiPb/LiPb/V than for the Li/Li/V; on the other hand, the LiPb blanket has reduced concerns with respect to chemical reactivity with environment. The R&D issues for Li_2O/He/FS concept include resolution of the tritium recovery/containment issue, achieving adequate tritium breeding and resolving other solid breeder issues such as swelling and fabrication concerns. Major concerns for the Li/He/FS concept are related to its rather poor economic performance. Improvement of its economic performance will be somewhat concept-dependent and will be more of a systems engineering issue.

BLANKET COMPARISON AND SELECTION STUDY

CHAPTER 1 - INTRODUCTION

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1. INTRODUCTION

Development of a viable blanket system is essential before the feasibility of fusion as a commercial energy source can be established. In addition, a tritium breeding and heat rejection blanket will be required for any fusion test device which produces more than a few ten's of magawatts of power for extended periods of time. The blanket must operate reliably for extended lifetimes in the severe radiation, thermal, chemical, stress, and electromagnetic environment of a fusion reactor core. Since the primary functions of the blanket system relate to energy extraction and tritium breeding for a D-T fuel cycle, both the economics and safety of fusion power will be greatly influenced by the blanket design and performance characteristics. Demonstrating the scientific and engineering feasibility of the blanket system will require extensive research and development (R&D).

Numerous studies conducted worldwide over the past fifteen years have proposed a large number of blanket concepts. Many of these concepts vary in material choices and major design features, and they pose widely different types of critical issues. Ideally, R&D programs should seek to develop a broad data base sufficient for resolving the critical issues for all promising design options in order to select with confidence the most attractive blanket for fusion reactors. Realistically, however, a resource-limited R&D program inevitably must select and focus on only a very limited number of options. An exceedingly important concern with this inevitable approach is the decision-making process to identify the fewest low-risk, high pay-off options. One great difficulty is that the information required for complete technical evaluation of all possible options is usually not available. There is no unique scientific formula for dealing with this situation; there are only guidelines based on expert judgement. The two-year Blanket Comparison and Selection Study was initiated by the U.S. Department of Energy/Office of Fusion Energy in October 1982 to develop these guidelines and to utilize them in identifying a very limited number (3 or 4) of blanket concepts that should receive the highest R&D priority over the next several years.

The objectives of the Blanket Comparison and Selection Study (BCSS) can be stated as follows:

1) Define a small number (3 or 4) of blanket design concepts that should provide the focus of the blanket R&D program. A design concept is defined by the selection of all material (e.g., breeder, coolant, structure and multiplier) and the specification of other major characteristics that significantly influence the R&D requirements.

2) Identify and prioritize the critical issues for the leading blanket concepts.

3) Provide the technical input necessary to develop a blanket R&D program plan. Guidelines for prioritizing the R&D requirements include: a) critical feasibility issues for the leading blanket concepts will receive the highest priority, and b) for equally important feasibility issues, higher R&D priority will be given to those that require minimum cost and short time.

The BCSS was a multilaboratory effort led by Argonne National Laboratory with support from industry, universities, and other national laboratories (see Table 1-1). The executive committee for the project, which is listed in Table 1-2, consisted of managers from Argonne and the lead support organizations. The major support organizations provided teams of experts, while the other special support consisted primarily of individual experts in selected areas. This executive committee served as the decision making body for the project. The evaluation criteria, design guidelines and individual blanket ratings were reviewed in detail by the executive committee throughout the course of the study. Most of the decisions were either a consensus or a strong majority opinion of the committee. Only rarely were decisions made on the basis of a close vote by the committee.

A review committee, listed in Table 1-3, was appointed by DOE's Office of Fusion Energy to provide periodic evaluation of the progress and direction of the study. Reviews were held at the end of the first year of the study, midway through the second year, and at the end of the study.

The objectives and scope of the BCSS are substantially different from past reactor studies. Previous studies were generally concerned with developing specific conceptual designs, whereas the focus of the BCSS was on concept

TABLE 1-1. BLANKET COMPARISON AND SELECTION STUDY TEAM

LEAD LABORATORY
Argonne National Laboratory
MAJOR SUPPORT ORGANIZATIONS
McDonnell Douglas Astronautics Company
GA Technologies, Inc.
TRW, Inc.
EG&G Idaho, Inc.
Lawrence Livermore National Laboratory[a]
University of California, Los Angeles[a]
SPECIAL CONTRIBUTORS
Grumman Aerospace Corporation[b]
Energy Technology Engineering Center[b]
Westinghouse Electric Corporation[b]
University of Wisconsin[b]
Rensselaer Polytechnic Institute[b]
Hanford Engineering Development Laboratory[b]
Oak Ridge National Laboratory

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

TABLE 1-2. BCSS PROJECT EXECUTIVE COMMITTEE

D. L. Smith (ANL)	Project Manager
G. D. Morgan (MDAC)	Deputy Project Manager
M. A. Abdou (UCLA)[a]	
C. C. Baker (ANL)	
J. Gordon (TRW)	
R. Moir (LLNL)	
S. Piet (EG&G)	
K. Schultz (GA)	
D. K. Sze (ANL)	

[a]Project Manager FY 1983.

TABLE 1-3. BCSS REVIEW COMMITTEE

R. Krakowski (LANL) - Chairman
D. Cohn (MIT)
C. Flanagan (W/FEDC)
R. Gold (W)[a]
C. Henning (LLNL)
G. Kulcinski (U of WI)
R. Little (PPPL)
J. Scott (ORNL)
F. Garner (HEDL)[b]

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

selection. Therefore, the project organization and emphasis were carefully planned at the initial stage to best serve the purpose of the study. A few examples illustrate the point. Blanket designs were developed in the project to serve as a tool (not a goal) for identifying the issues and facilitating comparison of blanket options. Therefore, while the blanket designs are pursued in sufficient depth to permit meaningful comparisons, minor details that do not represent significant issues are not considered. In contrast, key issues related to the feasibility and performance of blanket concepts were evaluated to the maximum possible extent permitted by the resources of the project. Wherever possible, key feasibility issues that are relatively independent of the design were evaluated prior to pursuing the design details. In some cases, the results of the key issues evaluation were negative and the resources required for developing designs of the affected blanket concept were conserved. An important aspect of this effort was to develop and evaluate all concepts on an equal basis. This included use of a common data base wherever possible and normalization of the relative conservatism associated with the design and performance characteristics of each concept.

The approach used in the BCSS involved the following:

- Develop reference design guidelines.

- Identify concepts for consideration.
- Develop detailed evaluation criteria and methodology for ranking concepts.
- Compile materials data base and develop uniform systems analysis.
- Develop conceptual designs for evaluation purposes.
- Identify critical feasibility issues for each concept.
- Evaluate blanket concepts.
- Identify and prioritize R&D requirements for leading concepts.

The BCSS Project task organization chart, Fig. 1-1, includes three major areas, viz., Project Tasks, Special Issues and Blanket Design. The Project Tasks include such areas as design guidelines, evaluation criteria; safety engineering, economics and R&D evaluation; and special tokamak - TMR considerations. Special Issues Tasks provided a consistent data base and analysis methodology to be used for all design concepts. The Blanket Design Tasks developed reference designs for the various blanket concepts.

The BCSS is focused on the mainline approach for fusion reactor development. Thus, the study is limited to the deuterium-tritium fuel cycle, tokamak and tandem mirror reactors for electrical energy production, and the reactor parameter space of each that is generally believed to have a reasonable probability of being achievable. The STARFIRE and MARS reactor and plant designs were used as reference designs for this study. Alternate confinement concepts that may require a totally different blanket approach are not considered. Likewise, exotic blanket concepts with no data base for meaningful technical evaluation are not included in the study.

It was recognized from the outset of the study that considerable effort had to be devoted to developing a comparison methodology and a set of evaluation criteria which would facilitate the primary goal of the study; namely, the selection of a limited number of a promising blanket concepts that should

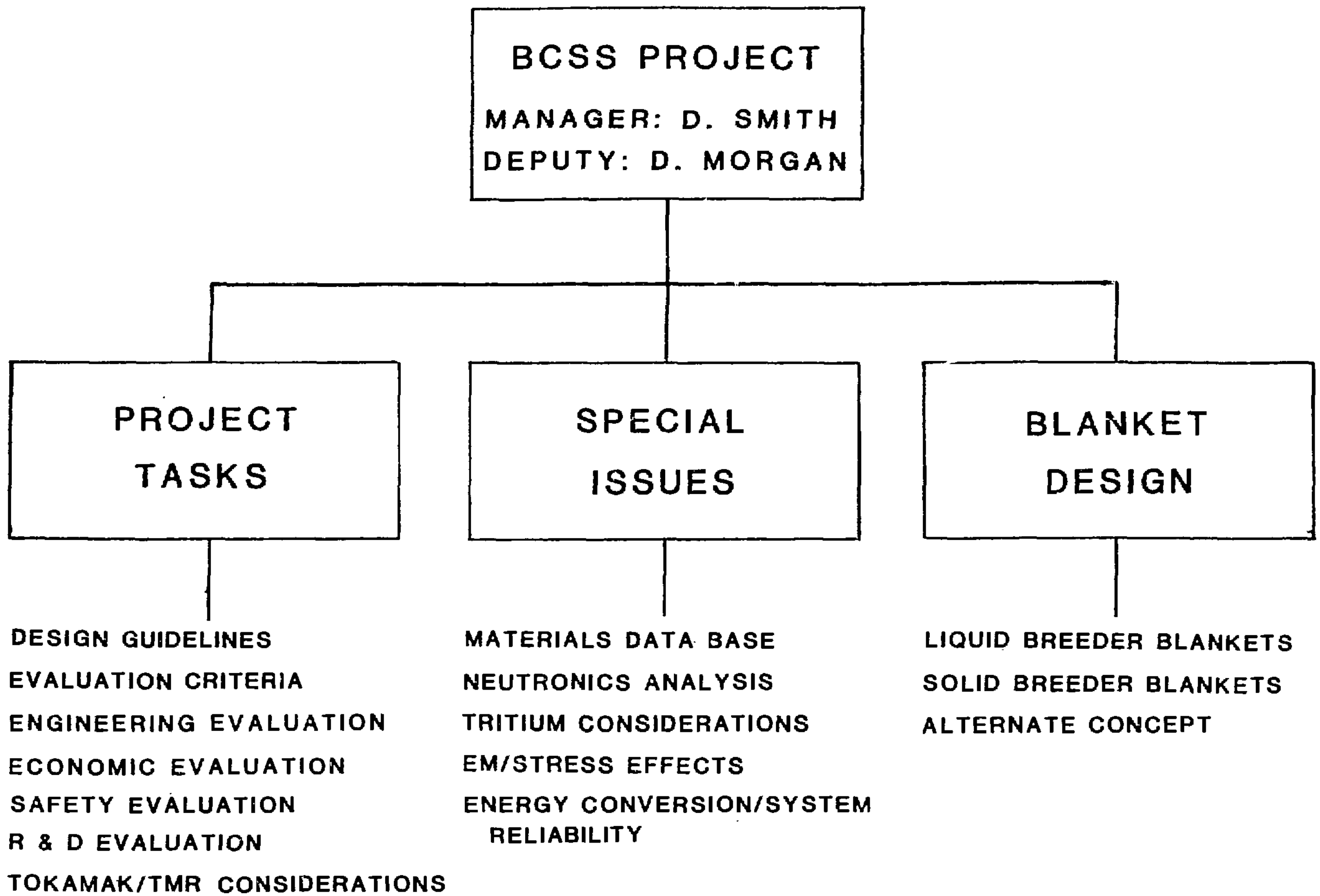

Figure 1-1. BCSS Project task organization chart.

be the focus of blanket R&D. Detailed evaluation criteria were developed for comparison of the blanket concepts in the following four areas:

- Engineering Feasibility
- Economics
- Safety
- R&D Requirements

A more qualitative judgement approach was used to narrow the large number of concepts considered (>100) down to less than ten concepts. The detailed evaluation criteria developed in this study was then applied to the leading concepts to identify the top concepts to be recommended as a focus for the near term R&D effort. This detailed evaluation methodology can be applied to any future blanket concepts for comparison with those evaluated in the present study.

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 first year of the study focused on (1) development of reference designs for the mainline concepts and (2) a concept screening of alternate concepts to determine whether any alternate concepts should be evaluated in detail (see Ref. 1-1).

Table 1-4 lists the candidate blanket materials that generally served as a focus for the present study. The liquid metals, Li and LiPb alloy, were evaluated both as separate breeder materials and as combination breeder-coolants. Although both Li_2O and Li_8ZrO_6 were originally thought to provide adequate tritium breeding without a neutron multiplier, early in the study it was determined that Li_8ZrO_6 would not provide adequate tritium in a practical system. The $LiAlO_2$ was generally considered as representative of the stable ternary lithium oxides. The molten salt breeder, FLIBE, was one of the original alternate concepts that was considered in detail. Of the five coolants considered in detail, only the nitrate salt came from the alternate concept evaluation.

TABLE 1-4. CANDIDATE FIRST-WALL/BLANKET MATERIALS

Breeding Materials	Coolants	Structure	Neutron Multiplier
Liquid Metals Li 17Li-83Pb	H_2O Li 17Li-83Pb He Salt[c]	Austenitic Steel PCA Mn Steel[a]	Be Pb
Ceramics Li_2O Li_8ZrO_6 $LiAlO_2$[b]		Ferritic Steel HT-9 Mod. Ferr. St.[a]	
Salt FLIBE[d]		Vanadium Alloy V15Cr5Ti	

[a]Low-activation structural alloys. V15Cr5Ti is inherently low activation.

[b]$LiAlO_2$ is representative of ceramics that includes Li_2SiO_3, Li_2ZrO_3, etc.

[c]Nitrate salt.

[d]Fluoride salt.

Three classes of structural alloys were selected for evaluation in the study. The reference alloys include: (1) austenitic stainless steel represented by the primary candidate alloy (PCA) in the OFE Alloy Development Program, (2) high chromium (9-12%) ferritic steel represented by the commercial alloy HT-9, and (3) advanced vanadium-base alloys represented by the experimental V-15Cr-5Ti alloy. In addition, low activation variants of the austenitic (Mn-stabilized) and ferritic steels (no Mo or Ni) were considered. The reference vanadium alloy composition falls within the low activation definition.

Beryllium and lead were the only neutron multiplier options considered for enhancing the tritium breeding ratio of the candidate breeder materials. Since lead was eliminated from consideration early in the study because of perceived design difficulties and limited performance, beryllium was the only neutron multiplier evaluated in detail.

A major part of the effort was devoted to compiling the materials data base and development of analysis capability required for blanket concept design and evaluation. In many cases it was necessary to extrapolate available data to proposed conditions. An important part of the study was to evaluate the sensitivity of blanket performance to uncertainties in the data base. Blanket designs were developed in sufficient detail to evaluate the performance and safety characteristics of each concept. A consistent methodology was applied to all of the leading concepts in order to normalize the evaluations. For example, the same set of materials properties, stress limits, and corrosion limits, were used for all concepts. Also, self-consistent analyses were used to obtain 3-D tritium breeding ratios and shielding requirements for all designs.

Based on the blanket designs developed and the analyses conducted, critical feasibility issues for each concept were identified. A relative ranking of the leading blanket concepts was developed from the evaluation criteria for each of the four areas, viz., engineering feasibility, economics, safety, and R&D. The R&D requirements for the leading concepts were identified and prioritized.

This report concentrates on the design and evaluation of the leading concepts. Much of the detailed results used to arrive at the leading concepts was presented in the Interim BCSS Report[1-1] and are, therefore, not reproduced here. Chapter 2 gives an Overview and Summary of the results of this study. Chapter 3 presents the results of the four evaluations. The R&D Assessment is presented in Chapter 4 and details of the Evaluation Methodology are covered in Chapter 5. Results of the Special Issues are presented in Chapter 6. Chapters 7-10 provide details of the Blanket Designs.

REFERENCES FOR CHAPTER 1

1-1 M. A. Abdou et al., "Blanket Comparison and Selection Study Interim Report," ANL/FPP-83-1, October 1983.

BLANKET COMPARISON AND SELECTION STUDY

CHAPTER 2 - OVERVIEW AND SUMMARY

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2. OVERVIEW AND SUMMARY

2.1 Introduction

The Blanket Comparison and Selection Study (BCSS) was a two-year, multi-laboratory project initiated by the U.S. Department of Energy/Office of Fusion Energy (DOE/OFE) with the primary objectives of: (1) defining a limited number of blanket concepts that should provide the focus of the blanket R&D program, and (2) identifying and prioritizing critical issues for the leading blanket concepts. The BCSS focused on the mainline approach for fusion reactor development, viz., the D-T-Li fuel cycle, tokamak and tandem mirror reactors for electrical energy production, and a reactor parameter space that is generally considered achievable with modest extrapolations from the current data base. The STARFIRE[2-1] and MARS[2-2] reactor and plant designs, with nominal first wall neutron load of 5 MW/m^2, were used as reference designs for the study.

The BCSS was led by Argonne National Laboratory with major support from industry, universities and other national laboratories (see Table 2-1). 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 Argonne 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.

The BCSS was conducted with the following two-phase approach:

PHASE I

- Develop reference design guidelines.
- Develop evaluation critiera and methodology for ranking concepts.
- Identify concepts for considerations.
  - mainline concepts
  - alternate concepts
- Compile data base and develop uniform systems analysis.
- Develop conceptual designs for evaluation purposes.

PHASE II

- Identify critical issues for each concept.
- Evaluate blanket concepts.
- Identify and prioritize R&D requirements for leading concepts.

TABLE 2-1
BLANKET COMPARISON AND SELECTION STUDY TEAM

LEAD LABORATORY
Argonne National Laboratory
MAJOR SUPPORT ORGANIZATIONS
McDonnell Douglas Astronautics Company
GA Technologies, Inc.
TRW, Inc.
EG&G Idaho, Inc.
Lawrence Livermore National Laboratory[a]
University of California, Los Angeles[a]
SPECIAL CONTRIBUTORS
Grumman Aerospace Corporation[b]
Energy Technology Engineering Center[b]
Westinghouse Electric Corporation[b]
University of Wisconsin[b]
Rensselaer Polytechnic Institute[b]
Hanford Engineering Development Laboratory[b]
Oak Ridge National Laboratory

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

The BCSS Project was organized into three major tasks, viz., Project Tasks, Special Issues, and Blanket Design, as indicated in Fig. 2-1. The Project Tasks included such areas as design guidelines; evaluation critiera; and engineering, economics, safety and R&D evaluations. Special Issues Tasks provided a consistent 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 tandem mirror 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, viz., breeder, coolant, structure, and neutron multiplier if required, and by the geometric characteristics of the design. Table 2-2 lists the candidate materials that served as a focus for the BCSS.

TABLE 2-2
CANDIDATE FIRST-WALL/BLANKET MATERIALS

Breeding Materials	Coolants	Structure	Neutron Multiplier
Liquid Metals	H_2O	Austenitic Steel	Be
Li	Li	PCA	Pb
17Li-83Pb	17Li-83Pb	Mn Steel[d]	
	He		
	Salt[c]		
Ceramics		Ferritic Steel	
Li_2O		HT-9	
Li_8ZrO_6		Mod. Ferr. St.[a]	
$LiAlO_2$[b]			
Salt		Vanadium Alloy	
FLIBE[d]		V15Cr5Ti	

[a]Low-activation structural alloys. V15Cr5Ti is inherently low activation.

[b]$LiAlO_2$ is representative of ceramics that includes Li_2SiO_3, Li_2ZrO_3, etc.

[c]Nitrate salt.

[d]Fluoride salt.

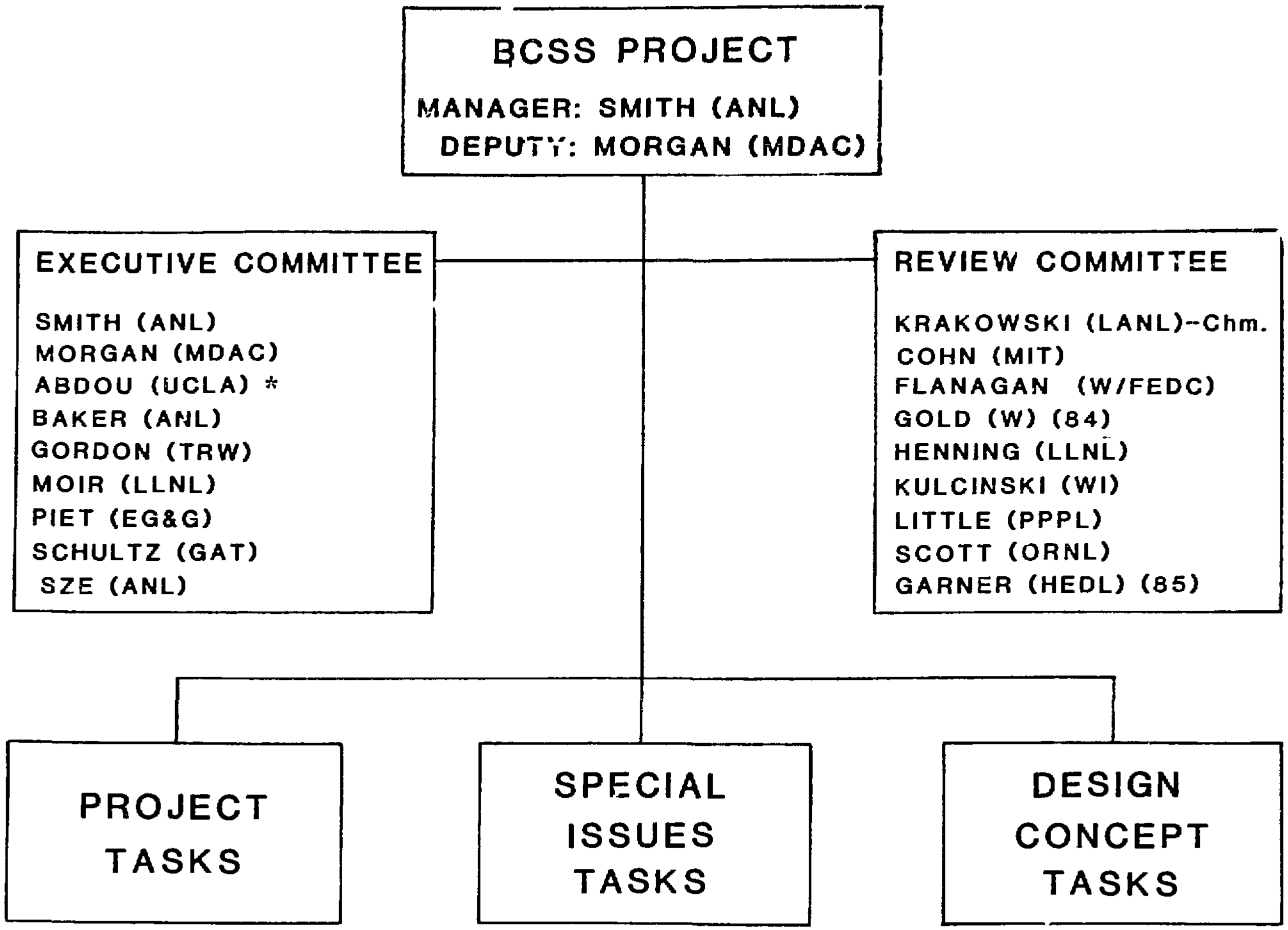


Figure 2-1. Blanket Comparison and Selection Study Project Organization.

*Project Manager FY 1983.

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 I effort focused on (1) development of reference designs for the mainline concepts and (2) a concept screening of alternate concepts to determine whether any alternate concepts should be evaluated in detail.[2-3] The mainline concepts considered included the following breeder/coolant combinations with each of the three candida 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 development of the analytical capability required for blanket concept development and evaluation. From the various combinations of 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 I of the study were evaluated in detail in Phase II of the study. A detailed evaluation methodology was developed in each of the four areas:

- Engineering Feasibility
- Economics
- Safety
- 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.

2.2 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.

2.2.1 Design Guidelines

The purposes of the design guidelines were:

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

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

Table 2-3 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 TF coils are discussed in Section 2.3.

TABLE 2-3
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/y	1	0.1
Dose to TF Coils, rads	10^{10}	10^{10}

2.2.2 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:

R=1: Potentially attractive, recommended for further development.

R=2: Set aside for possible future consideration.

- These concepts judged to be potentially acceptable but less attractive than the R=1 concepts.

R=3: Rejected.

- Those concepts that did not meet initial screening criteria.
- 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 2-4 were finally rated R=1 and evaluated in detail.

TABLE 2-4

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	

(FS: Ferritic Steel, NS: Nitrate Salt)

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

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

- Engineering
- Economics
- Safety
- R&D Requirements

The evaluation methodologies and results are summarized in Section 2.5. Details of the procedures and rankings are presented in Chapter 3 and 5. The overall ranking of concepts was based primarily on the engineering, economics, and safety evaluations.

2.3 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, and (6) auxiliary components (limiter/divertor, energy conversion system, etc.).[1]

2.3.1 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, 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 austenitic and ferritic steels 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 20-25% cold worked 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-15Mn-15Cr-0.05Cr-0.01N 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 ferritic (martensitic) steels, e.g., HT-9 and Fe-9Cr-1Mo, 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 Li-Pb alloy. The HT-9 (Fe-12Cr1MoVW) 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 has not been optimized for radiation damage resistance as in the case of the PCA alloy. Welding and radiation embrittlement are primary concerns.

The low activation ferritic steel proposed for evaluation is Fe-11Cr-2.5W-0.3V-0.15C. 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 higher temperature operation better corrosion resistance in lithium (and probably Li-Pb), and possibly better radiation damage resistance. The V-15Cr-5Ti 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 will require 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.

Table 2-5 provides a summary of predicted performance characteristics and limitations of the candidate structural alloys. Key conclusions from the study are summarized in Table 2-6.

2.3.2 Corrosion/Compatibility

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

- Liquid Metal Corrosion/Compatibility with Li and 17Li-83Pb
- Molten Salt (Nitrate Salt and FLIBE) Corrosion/Compatibility
- Water (200-350°C) Corrosion of Vanadium Alloys and CW-PCA
- Gaseous Corrosion/Compatibility of Vanadium

2.3.2.1 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. Section 6.2 provides an assessment of the corrosion behavior of austenitic PCA, ferritic HT-9, and vanadium V-15Cr-5Ti alloys in 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 including 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 Li-Pb to define the importance of each of these factors. However, the data for austenitic and ferritic steels are sufficient to provide reasonable projections of corrosion rates for Li and LiPb under anticipated conditions. Only limited data exist for corrosion of

TABLE 2-5
STRUCTURAL MATERIALS ASSESSMENT

Candidate Alloys	Austenitic Steel PCA-CW	Ferritic Steel HT-9	Vanadium V-15Cr-5Ti
Physical Properties			
Melting Temp. (°C)	1400	1420	1880
Nuclear Properties[a]			
dpa/MW • Y/m^2	11	11	11
appm He/MW • Y/m^2	174	130	57
appm H/MW • Y/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
Max. Surf. 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 h, 100 dpa)			
500 °C	100	155	165
550 °C	85	100	165
700 °C	--	--	165
Maximum Allowable Temperature, °C (~.5 T_m) (Irrad. Embrit.)	550	550	720
Corrosion Rate, mg/m^2 • h[c]			
Lithium (500°C)	60	2	<.01
LiPb (500°C)	>100	100	.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 Temp. Limit	• Weld Procedure (PWHT) • DBTT above RT • Operating Temp. Limit • Liquid Metal Embrit. • Ferromagnetic Properties	• R&D Requirements • Weld Procedure (Inert environment) • Oxidation Characteristic • High T Permeation Rates • Costs

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

TABLE 2-6
KEY CONCLUSIONS OF BCSS

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

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 Li 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/y. The corrosion limit to avoid problems from excessive deposition of corrosion product in localized regions is generally believed to be ~5 μm/y. Because of the specific design dependency and uncertainties associated with this limit, a more liberal limit of 20 μm/y 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 >3mm during a service life of 2 to 4 years. In most cases the most important consideration in establishing the operating temperature limits for fusion reactor blankets is mass transfer and deposition.

Table 2-7 lists the proposed design temperature limits based on mass transfer/deposition and radioactive mass transfer for the three structural materials in flowing lithium and Li-Pb. The corrosion rates for PCA in LiPb are clearly excessive for acceptable thermal hydraulic performance. The rates for PCA in lithium and ferritic steel in LiPb pose severe constraints that would generally make such systems unattractive. The corrosion rates for ferritic steel in lithium meet the mass transfer/deposition criteria; however, radioactive mass transfer will be sufficient to require remote maintenance. The predicted corrosion rates for V-15Cr-5Ti 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.

2.3.2.2 Molten Salt/Corrosion

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

TABLE 2-7
DESIGN TEMPERATURE LIMITS (°C) FOR LIQUID METAL SYSTEMS AT 1.5 m/s

Liquid-Metal	Criteria[a] μm/y	Austenitic Steel PCA	Ferritic Steel HT-9	Vanadium Alloy VCrTi
Lithium	20	470	580	>750
	5	430	550	>750
	0.5	370	460	>750
LiPb	20	410	450	>750
	5	375	415	>750
	0.5	320	360	650

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

The salts are somewhat conducting and, when moved through a magnetic field, will generate a voltage which can cause dissociation of the salt. Increasing the ionic content of the salt will increase 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 have been obtained with austenitic stainless steels. Based on the Molten Salt Reactor Experiment, corrosion rates of Type 316 stainless steel loops containing $2LiF-BeF_2$ mixture under heat transfer conditions average about 8 μm/year 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 ferritic steels 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.

2.3.2.3 Water Corrosion

Most earlier studies have concluded that vanadium alloys could not be used in pressurized water-cooled systems because of excessive corrosion. However, evaluation of recent scoping data 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 cold-work 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 cold-worked PCA in pressurized water systems.

2.3.2.4 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 will be 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.

2.3.3 Breeder Materials

Lithium is the only viable tritium breeding material for a 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.

2.3.3.1 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 will 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:

- fabrication/refabrication of the ceramic,
- property data base,
- tritium release from solid,
  - temperature limits
  - specie
- radiation effects (swelling),
- tritium breeding.

Important aspects of the first four issues are discussed in Section 6.3 and briefly here. The tritium breeding considerations are discussed in Sections 6.8 and 6.10.

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% TD) spheres to achieve about 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.

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 effects 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 2-8 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, viz., elastic moduli, fracture strength and creep properties, are non-existent 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 to the mechanical response of solid breeder materials remain.

TABLE 2-8

PROPERTIES OF CANDIDATE SOLID BREEDER MATERIALS[a]

Breeder	MP, °C	ρ ^{6}Li, g/cm^3	k,[b] W/m-K	T_{min},[d] °C	T_{max},[f] °C	ΔT, °C	Grain[h] Dia., μm	Tritium[b] Diffusivity cm^2/s
Li_2O	1433	0.93	2.5[b] 1.27[c]	410[e]	800[g]	390	3.0[i]	10^{-7}
$LiAlO_2$	1610	0.28	1.6[b] 1.1[c]	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	--	

[a]Estimates based on limited unirradiated and irradiated data for candidate solid breeders and other ceramic materials.

[b]Estimated for 85%-dense, sintered material at 1000°K.

[c]Estimated for 87% dense sphere-pac material.

[d]Values are estimated based on diffusive inventory considerations.

[e]Based on solubility consideration.

[f]Based on sintering at 0.66 T_m.

[g]Based on high-temperature mass transfer (LiOT/LiOH) considerations.

[h]Based on the smallest grain diameters with existing fabrication technology.

[i]Grain growth has been observed after irradiation of Li_2O.

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 flowrate 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 2-8.

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 has been observed after irradiation of Li_2O at temperatures of 500-700°C. The $LiAlO_2$ is much more resistant to swelling and grain growth. However, significant retention of tritium was observed after capsule irradiations.

2.3.3.2 Liquid Breeder Materials

Three liquid breeder materials, viz., lithium, 17Li-83Pb eutectic alloy, and FLIBE (LiF-BeF_2), have been considered for the liquid breeder blankets. Table 2-9 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 Li-Pb 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 which results in low inventories and relatively high tritium pressures.

TABLE 2-9
PROPERTIES OF LIQUID BREEDER MATERIALS[a]

	Li	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

[a]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.

2.3.4 Special Materials

Neutron multipliers, electrical insulators, and nitrate salts (NS) were evaluated for special applications in the BCSS.

2.3.4.1 Beryllium

Based on the Phase I 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 GWe-y, respectively) of fusion reactor service. Recycle of beryllium will be required and close attention to beryllium recycle losses will be important. Since beryllium will become radioactive in the fusion environment (due to impurities), a remote fabrication technology will be 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 Be 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 intragrannular helium bubble swelling will 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 Be 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.

2.3.4.2 Electrical Insulators

Electrical insulators are important to liquid metal blankets because they can significantly reduce the magnetohydrodynamic 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 is a major concern. 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 volt) 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.

2.3.4.3 Nitrate Salts

Nitrate salts ($NaNO_3$-KNO_3 and $NaNO_3$-$NaNO_2$-KNO_3) have been used for many years in a non-nuclear environment. Some thermophysical properties of the reference nitrate salt (50% $NaNO_3$-50% KNO_3), also called draw salt, are listed in Table 2-10. Primary concerns related to the use of nitrate salt 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 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 Na, K and N.

TABLE 2-10
SELECTED PROPERTIES OF NITRATE SALT
(50% $NaNO_3$ - 50% KNO_3)[a]

Melting Temperature (°C)	220
Density (kg/m^3)	1840
Thermal Conductivity (W/m°K)	0.52
Viscosity (mPa • s)	1.8
Heat Capacity (J/kg°K)	1605
Electrical Conductivity (1/ohm • cm)	1.04

[a]Properties except melting temperature are at 400°C.

2.3.5 Tritium Containment

The BCSS has concentrated on the issues of tritium containment in a D-T fusion reactor blanket and coolant system. 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. He-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 curie/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.

2.3.6 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 for candidate blanket concepts. The required tritium breeding ratio must exceed unity by a margin, G, to supply inventory for startup of other fusion reactors, to compensate for losses and radioactive decay between production and use, and to compensate for hold-up inventories in various components as well as reserve storage inventory. This margin, G_o, is found to strongly depend on the desired doubling time and many of the reactor plasma and engineering parameters, e.g., 1) the tritium fractional burnup in the plasma, 2) the equilibrium tritium inventories in various components, particularly the blanket, 3) the time constants to reach equilibrium tritium inventories, 4) the frequency of failure and time to repair components in the tritium processing system, and 5) efficiencies of and non-radioactive (e.g., chemical) losses from various subsystems.

In comparing blanket concepts, as well as plasma and technology choices, as to the potential for attaining DT fuel self sufficiency, one needs a figure of merit. One such figure of merit, F, which has been used in the BCSS final comparative evaluation is

$$F = \frac{T_c - (1 + G_o)}{\sqrt{\Delta_G^2 + \Delta_s^2 + \Delta_p^2}} \qquad [2\text{-}1]$$

where

T_c = Calculated tritium breeding ratio for a reference reactor system.

G_o = 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 breeding ratio in the reference system due to uncertainties in nuclear data, calculational methods, and geometrical representation.

Tables 2-11 and 2-12 show the results for T_c, $1+G_o$, Δ_G^2, Δ_s^2 and Δ_p^2 for tokamaks and mirrors, respectively.

The general conclusions are as follows. The G_o is relatively insensitive to blanket concept with a value of ~0.07. Of the three uncertainty factors, viz., 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 largest uncertainty is associated with Δ_G^2. This term is affected most by the following factors:

- tritium fractional burnup in the plasma,
- required doubling time,
- tritium processing efficiency.

2.3.7 3-D 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 study. These designs include the combinations of breeder, coolant and structural materials presented in Tables 2-11 and 2-12. All the ternary-ceramic 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 tritium breeding ratios (TBR's).

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 a with water coolant and beryllium coating. An alternate limiter design that is used for the two liquid-lithium blanket concepts, Li/Li and Li/

TABLE 2-11
RESULTS OF TRITIUM BREEDING REQUIREMENTS, POTENTIAL AND UNCERTAINTIES
FOR CANDIDATE BLANKET CONCEPTS IN TOKAMAKS

Concept	T_c	$1+G_o$	Δ_G^2	Δ_S^2	Δ_p^2	$\Sigma\Delta_i^2$	$T_c-(1+G_o)-\sqrt{\Sigma\Delta_i^2}$
A $LiAlO_2$/NS/FS/Be	1.24	1.073	.05	.0094	.0009	.0603	-.079
B Li/Li/FS	--	--	--	--	--	--	--
C LiPb/LiPb/V	--	--	--	--	--	--	--
D Li/Li/V	1.28	1.068	.05	.0094	.0041	.0635	-.040
E Li_2O/He/FS	1.11	1.067	.05	.0094	.0029	.0623	-.207
F $LiAlO_2$/He/FS/Be	1.04	1.067	.05	.0094	.0009	.0603	-.273
G Li/He/FS	1.16	1.068	.05	.0094	.0030	.0624	-.158
H FLIBE/He/FS/Be	1.17	1.067	.05	.0094	.0017	.0611	-.144
I $LiAlO_2/H_2O$/FS/Be	1.16	1.071	.05	.0094	.0009	.0603	-.157

TABLE 2-12
RESULTS OF TRITIUM BREEDING REQUIREMENTS, POTENTIAL AND UNCERTAINTIES
FOR CANDIDATE BLANKET CONCEPTS IN MIRRORS

Concept	T_c	$1+G_o$	Δ_G^2	Δ_S^2	Δ_p^2	$\Sigma\Delta_i^2$	$T_c-(1+G_o)-\sqrt{\Sigma\Delta_i^2}$
A $LiAlO_2$/DS/FS/Be	1.29	1.069	.05	.0094	.0009	.0603	-.025
B Li/Li/FS	1.14	1.068	.05	.0094	.0035	.0629	-.179
C LiPb/LiPb/V	1.18	1.067	.05	.0094	.0024	.0618	-.136
D Li/Li/V	1.19	1.068	.05	.0094	.0041	.0635	-.130
E Li_2O/He/FS	1.14	1.067	.05	.0094	.0029	.0623	-.176
F $LiAlO_2$/He/FS/Be	1.16	1.067	.05	.0094	.0009	.0603	-.152
G Li/He/FS	1.17	1.067	.05	.0094	.0030	.0624	-.147
H FLIBE/He/FS/Be	1.29	1.067	.05	.0094	.0017	.0611	-.024
I $LiAlO_2/H_2O$/FS/Be	1.22	1.070	.05	.0094	.0009	.0603	-.096

He, employs a lithium-cooled V15Cr5Ti heat sink along with a beryllium coating. The geometrical configuration of the RF-waveguides which penetrate perpendicularly through the lower outboard sector is modeled after the STARFIRE design, i.e., HT9 grid structure cooled by water.

In order to account for the DT fusion taking place in the end-plug regions of TMRs, the TBR's calculated by MCNP for the TMR blankets have been reduced by 2.5%. In addition, the blanket area lost for start-up heating has been estimated to be ~0.5%. Thus the overall breeding adjustment required for the TMR designs is -3% of the MCNP estimates.

2.3.8 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: a) workers are permitted in the reactor hall one day after shutdown, and b) superconducting coils are required to function for 150 MW • y/m^2 DT neutron exposure at the first wall. The occupational exposure is limited to 0.5 mrem/h based on working 8 h per day and 40 h per 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 mirror reactors. A shielding criterion of 10^{10} rads was used to size the bulk shield in the inboard section of the tokamak reactors and the central cell sections under the coils for mirror reactors. 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 about 0.1 mW/cm^3 which is very close to the optimum design conditions for mirror reactors and quite satisfactory for the design of the toroidal field 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 Fe1422 steel as a bulk shielding material, B_4C as a neutron absorber, H_2O as a moderator and coolant, and Pb as a gamma-ray absorber.

All calculations were performed with the discrete ordinate code ANISN 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 CTR 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 2-13 gives the shield thickness and the blanket energy multiplication factors for each concept based on the above criteria.

TABLE 2-13
BCSS BLANKET/SHIELD DIMENSIONS, ENERGY MULTIPLICATION FACTORS AND ATOMIC DISPLACEMENT IN THE FIRST WALL FOR THE TOKAMAK BLANKET CONCEPTS AND THE MIRROR BLANKET CONCEPTS

Blanket Concept Breeder/Coolant/Structure/Multiplier	Blanket/Shield/Total Thickness, cm Inboard	 Outboard	Energy Multiplication Factor Blanket
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	80/80/161	1.259
Li/Li/FS	80/48/129	80/80/160	1.313
LiPb/LiPb/V	90/40/130	90/75/165	1.294

2.3.9 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. 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-15Cr-5Ti. The coolant/breeder/multiplier materials were $LiAlO_2$, 17Li-83Pb, Be, Li_2O, Lithium, Nitrate Salts, 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 are:

- All materials will require remote handling and shielding during operations and in the first ten years after removal from a reactor.

- At 100 years after removal from a reactor, only Li_2O, lithium, and FLIBE can be handled by workers without special protection.

- Near-surface disposal can be used for: V-15Cr-5Ti, Mod HT-9, TENELON, Beryllium, Li_2O, Lithium, and FLIBE.

- Special processing will be required before near-surface disposal can be used for: PCA, HT-9, $LiAlO_2$, 17Li-83Pb, and Nitrate Salt.

- 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.

2.3.10 Electromagnetic Effects

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

2.4 Design Concepts

The blanket concepts grouped by coolant type--liquid metal, helium, pressurized water, and nitrate salt are summarized below.

2.4.1 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 (LiPb) 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 will limit 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. Non-water-cooled invessel 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.

2.4.1.1 Final Rankings for Self-Cooled Liquid Metal Concepts

A summary of final rankings for the various liquid blanket concepts is given in Tables 2-14A and 2-14B for tokamak and TMR reactors. In general, the blanket designs of a tandem mirror reactor 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 tandem mirror reactor compared to that of a tokamak reactor. Liquid lithium, owing to its superior thermo-physical properties, is a better coolant than LiPb . From an engineering design point of view, the vanadium alloy is a better structural material than either ferritic steel or PCA since the vanadium alloy has both a higher allowable structural temperature and a higher allowable coolant to structure interface temperature.

2.4.1.2 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 tandem mirror reactor (Table 2-3) the MHD, heat transfer, and structural material requirements for a tokamak blanket are much more stringent than for a tandem mirror 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.2-2. 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 increasing significantly the total pressure drop through the blanket . 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 tandem mirror reactor is similar to that of the MARS design.[2-2] 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.

TABLE 2-14A
RANKING OF TOKAMAK AND TMR BLANKET CONCEPTS[a]
LIQUID METAL AND MOLTEN SALT CONCEPTS

Concept	PCA	Ferritic	Vanadium
A. Outboard Blanket Same as Inboard			
Li/Li	2B/2A	2A/1	2A/1
LiPb/LiPb	3	2B	1/1
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. LM Outboard Blanket Different Inboard Blanket[b]			
Li/Li: -/He	2A	2A	1B
LiPb/LiPb: -/He	2B	2A	2A
LiPb/LiPb: -/H_2O	2B	2A	2A
C. Either A or B but 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

[a]Same ranking for tokamak and TMR except where two numbers are listed; where different, tokamak ranking is listed first.

[b]Concepts considered for tokamak only.

TABLE 2-14B
RANKING OF TOKAMAK AND TMR BLANKET CONCEPTS
SOLID BREEDER CONCEPTS

Concept	PCA	Ferritic	Vanadium
Li_2O/H_2O	2B	2B	2B
Li_2O/He	2A	1	3
Li_2O/HTS	2A	2A	3
Li_8ZrO_6/H_2O	2B	2B	2B
Li_8ZrO_6/He	2B	2B	3
Li_8ZrO_6/HTS	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
$Li_8ZrO_6/HTS/Be$	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 with neutron multiplier outboard, nonbreeding inboard.[b]			
$SB/He/BE: -/H_2O$	2A	2A	2B

[b] Concepts considered for tokamak only.

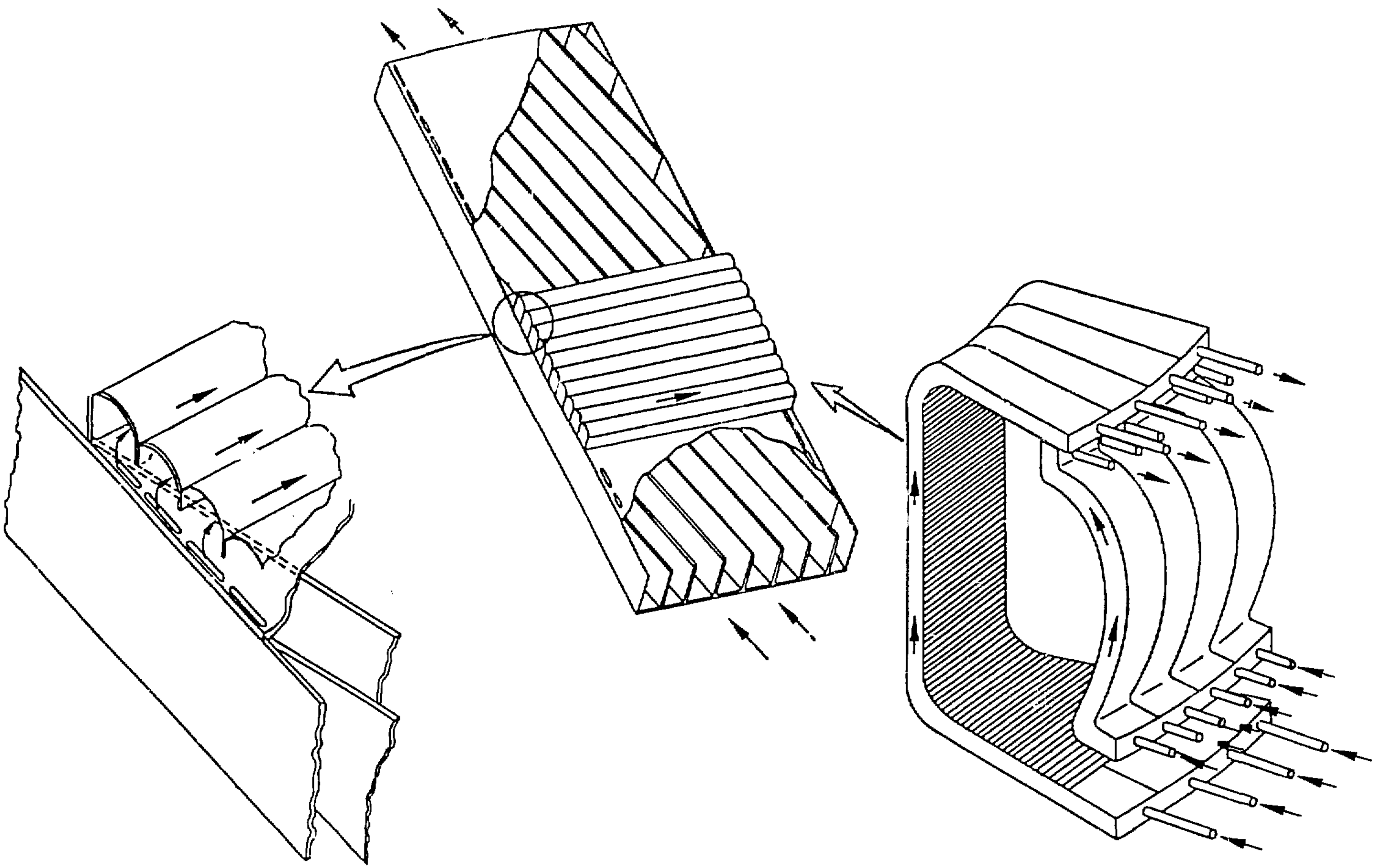


Figure 2-2. Schematic of the reference design for the self-cooled liquid-metal blanket (poloidal/toroidal flow) of a tokamak reactor.

However, adequate cooling of the first wall is still achieved with moderate pressure drops since the surface heat flux in the TMR first wall is relatively small.

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

- Liquid Metal MHD Contraints
  - (pressure drop and heat removal)
- Corrosion Limitations
- Reactivity of Lithium
- Tritium Recovery and Containment for LiPb
- High Mass and Cleanup of LiPb

2.4.2 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. This leads to the need to operate the helium pressure in the range of 40 to 80 atm. The pumping powers for the helium-cooled design 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 U.S.

2.4.2.1 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 2-14A and 2-14B. Rationale for the rankings is also given for all except R=1 concepts.

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 ferritic steel 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 tritium breeding ratios. Concepts with PCA structure generally rank lower than those with HT-9 ferritic steel 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 ferritic steel, which can restrict the allowable system ΔT. Concepts with Pb neutron multiplier were generally ranked lower than those with Be; the relatively high melting point of lead (327°C) sharply restrict 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.

2.4.2.2 Blanket Configuration for Reference Helium-Cooled Concepts

The lobular pressurized-module concept was selected for the reference design for all He-cooled concepts in the BCSS. All the helium-cooled blanket concepts appear to be equally applicable to the tandem mirror 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 tandem mirror 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.

Li_2O/He/FS Concept (Tokamak and TMR)

The Li_2O/He/FS concept is shown schematically in Fig. 2-3. 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 1 atm pressure. This allows the 50 atm 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 that need to be addressed for the Li_2O/He concepts are as follows:

- Irradiation-Induced Swelling of Li_2O
- Tritium Recovery and Containment
- LiOH Mass Transfer
- First Wall Cooling/Stress Limits
- Marginal Tritium Breeding Ratio Without Neutron Multiplier
- Helium Leakage into Plasma

$LiAlO_2$/He/FS/Be Concept (Tokamak and TMR)

The $LiAlO_2$/Be/ferritic steel blanket is very similar in configuration to the Li_2O design shown in Fig. 2-3. The rectangular fuel plate approach for 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 diameter cylindrical rods cooled by crossflowing helium. This configuration allows for easy manufacturing and assembly. Further, the rod arrangement provides accommodation of radiation-induced small dimension changes in the beryllium without allowing high stresses to develop. The tritium breeding ratio 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

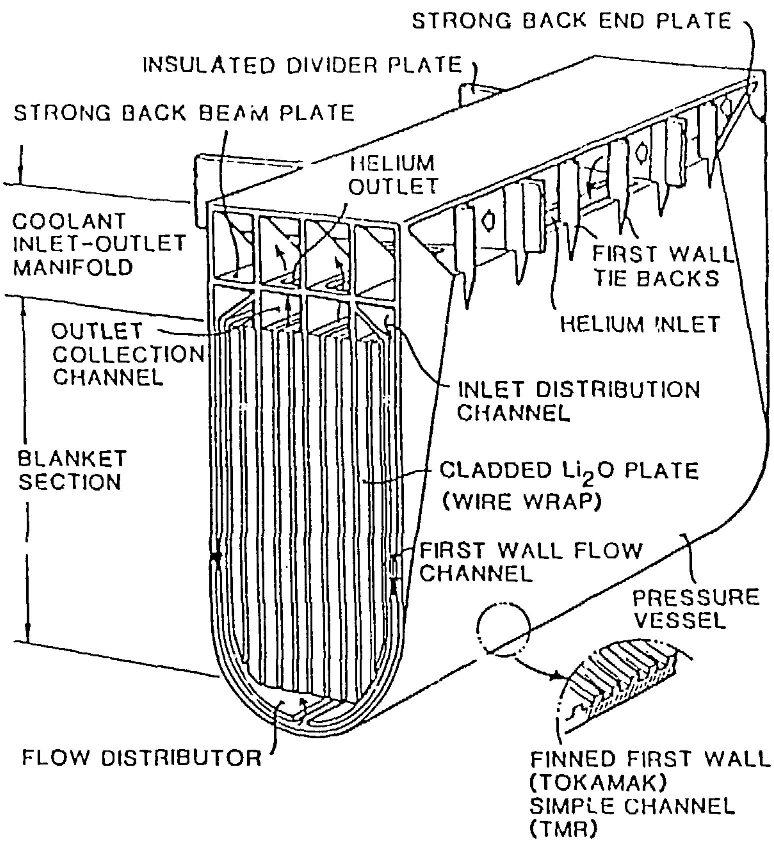


Figure 2-3. Li_2O/helium blanket design.

have been better to include about 10 cm of beryllium on the inboard blanket to achieve a higher tritium breeding ratio, 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 Be are:

- Tritium Recovery and Containment
- Temperature Control of Breeder
- Control of Tritium from Be
- First Wall Cooling/Stress Limits
- Irradiation Damage of Be
- Helium Leakage into Plasma

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. 2-4. 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 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:

- Corrosion Temperature Limitations
- Reactivity of Lithium
- First Wall Cooling/Stress Limits
- Tritium Containment/Recovery
- Coolant Leakage into Plasma

FLIBE/He/FS/Be Concept (Tokamak and TMR)

The breeder in this concept is Li_2BeF_4, the low-melting-temperature (363°C) version of the salt commonly known as FLIBE. The blanket reference design uses a lobe-shaped module essentially identical to the other He-cooled designs to contain the 50-atmosphere helium gas. Helium cools the

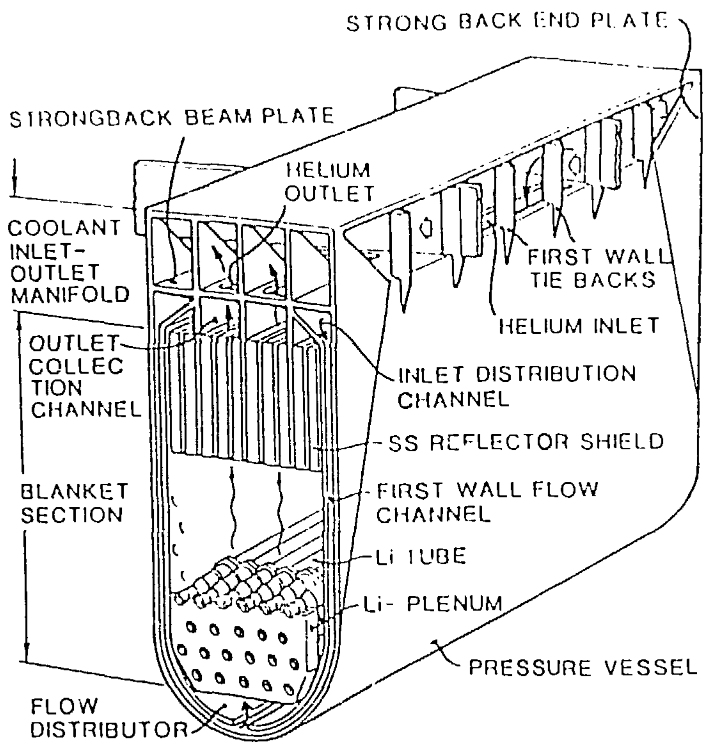


Figure 2-4. Liquid lithium/helium blanket design.

first wall and blanket internals. The internals consist of a bed of beryllium balls, nominally 1 cm 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/sec) in ferritic steel 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 will be 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 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 years) 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:

- Tritium Containment
- First Wall Cooling/Stress Limits
- Be Resources and Recycling Losses
- Be Pellet Radiation Damage
- Cleanup of FLIBE in Event of Leak

2.4.3 Water-Cooled Concepts

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

2.4.3.1 Final Rankings for Water-Cooled Concepts

The rankings for all water-cooled blanket concepts considered in the BCSS are presented in Tables 2-14A and 2-14B. 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. $LiAlO_2$ 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 Pb as a neutron multiplier were also ranked R=2B. The use of molten Pb 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 a net TBR 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.

2.4.3.2 $LiAlO_2/H_2O$/FS/Be Concept Reference Design (Tokamak and TMR)

The blanket configuration is modular in nature, with a lobe-shaped semi-cylindrical actively cooled first wall. Nominal dimensions are 30 cm width poloidally (15 cm radius for the first wall) and 70 cm depth measured radially away from the plasma. The first 20 cm of the breeding zone is a 90:10 volume mixture ratio of beryllium (Be) and the $LiAlO_2$ ternary ceramic (TC) breeder. Both materials are fabricated in sphere-pac form; the individual Be and TC spheres are ~100% dense. 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 which 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. 2-5.

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 6-atm 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 first wall is 28 cm, and 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:

- Reliability of Coolant Tubes
- Tritium Recovery/Containment
- Safety Related to Pressurized H_2O
- Fabrication/Refabrication of Breeder
- Limited Power Variation Capability
- Beryllium Reprocessing

2.4.4 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 tempera-

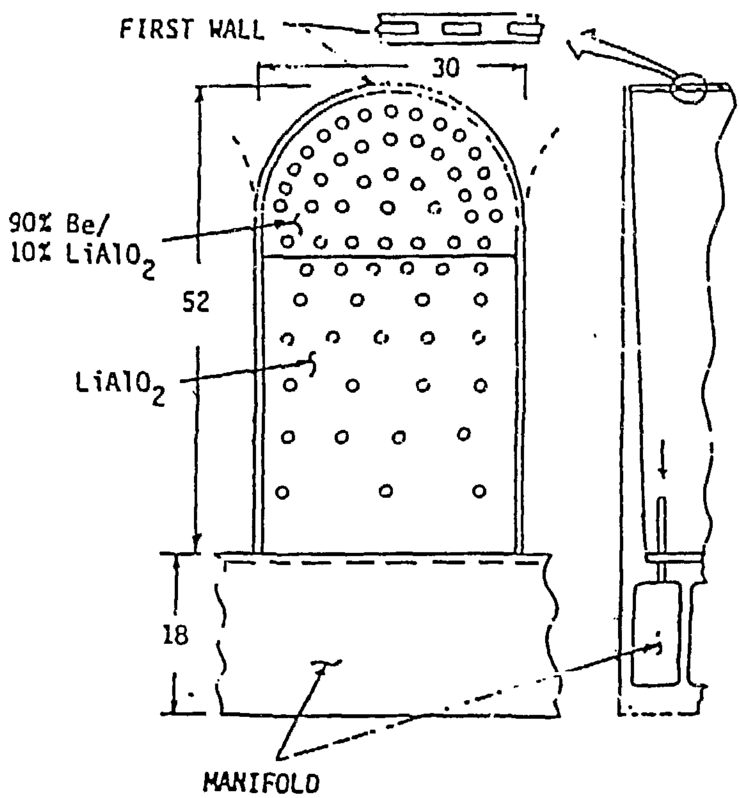


Figure 2-5. Reference design configuration for $LiAlO_2$/H_2O/FS/Be concept - tokamak.

ture and a high heat transfer coefficient. These characteristics are best met by molten salt coolants. The family of nitrate salts (NS) and nitrate/nitrite salts were specifically considered. The many desirable features of molten salt coolants are mitigated 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.

2.4.4.1 Final Rankings for NS-Cooled Concepts

The nitrate-salt-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 nitrate salt-cooled concepts are given in Table 2-14A and 2-14B.

2.4.4.2 $LiAlO_2$/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. 2-5).

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 approximately 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 sidewalls 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 which 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 semi-eliptical toroidal end caps. Coolant tubes are routed axially; the 6.32 m TMR module length 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 nitrate salt cooled concept include:

- Salt Stability
- Activation Product Control
- Tritium Recovery/Containment
- Voltage Enhanced Corrosion
- Coolant Compatibility with Beryllium

2.5 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 (only seven for the tokamak (See Table 2-4) in the four areas:

- engineering feasibility
- economics
- safety
- 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.

2.5.1 Engineering Feasibility

The items included under "Engineering Feasibility," listed in Table 2-15, include important blanket criteria that either deserve separate consideration or do not readily fit under the categories of safety and economics.

TABLE 2-15
ENGINEERING EVALUATION INDICES

		Weighting Values
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. Startup/Shutdown Requirements		10

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

2.5.1.1 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 2-15:

$$EFM = \sum_i I_i W_i \qquad [2-2]$$

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 tandem mirror reactors.

Each of the seven indices is briefly described below.

Tritium Breeding and Inventory (I_1) - This is considered a major feasibility issue and given a weighting of 25 points. A figure of merit (equal to I_1) was calculated for each concept based on its 3-D tritium breeding ratio (TBR) and steady-state tritium inventory. Uncertainties in estimating actual breeding requirements, in reactor design definition, and in calculation and modeling accuracies were considered.

Engineering Complexity and Fabrication (I_2) - Eight important features of blanket designs were identified which affect complexity and fabrication. Concepts were judged as to how well they rated in each area.

Maintenance and Repair (I_3) - Four general blanket features were identified that impact the reactor operator's ability to maintain the reactor.

Resources (I_4) - Concepts were rated on their consumption of scarce resources, based on a 1000 GWe fusion plant capacity over a 40-year span.

Power Variation (I_5) - The concept's capabilities to permit reactor operation below or above the nominal design points were measured.

Increased Capability (I_6) - This indice measured each concept's ability to (1) operate at higher neutron wall loads (P_{nw}), and (2) to accommodate higher first wall surface heat loads and/or higher particle fluxes; this is particularly important for tokamaks.

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.

2.5.1.2 Results and Conclusions of Engineering Evaluation

The scores for all engineering evaluation indices are compared in Table 2-16 for TMR and tokamak concepts.

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 indices, 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 He-cooled concepts, primarily due to lower scores in the complexity and resources categories which result from the need to add

TABLE 2-16
ENGINEERING EVALUATION - SUMMARY OF SCORES

REACTOR		CONCEPT	BREEDING & INVENTORY (25)	COMPLEXITY & FABRICATION (25)	MAINTENANCE (15)	RESOURCES (5)	POWER VARIATION (10)	P_{NW}, q'a, t_e INCREASE CAP. (10)	STARTUP/ SHUTDOWN (10)	TOTAL SCORE (OF 100)	(SCORE ÷ HIGHEST SCORE)
	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	.917
	C	LiPb/LiPb/V	10.4	21.6	6.0	5	10	7.5	5	65.5	.959
T	D	Li/Li/V	11.1	21.6	7.5	5	10	8	5	68.2	.999
M	E	Li_2O/He/FS	6.8	12.5	13.5	5	8.5	10	7	63.3	.927
R	F	$LiAlO_2$/He/FS/Be	8.8	10.6	13.5	2.5	8	10	7	60.4	.884
	G	Li/He/FS	5.9	16.7	10.5	5	10	10	6	64.1	.939
	H	FLIBE/He/FS/Be	20.3	15.3	9.8	1.5	6	10	5	67.9	.994
	I	$LiAlO_2/H_2O$/FS/Be	13.9	10.7	11.3	1.5	3	8	7	55.4	.811
	A	$LiAlO_2$/NS/FS/Be	15.4	13.7	9.8	1.5	10	6	5	61.4	.849
T	B	-	-	-	-	-	-	-	-	-	-
O	C	-	-	-	-	-	-	-	-	-	-
K	D	Li/Li/V	18.4	19.4	10.5	5	10	4	5	72.3	1.000
A	E	Li_2O/He/FS	4.0	12.5	13.5	5	8	2	7	52.0	.719
M	F	$LiAlO_2$/He/FS/Be	0	10.7	13.5	2.5	8	2.5	7	44.2	.611
A	G	Li/He/FS	4.8	16.7	10.5	5	9	2.2	6	54.2	.750
K	H	FLIBE/He/FS/Be	9.6	15.3	9.8	1.5	6	2.3	5	49.5	.685
	I	$LiAlO_2/H_2O$/FS/Be	8.3	10.7	11.3	1.5	3	7.5	7	49.3	.682

Be to the blanket. The FLIBE blanket scores considerably higher than the other He-cooled concepts primarily because of the higher breeding ratio 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 Be neutron multiplier.

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 Be neutron multiplier into the inboard blanket. This would have given it ~4-5 points for the breeding category. The FLIBE concept does not score as well as its TMR counter-part, 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 He-cooled and H_2O-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.

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 consider-

ably larger for tokamaks than for TMR's. 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 TMR's, 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 TMR's where its relatively good cooling capabilities do not give it an advantage, and 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.

2.5.2 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. Cost of Electricity 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 tandem mirror and tokamak reactor power plants. A system performance and economic computer code was written to compare the blanket options. The costs categories which may be affected by the blanket design parameters were the costs of the 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 which 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 the tandem mirror reactor. Although the costs are shown for both reactor types, many of the underlying technical performance and economic assumptions and groundrules 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 six year construction period.

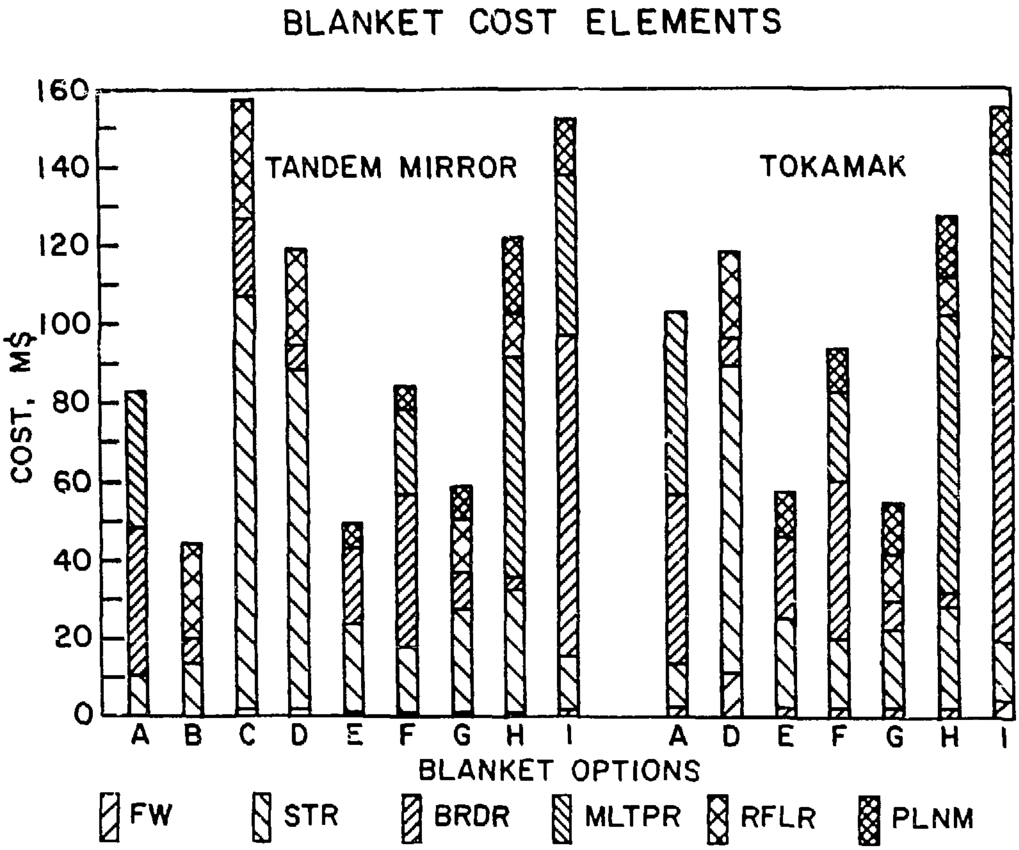
Reference Designs Economic Evaluation Results - The Cost of Electricity (COE) is composed of several factors, namely

$$\text{COE} \sim \frac{\text{Total Capital Cost} \times \text{Fixed Charge Rate} + \text{Annual Costs}}{(\text{Thermal Power} \times \text{Gross Efficiency} - \text{Recirculating Power}) \times \text{Availability}}.$$

The cost of the blanket components (first wall, blanket structure, breeder, multiplier, reflector and plenum) are shown in Fig. 2-6. 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 total cost of the blankets ranged from \$44M to \$157M.

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 \$80M to \$122M. The specific cost values can be found in Section 3.2 and Section 5.3.

The use of the reference reactors, STARFIRE and MARS, for the BCSS required that specific groundrules and methodology be adopted regarding how the fusion power scaled with respect to the blanket and shield thicknesses. Section 5.3 defines in detail those groundrules and methodologies. Briefly, the fusion power for the tandem mirror is held constant and the blanket and shield thickness variations then influence only the central cell magnet costs. In



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
I – $LiAlO_2$/H_2O/HT-9/Be

Figure 2-6.

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. It also should be noted that a portion of the tandem mirror 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 main heat transport system.

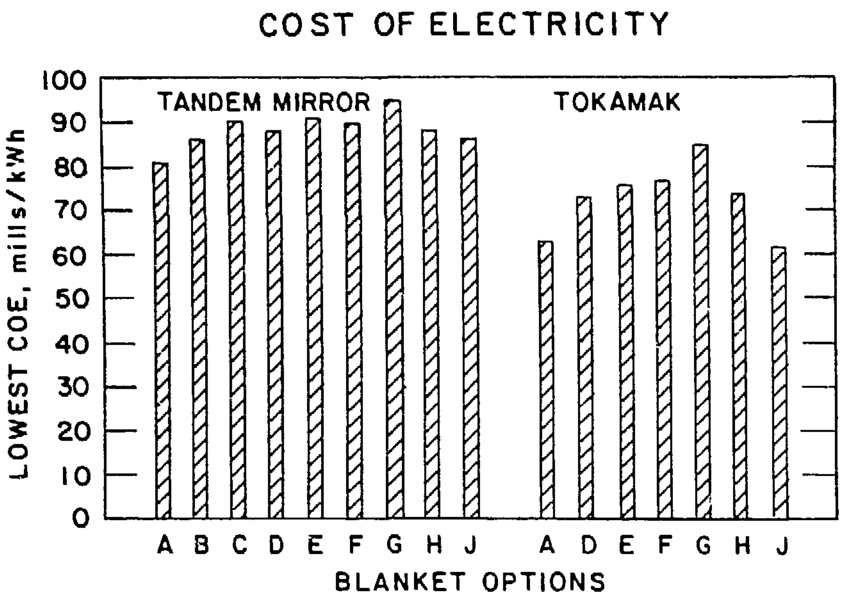
Although there are other costs affected by the blanket choice, the major costs are in the Reactor Plant Equipment, which include the first wall and blanket, shield, magnets, heat transport system, and fuel handling and storage. There are significant variations in the individual elements. However, the variations 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 nitrate salt and water-cooled designs are the lowest cost designs.

Figure 2-7 presents the COE values for the R=1 blanket concepts in both reactor types. These results can be summarized as follows:

- The helium-cooled lithium breeder is the least economically attractive concept in either reactor type.
- The nitrate salt-cooled design is a good candidate for either reactor type.
- The water-cooled design, because of its thinner inner blanket and shield design, is attractive for the tokamak.
- 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:

- Material cost for lithium, beryllium and vanadium.
- Blanket lifetime.
- Blanket energy multiplication factor (EMF).



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

Figure 2-7.

- Gross energy conversion efficiency.
- Blanket/shield thickness.
- Coolant pumping power.

The results of the analyses indicate that the latter four factors provide the greatest leverage for improved performance.

2.5.3 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, eleven specific evaluation indices have been established to compare blanket designs. Individual indices 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 2-17. 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:

$$SFM = \sum_{i=1}^{11} I_i W_i \qquad [2\text{-}3]$$

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

TABLE 2-17
SAFETY EVALUATION INDICES

Index	Index Name	Index Weight
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 Interaction	6
8	Fault Tolerance of the Reactor Building to Blanket Transients	6
9	Normal Radioactive Effluents	20
10	Occupational Exposure	10
11	Waste Management	10

The safety evaluation indices can be grouped into four major evaluation categories: accident source term characterization (30 percent of SFM), accident fault tolerance (30 percent), effluent control (20 percent), and maintenance and waste management (20 percent). The balance (60-40 percent) between accidents and non-accident 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. 2-8 and 2-9. 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 blankets designs. The top-ranked designs, LiPb/LiPb/V (TMR only), Li_2O/He/HT9, 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.

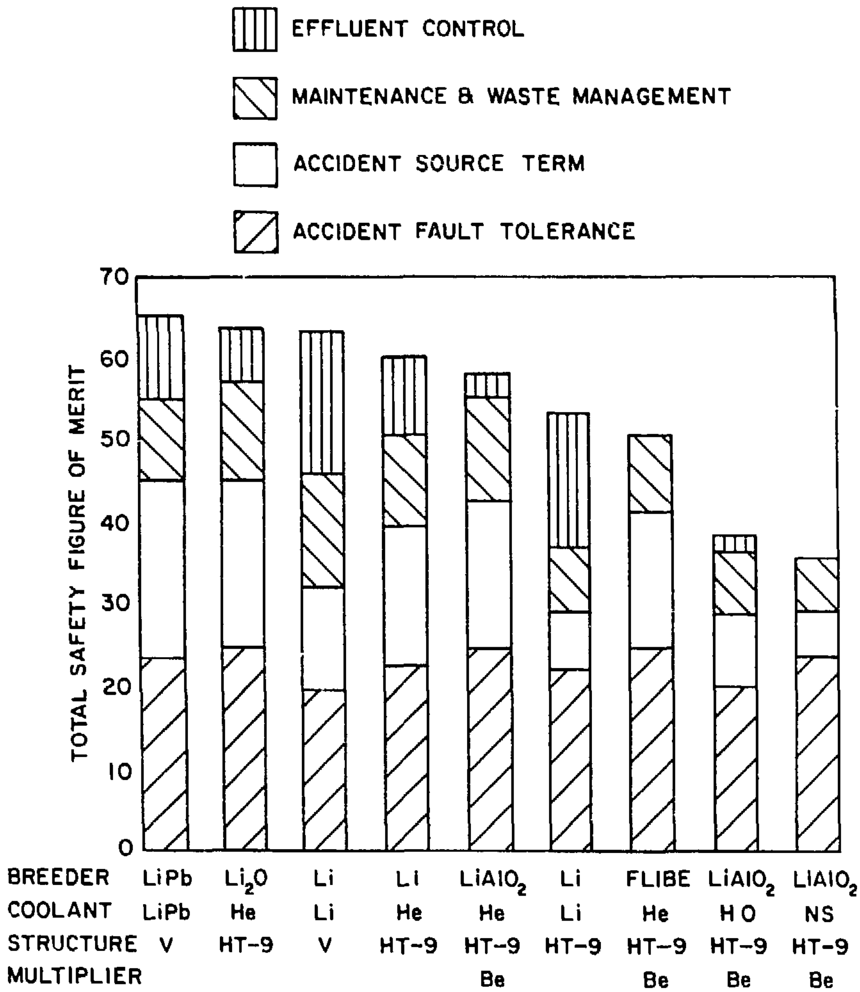


Figure 2-8. Safety evaluation results for mirror blankets.

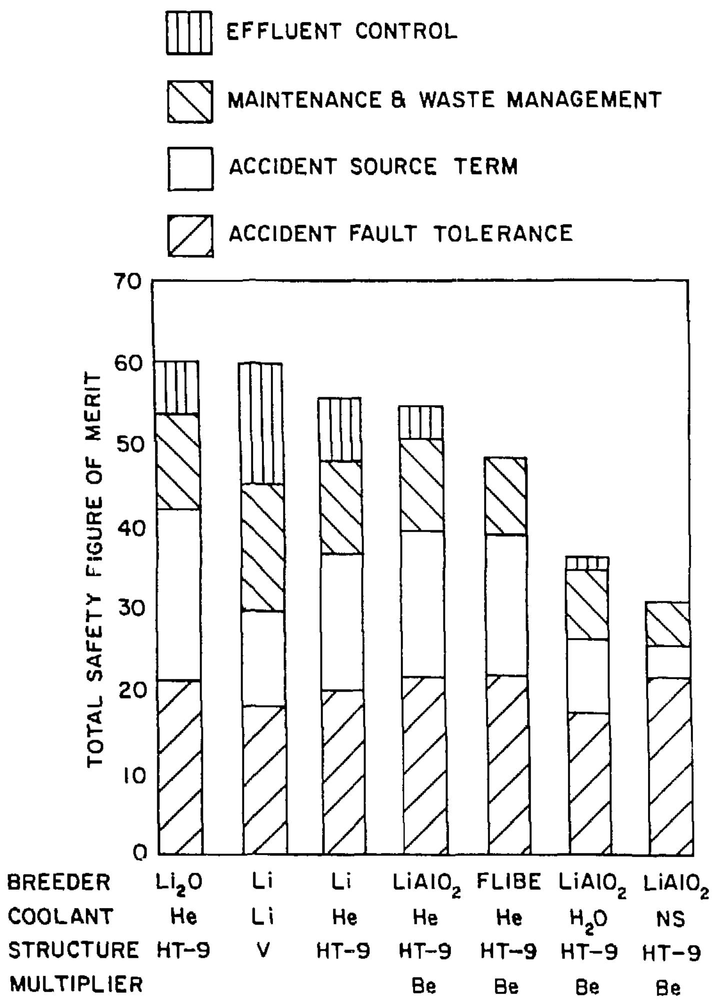


Figure 2-9. Safety evaluation results for tokamak blankets.

The two key areas of uncertainties and assumptions are tritium control and chemical reaction control. The reference results depend on the following: (a) use of nitrogen cover gas to reduce the severity of air-metal chemical reactions, e.g., lithium-fires and air-vanadium oxidation, (b) use of non-water-cooled limiters to reduce water-metal chemical reactions, (c) assumption that water-metal separation will be adequate to keep water-metal chemical reaction risk to a low level, (d) assumptions and data indicating that tritium control will be exceedingly difficult, and (e) assumptions that some tritium control techniques will 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 year) isotopes but replace them with elements, tungsten and manganese, that give 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. V15Cr5Ti 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 Sections 3 and 5.

2.5.4 Summary of Research and Development Concept Evaluation Results

The evaluation methodology for research and development (R&D) provides for an overall figure of merit (RDFM) which is given by

$$\mathrm{RDFM} = \frac{30}{\mathrm{RDR}} + \frac{1}{\mathrm{RDI}} \quad , \tag{2-4}$$

where RDR is a parameter that assesses the "risk" in carrying out the R&D for a particular blanket option and RDI is a 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.

2.5.4.1 R&D Investment Evaluation Results

The R&D investment parameter (RDI) 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

$$\mathrm{RDI} = \frac{X_1 + X_2 + X_3}{3} \quad . \tag{2-5}$$

Table 2-18 defines the score for X_1, X_2 and X_3. The results of the RDI evaluation are shown in Table 2-19. No significant differences for R&D resource requirements were identified for a given blanket concept between tokamak and tandem mirrors.

TABLE 2-18
RDI CATEGORIES

Time Scale	Score X_1	Average Annual Operating Cost	Score X_2	Required Facilities	Score X_3
< 10 years	(1)	< \$5M	(1)	No New Facilities > \$10M	(1)
10 years → 20 years	(2)	\$5-20M	(2)	New Facilities \$10-\$50M	(2)
> 20 years	(3)	> \$20M	(3)	New Facilities > \$50M	(3)

TABLE 2-19
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 2-18 for definition of categories for X_1, X_2, and X_3.

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 ferritic steels (i.e., HT-9) could be developed to that point in less than 10 years, thus $X_1 = 1$. Vanadium alloy blankets would require 10 to 20 years, thus $X_1 = 2$. The low-activation ferritic steels were also judged to require 10 to 20 years for development. No blanket option was judged to take longer than 20 years, given adequate funding.

All blankets were judged to require annual R&D operating costs of > $5M. The blanket concepts employing Li and Li_2O without neutron multipliers were judged to be in the range of $5M to $20M per year, thus $X_2 = 2$. LiPb was judged to require more resources than Li. Similarly, ternary ceramics (TC) blankets with Be 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 ferritic steel 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 will be 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 fission reactor and ion irradiation 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 ferritic steel 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 ternary ceramics with a Be neutron multiplier have somewhat less resource requirements than LiPb.

2.5.4.2 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).

A total of 29 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 2-20) 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 Chapter 4.

In general, the issues are grouped in Table 2-20 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, electromagnetic effects - 29) apply to almost all blanket concepts.

There are some important differences between tokamaks and tandem mirrors;

TABLE 2-20
BLANKET CONCEPT KEY ISSUES RATING

[R=CxP, TOK
TMR]

Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He FS/Be	$LiAlO_2$/H_2O/ FS/Be	$LiAlO_2$/NS/ FS/Be	FLiBe/He FS/Be
1. Unsatisfactory weld/fabrication of structural materials	3x2 3x2	-- 3x2	-- 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2
2. Excessive embrittlement of structure by hydrogen		--	-- 2x1		2x1 2x1	2x1 2x1	2x1 2x1	2x1 2x1	2x2 2x2
3. Unacceptable radiation-induced embrittlement of structure including DBTT concerns	3x1 3x1	-- 3x1	-- 3x1	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x1 3x1	3x2 3x2
4. V-blanket requires non-V balance of plant (BOP)	2x1 2x1	--	-- 2x1						
5. Risk from reactivity of structure with environment (inability to use inert atmosphere)	3x1 3x1	--	-- 3x1						
6. Risk from reactivity of coolant and breeder with environment (inability to use inert atmosphere)	3x1 3x1	-- 3x1	-- 2x1	2x1 2x1					
7. Corrosion worse than expected (includes non-V BOP in V designs)	2x2 2x2	-- 2x2	-- 2x2	2x2 2x2	1x2 1x2			2x2 2x2	2x1 2x1
8. MHD effects substantially worse	3x2 2x1	-- 2x1	-- 2x1	2x1					
9. Insulators not developed for liquid metal blanket	3x1 2x1	-- 2x1	-- 2x1	2x1					
10. Inability to develop non-water cooled near plasma components	3x2 3x1	-- 3x1	-- 2x1	2x2 2x1					
11. Difficult to meet seismic requirements		--	-- 2x2						

Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He FS/Be	$LiAlO_2$/H_2O/ FS/Be	$LiAlO_2$/NS/ FS/Be	FLiBe/He FS/Be
12. Inadequate tritium breeding	2x1 2x1	-- 2x1	--	2x1 2x1	3x2 3x2	3x1 2x1	3x1 2x1	2x1	3x1
13. Inability to accommodate breeder swelling		--	--		3x2 3x2	2x1 2x1	2x1 2x1	2x1 2x1	
14. Temperature range for tritium release much less than predicted		--	--		3x2 3x2	2x2 2x2	2x2 2x2	2x2 2x2	
15. Unacceptable temperature predictability of breeder (e.g., breeder-to-structure gaps are created)		--	--		3x1 3x1	3x1 3x1	3x2 3x2	3x1 3x1	
16. Unacceptable power variation capability		--	--		2x1 2x1	2x1 2x1	2x2 2x2	2x1 2x1	
17. Fabrication/refabrication of solid breeder		--	--		2x2 2x2	2x1 2x1	2x2 2x2	2x2 2x2	
18. Tritium recovery/leakage/control worse than predicted		--	-- 3x2	3x1 3x1	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x3 3x3
19. Loss of Be integrity is a major problem		--	--			1x3 1x3			1x2 1x2
20. Inability to reprocess Be in efficient manner		--	--			2x1 2x1	2x2 2x2	2x2 2x2	2x1 2x1
21. T-release from Be to primary coolant		--	--			2x2 2x2			2x2 2x2
22. Excessive chemical reactivity of Be with salt		--	--					3x2 3x2	
23. Salt stability/decomposition worse than predicted		--	--					3x2 3x2	

Key Issue		Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He/FS/Be	$LiAlO_2$/H_2O/FS/Be	$LiAlO_2$/NS/FS/Be	FLiBe/He/FS/Be
24. Inadequate performance of grooved first wall (tokamaks only)		2x2	--	--	3x2	3x2	3x2	2x2	2x2	3x2
25. Excessive coolant leakage to plasma		2x1 2x1	-- 2x1	-- 2x1	2x2 2x2	2x2 2x2	2x2 2x2	2x2 2x2		2x2 2x2
26. Difficult cleanup after spill			--	-- 2x2						2x2 2x2
27. Coolant containment reliability of double tubed wall less than predicted			--	--				3x1 3x1		
28. Excessive activation products, difficult to control			--	-- 2x2					2x3 2x3	
29. Electromagnetic effects worse than assumed		3x1 2x1	-- 2x1	--	2x1	2x1	2x1	2x1	2x1	2x1
<u>TOTALS</u>	Tokamak	47	---	---	43	61	57	60	66	54
RDR	TMR	34	32	48	29	53	48	53	58	43

thus an entry is made in Table 2-20 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 tandem mirrors. The results are shown at the bottom of Table 2-20.

In summary, the Li/He/FS blanket represents the minimum R&D risk (lowest total of Σ C x P) for both tokamaks and tandem mirrors. The Li/Li/FS and Li/Li/V designs are the next lowest risk designs for mirrors while Li/Li/V is the second lowest risk design for tokamaks. The highest risk designs are the TC concepts with Be multipliers, particularly with a nitrate salt coolant, for both tandem mirrors and tokamaks.

2.5.4.3 Composite R&D Evaluation Results

The composite score for the R&D evaluations are shown in Table 2-21. The relative ranking of the blanket concepts is also indicated in each table.

TABLE 2-21
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

2.5.5 Overall Evaluation

The previous sections present the results of the engineering feasibility, economic, safety, and R&D evaluations. Tables 2-22 and 2-23 summarize the relative ratings and the rankings of the tokamak and tandem mirror blankets, respectively, in each of the four categories. The rating of the top blanket concept in each category has been normalized to unity.

TABLE 2-22
TOKAMAK BLANKET RANKING

	Engineering	Economics	Safety	R&D	Overall[b]
Li/Li/V Li/Li/FS LiPb/LiPb/V	1.000 (1)	.85 (3)	.998 (2)	.886 (2)	1.000 (1)
Li/He/FS	.750 (3)	.73 (7)	.925 (3)	1.000 (1)	.842 (3)
Li_2O/He/FS	.719 (4)	.79 (5)[a]	1.000 (1)	.840 (3)	.878 (2)
$LiAlO_2$/He/FS/Be	.611 (7)	.79 (5)[a]	.904 (4)	.754 (5)	.806 (6)
$LiAlO_2$/H_2O/FS/Be	.682 (5)	1.00 (1)	.597 (6)	.723 (6)	.805 (7)
$LiAlO_2$/NS/FS/Be	.849 (2)	.98 (2)	.515 (7)	.692 (7)	.831 (4)
FLIBE/He/FS/Be	.658 (6)	.84 (4)	.807 (5)	.824 (4)	.809 (5)

[a]Tie in ranking.

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

TABLE 2-23
TMR BLANKET RANKING

	Engineering	Economics	Safety	R&D	Overall[b]
Li/Li/V	.999 (2)	.92 (4)[a]	.974 (3)	.852 (3)	1.00 (1)
Li/Li/FS	.917 (7)	.94 (2)[a]	.832 (6)	.944 (2)	.922 (7)
LiPb/LiPb/V	.959 (4)	.90 (6)[a]	1.000 (1)	.617 (7)[a]	.982 (2)
Li/He/FS	.939 (5)	.85 (9)	.936 (4)	1.000 (1)	.943 (4)
Li_2O/He/FS	.927 (6)	.89 (8)	.987 (2)	.716 (5)	.970 (3)
$LiAlO_2$/He/FS/Be	.884 (8)	.90 (6)[a]	.905 (5)	.654 (6)	.927 (5)
$LiAlO_2$/H_2O/FS/Be	.811 (9)	.94 (2)[a]	.595 (8)	.617 (7)[a]	.793 (9)
$LiAlO_2$/NS/FS/Be	1.000 (1)	1.00 (1)	.552 (9)	.586 (9)	.863 (8)
FLIBE/He/FS/Be	.994 (3)	.92 (4)[a]	.782 (7)	.741 (4)	.924 (6)

[a]Tie in ranking.

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

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. However, because performance cannot be guaranteed for any of the blankets, 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 2-22 and 2-23 to identify the other top rated blanket concepts that should provide a focus for the R&D effort.

- 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.

- 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.

- Numerical averaging of the individual evaluations with either the three factor 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.

- A more qualitative comparison of the evaluation results including either the three factor 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 are summarized in Tables 2-22 and 2-23. The results obtained by this method of comparison with some additional judgemental considerations have been used to identify the four leading blanket concepts that should provide the focus for the R&D effort.

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 non-water-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 relate 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 judgemental 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:

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

- The evaluation indicates a relatively high importance factor for design simplicity. None of the four concepts require a neutron multiplier.

- 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 ranke concepts.

- 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.

2.6 Summary

A two-year multilaboratory Blanket Comparison and Selection Study (BCSS) initiated by the U.S. Department of Energy/Office of Fusion Energy and led by Argonne National Laboratory was conducted to: (1) define a limited number of blanket concepts that should provide the focus for the blanket R&D program, and (2) identify and prioritize critical issues for the leading blanket concepts. The BCSS focused on the mainline approach for fusion reactor development, viz., the D-T-Li fuel cycle, tokamak and tandem mirror reactors 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:

- Development of reference design guidelines, evaluation criteria, and a methodology for evaluating and ranking candidate blanket concepts.

- Compilation of the required data base and development of a uniform systems analysis for comparison.

- Development of conceptual designs for the comparative evaluation.

- Evaluation of leading concepts for engineering feasibility, economic performance, and safety.

- Identification and prioritization of R&D requirements for the leading blanket concepts.

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

(Breeder/Coolant/Structure/Neutron Multiplier)

Li/Li/V	Li_2O/He/FS
Li/Li/FS*	$LiAlO_2$/He/FS/Be
LiPb/LiPb/V*	$LiAlO_2$/H_2O/FS/Be
Li/He/FS	$LiAlO_2$/NS/FS/Be
FLIBE/He/FS/Be	

(FS: Ferritic Steel; NS: Nitrate Salt)

*Evaluated only for TMR configuration.

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

- engineering
- economics
- safety
- 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.

The BCSS has met its primary objective of identifying a limited number of blanket concepts which should provide the focus of the blanket R&D program. These concepts include Li/Li/V, LiPb/LiPb/V (for TMR only), Li_2O/He/FS and Li/He/FS. The primary R&D issues for the Li/Li/V concept are the development of an advanced structural alloy, resolution of MHD and corrosion problems, provision for an inert atmosphere (e.g., N_2) in the reactor building, and the development of non-water cooled near-plasma components, particularly for the tokamak. The main issues for the LiPb/LiPb/V concept are similar to the Li/Li/V blanket with the addition of resolving the tritium recovery issue. Furthermore, resolution of MHD and corrosion problems will be more severe for LiPb/LiPb/V than for the Li/Li/V; on the other hand, the LiPb blanket has reduced concerns with respect to chemical reactivity with environment. The R&D issues for Li_2O/He/FS concept include resolution of the tritium recovery/containment issue, achieving adequate tritium breeding and resolving other solid breeder issues such as swelling and fabrication concerns. Major concerns for the Li/He/FS concept are related to its rather poor economic performance. Improvement of its economic performance will be somewhat concept-dependent and will be more of a systems engineering issue rather than a materials or blanket technology R&D issue.

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BLANKET COMPARISON AND SELECTION STUDY

CHAPTER 3 - CONCEPT EVALUATION RESULTS

3. CONCEPT EVALUATION RESULTS

This chapter contains the results of the detailed evaluations of the top rated blankets in the BCSS: seven blankets for tokamaks and nine blankets for tandem mirrors. The four major areas of evaluation include:

- Engineering Feasibility (Section 3.1)
- Economics (Section 3.2)
- Safety and Environment (Section 3.3)
- Research and Development (Section 3.4)

The general perspective on the overall evaluation of blanket concepts is summarized in Section 3.5.

3.1 Engineering Feasibility

The Engineering Feasibility ("engineering") evaluation of the sixteen concepts ranked R=1 was conducted for the nine TMR and seven tokamak concepts as separate groups. The evaluation followed the methodology presented in Sec. 5 2. Results of the evaluation are presented and discussed in Sections 3.1.1 through 3.1.7 for each of the seven indices listed in Table 3.1-1. A discussion of all engineering evaluation results is presented in Sec. 3.1.8, together with some general conclusions drawn from those results.

Table 3.1-1. ENGINEERING EVALUATION INDICES

INDEX NAME		WEIGHTING VALUE (W_i)
1. Tritium Breeding and Inventory		25
2. Engineering Complexity and Fabrication		25
3. Maintenance and Repair		15
4. Use of Resources		5[a]
5. Accommodation of Power Variations		10
6. Increased Capability		10
6.1 Increased Neutron Wall Loading	5	
6.2 Higher Surface Heat Flux, Higher Erosion Rate	5	
7. Startup/Shutdown Requirements		10

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

3.1.1 Tritium Breeding and Inventory (I_1; 25 points)

The scores for this index for all sixteen concepts are listed in Table 3.1-2, together with tritium breeding ratio (TBR) values calculated using 3-D geometry, G_o (required doubling time gain), and estimated uncertainty values.

The dominant factor, other than TBR, is the uncertainty in the doubling time gain, δ_G. The score is only a weak function of the other values in the equation over the ranges indicated. Because δ_G for all concepts was fixed at 0.224 (see Sec. 6.8), TBR became the most important discriminator. The 3-D TBR values were calculated for the reference designs. Thus, although higher TBR values could likely be achieved in most cases to a greater or lesser extent, the scores in some other area(s), principally economics, would probably have suffered.

Clear trends are difficult to discern from these results. The salt-cooled concepts with Be score well, and appear to have adequate breeding margin for nearly all contingencies. The Li/Li/V tokamak blanket also scores well, but its TMR counterpart and the other two self-cooled liquid metal blankets do not score as well. The Li_2O/He/FS and Li/He/FS concepts score poorly for both reactor approaches; both are probably close to their maximum achievable TBR values. The water-cooled $LiAlO_2$ concept with Be scores reasonably well for the TMR, but not for the tokamak where a thin inboard blanket was used in an effort to maximize the concept's economics score. A similar trend is seen for the He-cooled Flibe blanket, which also ties for the top rank in the TMR group. The He-cooled $LiAlO_2$ TMR concept scored 1/3 of the possible points, but the tokamak version scores no points because its 1.04 TBR is below the minimum allowable for the index. (Note: This was principally the result of the design group's choice to keep Be out of the inboard blanket. In retrospect, Be should have been added, which would have given the concept ~5 to 6 points for I_1 with little added penalty in other evaluation areas.)

Based on these results and those of other neutronics analyses conducted within the BCSS, the following grouping of concepts by tritium breeding potential was constructed. Any potential improvement in TBR was assumed to be achieved only by changes to ^6Li enrichment levels or moderate increases in blanket thickness, but not by major configuration changes or the introduction

TABLE 3.1-2. TRITIUM BREEDING AND INVENTORY INDEX, I_1

- Methodology - See Sec. 5.2.1
- G_0 = Required Gain In Doubling Time
  δ_G = Uncertainty In Predicting Required G_0
  δ_s = Uncertainty Associated With System Definition
  δ_p = Uncertainty In Predicting TBR For Given System
- Maximum Possible Score = 25 Pts

REACTOR		CONCEPT	3-D TBR	$(1+G_0)$	δ_G^2	δ_s^2	δ_p^2	FIGURE OF MERIT, F	SCORE
	A	$LiAlC_2$/NS/FS/Be	1.29	1.069	.05	.0094	.0009	.811	20.3
	B	Li/Li/FS	1.14	1.068	.05	.0094	.0035	.265	6.6
	C	LiPb/LiPb/V	1.18	1.067	.05	.0094	.0024	.417	10.4
T	D	Li/Li/V	1.19	1.068	.05	.0094	.0041	.445	11.1
M	E	Li_2O/He/FS	1.14	1.067	.05	0094	.0029	.270	6.8
R	F	$LiAlO_2$/He/FS/Be	1.16	1.067	.05	.0094	.0009	.350	8.8
	G	Li/He/FS	1.13	1.067	.05	.0094	.0030	.234	5.9
	H	FLIBE/He/FS/Be	1.29	1.067	.05	.0094	.0017	.812	20.3
	I	$LiAlO_2/H_2O$/FS/Be	1.22	1.070	.05	.0094	.0009	.557	13.9
	A	$LiAlO_2$/NS/FS/Be	1.24	1.073	.05	.0094	.0009	.616	15.4
T	B	-	-	-	-	-	-	-	-
O	C	-	-	-	-	-	-	-	-
K	D	Li/Li/V	1.28	1.068	.05	.0094	.0041	.734	18.4
A	E	Li_2O/He/FS	1.11	1.067	.05	.0094	.0029	.160	4.0
M	F	$LiAlO_2$/He/FS/Be	1.04	1.067	.05	.0094	.0009	0	0
A	G	Li/He/FS	1.12	1.068	.05	.0094	.0030	.193	4.8
K	H	FLIBE/He/FS/Be	1.17	1.067	.05	.0094	.0017	.384	9.6
	I	$LiAlO_2/H_2O$/FS/Be	1.16	1.071	.05	.0094	.0009	.333	8.3

of different materials such as Be. Full coverage is assumed, and breeding uncertainties and risks (Sec. 5.2.1) are not considered.

Estimated TBR Potential		
High (>1.4)	Medium (>1.2)	Low
$LiAlO_2$/NS/FS/Be	Li/He/FS	Li_2O/He/FS
LiPb/LiPb/V	$LiAlO_2$/He/FS/Be	
Flibe/He/FS/Be	$LiAlO_2$/H_2O/FS/Be	
Li/Li/FS		
Li/Li/V		

3.1.2 Engineering Complexity and Fabrication (I_2; 25 points)

Table 3.1-3 lists the total score for each concept in this category together with the scores for the individual subindices (see Sec. 5.2.2).

The self-cooled liquid metal blankets score very well, and are at the top of both groups. This result reflects the perceived relative simplicity in the designs (minimum number of different materials and zones or components within the blankets) and the resultant relative simplicity in fabrication.

The helium-cooled Flibe and Li concepts score relatively well, ~60-70% of total possible, whereas the helium-cooled Li_2O and $LiAlO_2$/Be concepts score rather poorly, near the bottom of the two groups. The principal differences are (1) the relative simplicity in both breeder fabrication/installation for the two liquid breeders, and (2) the much simpler method of tritium removal for those two concepts--breeder recirculation as compared to a separate, sealed helium purge gas system for the two solid breeders. In addition, the He-cooled $LiAlO_2$/Be concept requires a more complex scheme for Be fabrication retention than does the Flibe design.

The salt-cooled and water-cooled designs are similar, but the salt-cooled concept gains points because of its very low coolant pressure, and because it has single wall coolant tubes as compared to the double walled tubes considered necessary in the water-cooled concept for adequate reliability against leaks.

Table 3.1-3. ENGINEERING COMPLEXITY AND FABRICATION INDEX, I_2

Methodology: See Sec. 5.2.2

REACTOR	CONCEPT	FIRST WALL (0 →3)	NEUTRON MULT. (0 →3)	BREEDER FAB'N (0 →3)	CLNT CONT. FLOW PATH (0 →3)	TRITIUM REMOVAL (0 →3)	MANI-FOLDS (0 →3)	INBD/OUTBD BLKT (0 →3)	MFG OP'NS (0 →6)	SCORE[a] (0 →25)
	A $LiAlO_2$/NS/FS/Be	2	1.5	1.5	1.3	1	2	3	2.5	13.7
	B Li/Li/FS	3	3	3	1.8	2	3	3	5.0	22.0
	C LiPb/LiPb/V	3	3	3	1.8	2	3	3	4.5	21.6
T	D Li/Li/V	3	3	3	1.8	2	3	3	4.5	21.6
M	E Li_2O/He/FS	2	3	0	1.0	0	2	3	2.5	12.5
R	F $LiAlO_2$/He/FS/Be	2	1	0	1.0	0	2	3	2.5	10.6
	G Li/He/FS	2	3	2	1.0	2	2	3	3.0	16.7
	H FLIBE/He/FS/Be	2	2	2	1.0	2	2	3	2.5	15.3
	I $LiAlO_2$/H_2O/FS/Be	2	1.5	1.5	0	0	2	3	1.5	10.7
	A $LiAlO_2$/NS/FS/Be	2	1.5	1.5	1.3	1	2	3	2.5	13.7
T	B	-	-	-	-	-	-	-	-	-
O	C	-	-	-	-	-	-	-	-	-
K	D Li/Li/V	2	3	3	1.0	2	3	3	4.0	19.4
A	E Li_2O/He/FS	2	3	0	1.0	0	2	3	2.5	12.5
M	F $LiAlO_2$/He/FS/Be	2	1	0	1.0	0	2	3	2.5	10.7
A	G Li/He/FS	2	3	2	1.0	2	2	3	3.0	16.7
K	H FLIBE/He/FS/Be	2	2	2	1.0	2	2	3	2.5	15.3
	I $LiAlO_2$/H_2O/FS/Be	2	1.5	1.5	0	0	2	3	1.5	10.7

[a]SCORE = (SUBTOTAL SUM) × $\dfrac{\text{25 [MAX. ALLOWABLE FOR } I_2\text{]}}{\text{27 [TOTAL POSSIBLE FOR SUBINDICES]}}$

3.1.3 Maintenance and Repair (I_3; 15 points)

The He-cooled solid breeder concepts score very well (Table 3.1-4) for this index. This is the result of relatively simple sector replacement operations, essentially no spill cleanup problems, and designs that are tolerant of some leaks occurring inside each module. The He-cooled Flibe and Li concepts do not score as well because the handling of the liquid breeders entails more problems and complexity in maintenance operations.

The salt-cooled and H_2O-cooled concepts score moderately well, in between the SB/He and self-cooled liquid metal groups. The salt-cooled concept scores below the similar H_2O-cooled concept because of greater difficulties in cleanup after a coolant spill. The water-cooled concept scored the maximum number of points in the leak tolerance category, reflecting in this case the perceived very high reliability against the occurrence of coolant leaks into the breeder afforded by the double-walled coolant tubes--a feature for which the blanket is penalized in many other areas.

The self-cooled liquid metal blankets are near or at the bottom of the tokamak and TMR groups for this index. Cleanup following a breeder/coolant spill is more difficult, especially for LiPb. The TMR blanket configuration, with multiple discrete tubes, scores poorly in leak tolerance compared to the monolithic configuration adopted for the tokamak blanket.

3.1.4 Resource Requirements (I_4; 5 points)

One of the major criticisms of Be as a potential neutron multiplier for power reactor blankets has been the question of resource limitations. Work was done during the study to determine the requirements for any potentially resource-limited material--e.g., Be, Li--assuming 1000 GWe produced for 40 years by reactors using any of the top-rated (R=1) blankets.

Table 3.1-5 shows the results. The requirements for making up losses due to burnup and remanufacturing, as a percent of U.S. or world resources, were used to determine the score. The most marked distinction is between the He-cooled $LiAlO_2$ concept and the others using Be. The other concepts use up to a factor of ~2.5 to 4 more Be over the 1000 GWe/40 yr span. Even so, the He-cooled $LiAlO_2$ concepts scores only one point more than the others, since these

Table 3.1-4. MAINTENANCE AND REPAIR INDEX, I_3

o Methodology: See Sec. 5.2.3.

REACTOR		CONCEPT	INSPECTION PROCEDURES (0 → .2)	REPLACEMENT OPERATIONS (0 → .3)	CLEANUP OPERATIONS (0 → .2)	LEAK TOLERANCE (0 → .3)	SCORE[a]
	A	$LiAlO_2$/NS/FS/Be	.1	.2	.05	.3	9.8
	B	Li/Li/FS	.1	.2	.1	.1	7.5
	C	LiPb/LiPb/V	.1	.2	0	.1	6.0
T	D	Li/Li/V	.1	.2	.1	.1	7.5
M	E	Li_2O/He/FS	.1	.3	.2	.3	13.5
R	F	$LiAlO_2$/He/FS/Be	.1	.3	.2	.3	13.5
	G	Li/He/FS	.1	.2	.1	.3	10.5
	H	FLIBE/He/FS/Be	.1	.2	.05	.3	9.8
	I	$LiAlO_2$/H_2O/FS/Be	.1	.2	.15	.3	11.3
	A	$LiAlO_2$/NS/FS/Be	.1	.2	.05	.3	9.8
T	B	-	-	-	-	-	-
O	C	-	-	-	-	-	-
K	D	Li/Li/V	.1	.2	.1	.3	10.5
A	E	Li_2O/He/FS	.1	.3	.2	.3	13.5
M	F	$LiAlO_2$/He/FS/Be	.1	.3	.2	.3	13.5
A	G	Li/He/FS	.1	.2	.1	.3	10.5
K	H	FLIBE/He/FS/Be	.1	.2	.05	.3	9.8
	I	$LiAlO_2$/H_2O/FS/Be	.1	.2	.15	.3	11.3

[a]SCORE = (SUM OF SUBINDICES) X 15 (MAX. ALLOWABLE POINTS)

Table 3.1-5. RESOURCE REQUIREMENTS INDEX, I_4

o Methodology: See Sec. 5.2.4.

				% OF REQUIREMENTS[c]				
				LOSSES ONLY (BURNUP + REMANUFACTURING)			LOSSES PLUS INVENTORY[d]	
REACTOR	CONCEPT	MAT'L	INVENTORY, Tonnes/GWe	U.S.[a]	WORLD[b]	SCORE[e]	U.S.	WORLD
	A $LiAlO_2$/NS/FS/Be	Be	55.8	43	2.8	1.5	119	7.9
	B Li/Li/FS	--	--	--	--	5.0	--	--
	C LiPb/LiPb/V	--	--	--	--	5.0	--	--
T	D Li/Li/V	--	--	--	--	5.0	--	--
M	E Li_2O/He/FS	--	--	--	--	5.0	--	--
R	F $LiAlO_2$/He/FS/Be	Be	37.8	15	1	2.5	67	4.4
	G Li/He/FS	--	--	--	--	5.0	--	--
	H FLIBE/He/FS/Be	Be	95.6	38	2.5	1.5	170	11
	I $LiAlO_2$/H_2O/FS/Be	Be	78.2	60	4	1.5	167	11
	A $LiAlO_2$/NS/FS/Be	Be	58.8	45	3	1.5	126	8.3
T	B -	--	--	--	--	--	--	--
O	C -	--	--	--	--	--	--	--
K	D Li/Li/V	--	--	--	--	5.0	--	--
A	E Li_2O/He/FS	--	--	--	--	5.0	--	--
M	F $LiAlO_2$/He/FS/Be	Be	28.1	11.0	0.75	2.5	50	3.3
A	G Li/He/FS	--	--	--	--	5.0	--	--
K	H FLIBE/He/FS/Be	Be	78.3	32	2.1	1.5	139	9.2
	I $LiAlO_2$/H_2O/FS/Be	Be	60.0	46	3	1.5	128	8.5

[a] U.S. Bureau of Mines estimate = 73 KT.
[b] U.S. Bureau of Mines estimate = 1105 KT.
[c] 40 yrs operation at 1000 GWe/yr.
[d] Information only; not for scoring.
[e] Score = (Index) x (5 total allowable points)

requirements are still only a few percent of world Be resources. Also shown for information is the total of Be losses plus the initial blanket inventory.

Lithium usage was examined also. The results indicated that Li recycling was required for all concepts, but that resource requirements were very low and would not affect the scores shown. All structural alloys use roughly the same amounts of chromium (Cr), which for the 1000 GWe/40 yr criteria would use up all U.S. resources but much less than 1% of world reserves. Thus, no ranking seemed reasonable among the concepts for Cr. Based on the Li and Cr results, all concepts not using Be were given a full score of 5 points for this index.

3.1.5 Allowable Power Variation (I_5; 10 points)

This index measures the capability built into the reference design for each concept to accommodate deliberate changes in reactor power level, assumed to be done by changing the neutron wall load, P_{nw}. The rationale for the methodology is explained in Sec. 5.2.5.

The results presented in Table 3.1-6 indicate some clear distinctions among concepts. The self-cooled liquid metal concepts score virtually all the possible points. They can be operated with extremely low mass flowrates without violating any limits. The vanadium self-cooled liquid metal concepts can be operated at 120% or more of the reference P_{nw} (5 MW/m^2) before the 750^0C temperature limit or alloy stress limits would be violated. The Li/Li/FS concept is already at the maximum allowable structural temperature for the nominal case.

The He-cooled designs with solid breeders or Li can be operated down to ~5% of the nominal P_{nw} by dumping steam at P_{nw} < 25%, while holding the 25% value of coolant mass flowrate constant and sharply increasing inlet temperature. The Flibe concept cannot operate below 25% because temperature limits for the Flibe tubes would be exceeded when coolant inlet temperature was increased. Upper limits on P_{nw} vary for the helium-cooled concepts. All would exceed breeder or structure temperature limits at P_{nw} < 120% except for the Li/He/FS TMR concept which can take up to 140% of the nominal P_{nw} before first wall temperature exceeds the 550^0C maximum allowable.

Table 3.1-6. POWER VARIATION INDEX, I_5

o Methodology: See Sec. 5.2.5.

REACTOR		CONCEPT	VARIATION RANGE	SCORE	COMMENT
	A	$LiAlO_2$/NS/FS/Be	0 → 120%	10	
	B	Li/Li/FS	0 → 100%	9	$(T_{STR})_{MAX} = 550^0C$ @ 100% POWER
	C	LiPb/LiPb/V	0 → 120%	10	
T	D	Li/Li/V	0 → 120%	10	
M	E	Li_2O/He/FS	5 → 106%	8.5	$(T_{DR})_{MAX} = 800^0C$ @ 106% POWER
R	F	$LiAlO_2$/He/FS/Be	5 → 100%	8	$(T_{STR})_{MAX} = 550^0C$ @ 100% POWER
	G	Li/He/FS	5 → 140%	10	
	H	FLIBE/He/FS/Be	25 → 100%	6	$(T_{STR})_{MAX} = 550^0C$ @ 100% POWER
	I	$LiAlO_2/H_2O$/FS/Be	~60 → 100%	3	$(T_{BR})_{MAX} > 1000^0C$ @ > 100% POWER; $(T_{BR})_{MIN} < 350^0C$ @ $\lesssim$ 60% POWER
	A	$LiAlO_2$/NS/FS/Be	0 → 120%	10	
T	B	-	-	-	
O	C	-	-	-	
K	D	Li/Li/V	0 → 120%	10	
A	E	Li_2O/He/FS	5 → 101%	8	$(T_{BRDR})_{MAX} = 800^0C$ @ 101% POWER
M	F	$LiAlO_2$/He/FS/Be	5 → 100%	8	SAME AS TMR
A	G	Li/He/FS	5 → 111%	9	$(T_{STR})_{MAX} = 550^0C$ @ 111% POWER
K	H	FLIBE/He/FS/Be	25 → 100%	6	SAME AS TMR
	I	$LiAlO_2/H_2O$/FS/Be	~60 → 100%	3	SAME AS TMR

[a]For reference design. Only coolant mass flowrate and inlet temperature are allowed to vary.

The salt-cooled concepts can operate at up to 120% of nominal P_{nw} before first wall temperature limits are violated, and can operate at very low P_{nw} values without violating breeder temperature limits by raising coolant inlet temperature such that breeder minimum allowable temperature (350^0C) is never violated.

The water-cooled concepts were not designed to operate at P_{nw} > 5 MW/m^2 without violating breeder maximum allowable temperature. As P_{nw} is reduced below 100%, breeder minimum temperature can be maintained at the 350^0C allowable level by (1) reducing breeder bulk thermal conductivity, by lowering helium purge gas pressure from 6 to 1 atm, and (2) increasing coolant inlet temperature above the 280^0C nominal design point. However, it is estimated that below ~60% power, the breeder minimum temperature will fall below 350^0C, implying an excessive tritium inventory buildup if the blanket is operated in this mode for extended time periods. The tritium would eventually be recovered over a period of time, once operation at P_{nw} >60% resumed.

3.1.6 Capability for Increases in Power Loading, Surface Heating and Particle Flux (I_6; 10 points)

This index is divided into two subindices worth a maximum of 5 points each. The results are summarized in Table 3.1-7.

Power Loading Increase ($I_{6.1}$) - The intent of this category is to measure the ultimate capability of the concept to operate at higher P_{nw} values, assuming that any necessary changes are made to the design. The relationships of surface heat flux and particle flux to P_{nw} were fixed, however, and design guidelines violations were not permitted. It should be noted that the changes necessary to accommodate the higher P_{nw} values might make the concept unattractive from the economic or engineering feasibility standpoints.

The TMR self-cooled liquid metal concepts are limited by the steam generator pinch point to ~50-60% P_{nw} increase, whereas the tokamak Li/Li/V concept is limited to a 20% increase because of thermal stress limits in the first wall resulting from the high surface heat flux. The He-cooled TMR concepts can each be designed to accommodate P_{nw} > 10 MW/m^2 by adjusting solid breeder plate or liquid breeder tube sizes. However, the tokamak versions are limited to increases of ~35-50% by the first wall 550^0C maximum structural temperature limit. The salt-cooled and water-cooled $LiAlO_2$ concepts, having

Table 3.1-7. CAPABILITY FOR INCREASES IN POWER LOADING, SURFACE HEATING AND PARTICLE FLUX, I_6

o Methodology: See Sec. 5.2.6

REACTOR		CONCEPT	MAXIMUM P_{nw} [a]	SCORE [c] ($I_{6.1}$)	COMMENTS	q_{max} @ $(t_e)_{ref}$ [a]	$(t_e)_{max}$ @ q_{ref} [b]	SCORE [c] ($I_{6.2}$)	COMMENTS
	A	$LiAlO_2$/NS/FS/Be	~8	3	HITS TBR LIMIT	>.10	>1	5	
	B	Li/Li/FS	7.5	2.5	HITS PINCH POINT	>.10	>1	5	
	C	LiPb/LiPb/V	7.5	2.5	HITS PINCH POINT	>.10	>1	5	
T	D	Li/Li/V	8	3	HITS PINCH POINT	>.10	>1	5	
M	E	Li_2O/He/FS	>10	5	ADJUST PLATE SIZE	>.10	>1	5	
R	F	$LiAlO_2$/He/FS/Be	>10	5	ADJUST PLATE SIZE	>.10	>1	5	
	G	Li/He/FS	>10	5	ADJUST Li TUBE SIZE	>.10	>1	5	
	H	FLIBE/He/FS/Be	>10	5	ADJUST FLIBE TUBE SIZE	>.10	>1	5	
	I	$LiAlO_2$/H_2O/FS/Be	~8	3	HITS TBR LIMIT	>.10	>1	5	
	A	$LiAlO_2$/NS/FS/Be	~8	3	HITS TBR LIMIT	1.2	>10	3	FW HITS $(T_{STR})_{MAX}$
T	B	-	-	-		-	-	-	
O	C	-	-	-		-	-	-	
K	D	Li/Li/V	6	1	THERMAL STRESS LIMIT	1.2	>10	3	FW HITS $(T_{STR})_{MAX}$
A	E	Li_2O/He/FS	7.0	2	$(T_{FW})_{MAX}$ IS LIMITING	1.01	0.5	0	FW HITS $(T_{STR})_{MAX}$
M	F	$LiAlO_2$/He/FS/Be	7.5	2.5	$(T_{FW})_{MAX}$ IS LIMITING	1.00	1.0	0	FW HITS $(T_{STR})_{MAX}$
A	G	Li/He/FS	6.7	1.7	$(T_{FW})_{MAX}$ IS LIMITING	1.01	0.5	0.3	FW HITS $(T_{STR})_{MAX}$
K	H	FLIBE/He/FS/Be	7.0	2	ADJUST TUBE SIZE	1.01	0.5	0.3	FW HITS $(T_{STR})_{MAX}$
	I	$LiAlO_2$/H_2O/FS/Be	~8	3	HITS TBR LIMIT	1.8	>10	4.5	FW HITS $(T_{STR})_{MAX}$

[a] In MW/m^2

[b] In millimeters

[c] Five points maximum possible score

rather similar breeding zones, are each estimated to reach the point at $P_{nw} \gtrsim 8$ MW/m^2 beyond which the minimum TBR value cannot be achieved because of extremely close coolant tube spacing and resultant high coolant and structure volume fractions.

Surface Heating and Particle Flux ($I_{6.2}$) - This index measures the capabilities of the reference design to accommodate either higher-than-predicted surface heat flux, q, at the nominal erosion thickness, t_e, or higher particle flux at the nominal surface heat flux. Reference q and t_e values are 1.0 MW/m^2 and 2 mm for the tokamak, and 0.05 MW/m^2 and 0.2 mm for the TMR. The erosion thickness allowance for tokamaks is assumed to be orthogonally grooved for low stresses, and is therefore not subject to structure temperature limits (Sec. 6.7).

All TMR concepts scored the maximum of 5 points. Any future increases in surface heat loads or particle flux, above the low initial values in the design guidelines, are expected to be small. These increases can easily be accommodated by the TMR concepts' first walls.

Among the tokamak concepts, the He-cooled designs are particularly limited in their capabilities for higher q or t_e. The q values are limited since they are already designed to operate with first wall structure at or near the maximum allowable temperature in order to maximize the helium coolant temperature and thus thermal conversion efficiency. The t_e values are limited because even the slight additional nuclear heating from increased t_e material puts the temperature of the first wall structure over the 550^{0}C temperature limit.

For the Li/Li/V concept, the "operating window" for the reference design disappears at q > 1.2 MW/m^2, but t_e can be 10 mm or more without violating structural temperature limits. The salt-cooled concept's reference design first wall exceeds the 550^{0}C structural temperature limit at q > 1.2 MW/m^2, but can handle t_e > 10 mm at q = 1.0. The water-cooled $LiAlO_2$ concept does the best of all the tokamak concepts, 7.5 points out of 10; its low coolant temperatures compared to the salt result in a lower structural temperature for the nominal q = 1.0 case, so that a greater q can be taken without exceeding the temperature limit.

3.1.7 Startup/Shutdown (SU/SD) Requirements (I_7; 10 points)

This index is intended to measure both the rapidity in bringing a reactor/blanket concept from "cold iron" conditions to operating readiness (($I_{7.1}$, 6 points maximum), and the degree to which the SU/SD operations do not force additional subsystems, e.g. trace heaters, to be added to the reactor ($I_{7.2}$, 4 points). Results are summarized in Table 3.1-8.

Table 3.1-8. STARTUP/SHUTDOWN (SU/SD) REQUIREMENTS, I_7

o Methodology: See Sec. 5.2.7.

REACTOR		CONCEPT	SCORES [a]SU/SD TIME REQTS, $I_{7.1}$	[b]SUBSYSTEMS NEEDED, $I_{7.2}$	COMMENTS
	A	$LiAlO_2$/NS/FS/Be	3	2	SAME AS TOKAMAK
	B	Li/Li/FS	3	2	DRAIN/FILL; He CIRC.
	C	LiPb/LiPb/V	3	2	"
T	D	Li/Li/V	3	2	"
M	E	Li_2O/He/FS	3	4	He CIRC.
R	F	$LiAlO_2$/He/FS/Be	3	4	He CIRC.
	G	Li/He/FS	3	3	He CIRC. DRAIN/FILL
	H	FLIBE/He/FS/Be	3	2	"
	I	$LiAlO_2$/H_2O/FS/Be	3	4	DRAIN/FILL
	A	$LiAlO_2$/NS/FS/Be	3	2	SALT LIQUEF.; DRAIN/FILL
T	B	-	-	-	
O	C	-	-	-	
K	D	Li/Li/V	3	2	DRAIN/FILL; He CIRC
A	E	Li_2O/He/FS	3	4	He CIRC.
M	F	$LiAlO_2$/He/FS/Be	3	4	He CIRC.
A	G	Li/He/FS	3	3	He CIRC.; DRAIN/FILL
K	H	FLIBE/He/FS/Be	3	2	DRAIN/FILL; He CIRC.
	I	$LiAlO_2$/H_2O/FS/Be	3	4	DRAIN/FILL

[a]Cold start time required. Six points maximum.

[b]Additional subsystems needed just for SU/SD requirements. Four points maximum.

SU/SD Time Required - From examination of SU/SD requirements for the concepts, it appeared that the governing factor for all cases would be filling and cleaning of the steam loop (time $\geq$ 1 day). Blanket startup and fusion power startup times together were roughly on the order of a day's time regardless of concept. Thus, combined startup times would be in the 1 to 3 day range. Based on this, all concepts scored 3 of the 6 possible points.

Subsystems Required - The helium-cooled solid breeder concepts score best here; blanket and coolant system warmup is accomplished using the existing He circulators. The Li/He/FS concept scores slightly lower because of the need for a relatively complex drain/storage/fill subsystem for the liquid Li. The water-cooled $LiAlO_2$ concept also scores the maximum of 4 points. The drain/storage/fill system is relatively simple with no heating system required, and blanket and coolant system warmup is done using the main coolant pumps to simultaneously circulate and heat the water. The salt coolant requires liquefaction by adding water to permit cooling it from above the ~220^{0}C melting point down to room temperature; the water must be removed before bringing the blanket up to operating temperature. This may involve high coolant presures, but these occur only for a short time and not during operation. The alternative to liquefaction is a drain/storage/fill subsystem identical to that for the liquid metals. The Flibe and self-cooled liquid metal breeder concepts all require a drain/storage/fill subsystem with heater systems. Blankets are preheated by circulating helium, an added subsystem for the self-cooled concepts. The Flibe's high melting temperature (363^{0}C) is an added complication since significant fusion power would probably have to be established before the He coolant inlet temperature could be lowered from the ~370-380^{0}C level to the 275^{0}C operating level.

3.1.8 Summary and Conclusions

The scores for all engineering evaluation indices are presented in Table 3.1-9 together with the score totals for each TMR and tokamak concept. The scores are also shown graphically in Fig. 3.1-1.

TMR Concepts - It is rather difficult to draw firm conclusions from the outcome of this evaluation, since the scores for all but the water-cooled concept fall within a relative narrow range. It seems fairly clear that the water-cooled concept is inferior to the other groups in this evaluation

Table 3.1-9. ENGINEERING EVALUATION - SUMMARY OF SCORES

REACTOR		CONCEPT	BREEDING & INVENTORY (25)	COMPLEXITY & FABRICATION (25)	MAINTENANCE (15)	RESOURCES (5)	POWER VARIATION (10)	P NW, q's, t e INCREASE CAP. (10)	STARTUP/ SHUTDOWN (10)	TOTAL SCORE (OF 100)	(SCORE ÷ HIGHEST SCORE)
	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	.917
	C	LiPb/LiPb/V	10.4	21.6	6.0	5	10	7.5	5	65.5	.959
T	D	Li/Li/V	11.1	21.6	7.5	5	10	8	5	68.2	.999
M	E	Li_2O/He/FS	6.8	12.5	13.5	5	8.5	10	7	63.3	.927
R	F	$LiAlO_2$/He/FS/Be	8.8	10.6	13.5	2.5	8	10	7	60.4	.884
	G	Li/He/FS	5.9	16.7	10.5	5	10	10	6	64.1	.939
	H	FLIBE/He/FS/Be	20.3	15.3	9.8	1.5	6	10	5	67.9	.994
	I	$LiAlO_2$/H_2O/FS/Be	13.9	10.7	11.3	1.5	3	8	7	55.4	.811
	A	$LiAlO_2$/NS/FS/Be	15.4	13.7	9.8	1.5	10	6	5	61.4	.849
T	B	-	-	-	-	-	-	-	-	-	-
O	C	-	-	-	-	-	-	-	-	-	-
K	D	Li/Li/V	18.4	19.4	10.5	5	10	4	5	72.3	1.000
A	E	Li_2O/He/FS	4.0	12.5	13.5	5	8	2	7	52.0	.719
M	F	$LiAlO_2$/He/FS/Be	0	10.7	13.5	2.5	8	2.5	7	44.2	.611
A	G	Li/He/FS	4.8	16.7	10.5	5	9	2.2	6	54.2	.750
K	H	FLIBE/He/FS/Be	9.6	15.3	9.8	1.5	6	2.3	5	49.5	.685
	I	$LiAlO_2$/H_2O/FS/Be	8.3	10.7	11.3	1.5	3	7.5	7	49.3	.682

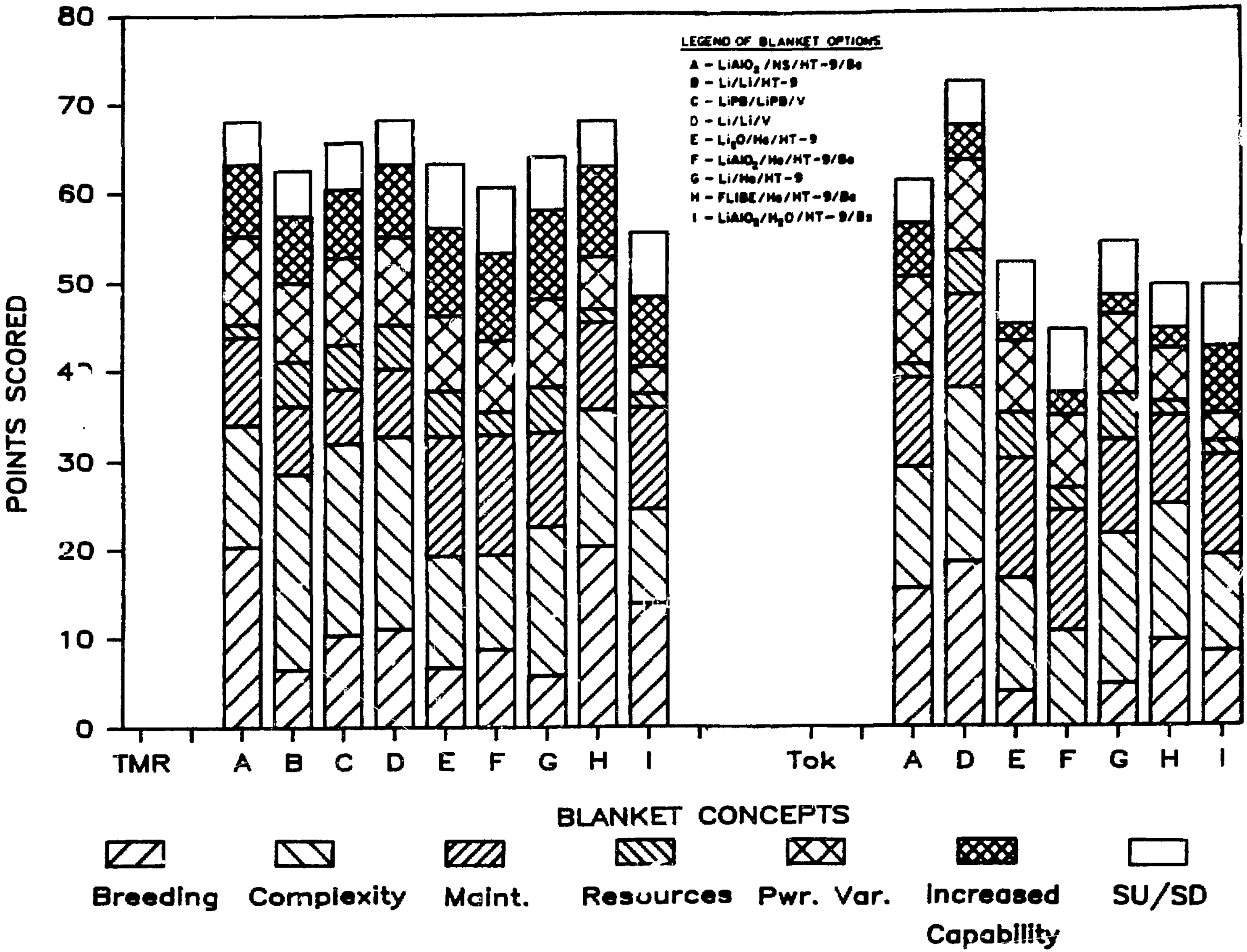


Fig. 3.1-1. Final results of engineering evaluation for concepts ranked R=1.

area. The concept scores poorly in the complexity and power variation indices, 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 as a class (Flibe excepted) do only slightly less well than the self-cooled liquid metal concepts. The $LiAlO_2$ concept scores below the other two He-cooled concepts, primarily due to lower scores in the complexity and resources categories which result from the need to add Be to the blanket. The Flibe blanket scores considerably higher than the other He-cooled concepts primarily because of the higher breeding ratio for its reference design; but it loses some points relative to the others in maintenance--because of the Flibe's presence--and resources--because of the Be neutron multiplier.

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 scores 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 high-temperature helium coolant, which affects the capability for increasing P_{nw}, q and t_e. The $LiAlO_2$/He/FS/Be concept would likely have scored in the same range as the He-cooled Li_2O and Li blankets if the decision had been made to incorporate Be neutron multiplier into the inboard blanket. This would have given it ~5-6 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 has much less capability for handling increases in q or 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 $P_{nw}/q/t_e$ increase categories relative to the He-cooled and H_2O-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 ~5-6 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. Note that if the Flibe breeder were to be enriched beyond the natural ^6Li level to take greater advantage of the thermal neutrons provided by the Be neutron multiplier, the Flibe concept's breeding score and overall score would improve significantly.

General Conclusions - From the results discussed above, .veral general conclusions can be drawn. From the engineering standpoint:

(1) The overall differences among ranked blanket concepts are considerably larger for tokamaks than for TMR's. These distinctions are brought out by relatively more difficult problems for blankets in tokamaks due to higher magnetic field strengths, higher surface heating and particle flux, 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 TMR's, 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 thinner 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 TMR's (where its relatively good cooling capabilities do not give it an advantage), and 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 presure, higher temperature capability, and better neutronics.

3.2 Economics Evaluation Results

The economic evaluation of the R=1 ranked blanket concepts considered not only the direct cost consequences but also how well the blankets performed in both tokamak and tandem mirror reactors. Due to the differing reactor and plant design features of the two principal reactor types (tokamak and tandem mirror), the blanket concepts were ranked and compared only within a particular reactor type. The final, single economic criterion is the Cost of Electricity (COE) with the blanket employed in either the tokamak or the tandem mirror. Major factors contributing and influencing the COE are the direct capital cost items, the annual cost items and the reactor and plant performance. The choice of blanket design may also affect the plant availability, but assessment of this influence was beyond the scope of this study and so, availability was held constant for all blanket designs.

The following subsections discuss cost and performance attributes of each blanket concept when employed in the two reactor types. The final economic evaluations are developed and presented in terms of their relative COE values. These values will be used in the overall ranking of the blanket concepts. Since several of the major factors contributing to the final economic rankings were subject to a significant degree of uncertainty, sensitivity studies were conducted to assess the impact of these uncertainties.

3.2.1 Discussion of Major Economic Factors

This subsection will discuss the major factors of each blanket which will influence the plant economics and which represent the elements contributing to the Cost of Electricity (COE). The next subsection (Section 3.2.2) will discuss the summation of these factors into the high level cost elements which contribute directly to the COE (e.g., Total Capital Cost, Net Electric Power, etc.). This Subsection 3.2.1 will highlight the underlying, causative economic factors affecting the blanket concepts.

A word of caution should be mentioned. All evaluation groundrules, economic and otherwise, were established early in this design process to allow the design advocates and designers to improve the blanket concepts' final scoring. Trial evaluation runs were conducted during the study to exercise the evaluation procedures and to allow feedback to the designers on the individual concepts' weak and strong points. However, as the study drew to a close, the design activity had to be halted and the final evaluation process completed. As the final evaluation values began to emerge, the designers would have liked to have had one more design cycle to more fully optimize their respective designs, but no more time was available. So the individual detriments presented herein on each design should be viewed with caution, remembering that some of these were created in order to optimize on other performance factors. Usually the detriments can be reduced to some degree with additional design activity and/or R&D work.

Blanket and Shield - The significant influencing factors in this area are the direct capital costs of the First Wall, Blanket Structure, Breeder, Multiplier, Reflector, Plenum, Coolant, and Shield. The methodology of these cost estimates and associated unit costs are discussed in Sec. 5.3.1 and 5.3.2. The volumes and masses were calculated from the blanket composition data and the 1-D neutronics requirements for the particular reactor configuration. The volume fractions and percent of theoretical density were input variables from the designers. Table 3.2-1 lists all the volumes and masses of blanket components and the primary coolant. These data are used for the Economic Evaluations and the Safety and Environmental Effects, Section 3.3.

The nitrate salt designs (A) have the lowest volume and mass of the structure and primary coolant in the blanket. The Flibe designs (H) have the heaviest structure. Not all designs required a reflector, which was usually composed of HT-9 except for a SiC reflector in the Flibe designs. The SiC is much lighter in weight than the HT-9 designs.

The breeder was either lithium or a lithium compound. In the case of the liquid lithium or lithium-lead, which functioned also as a coolant, the volume and weight was counted as a coolant in this table. The solid breeders (A, E, F and I) used the most volume and were the heaviest, while Flibe (H) was designed with the least volume and weight. The helium-cooled, lithium breeder was intermediate between the solids and Flibe. The lead multiplier in the lithium-lead is also listed as a coolant. A beryllium multiplier was used with the $LiAlO_2$ and the Flibe designs. The beryllium theoretical density ranged from 0.7 to 1.0, with fabrication forms including spherepac, rods, and balls. All blankets required coolant volumes from 150 to 380 m^3 regardless of coolant composition. The only exception is the nitrate salt which required 50

TABLE 3.2-1
BCSS VOLUMES AND MASSES

Concept	Structure (m3)	Structure (Mg)	Reflector (m3)	Reflector (Mg)	Breeder (m3)	Breeder (Mg)	Multiplier (m3)	Multiplier (Mg)	Blanket Coolant (m3)	Blanket Coolant (Mg)	Total Coolant (m3)	Total Coolant (Mg)
Tandem Mirror												
A - $LiAlO_2$/NS/HT-9/Be	36.5	292.9			224.2	495.9	59.4	76.9	50.1	92.7	436.3	714.6
B - Li/Li/HT-9/-	59.2	474.7	255.5	2048.8	See Coolant				299.0	149.5	699.7	349.8
C - LiPb/LiPb/V/-	71.5	427.8	309.9	2485.6	See Coolant		See Coolant		355.6	3342.4	1091.7	10261.9
D - Li/Li/V/-	59.2	353.9	255.5	2048.8	See Coolant				299.0	149.5	699.7	349.8
E - Li_2O/He/HT-9/-	79.9	641.1			306.9	493.5			156.2	0.5	3622.8	12.0
F - $LiAlO_2$/He/HT-9/Be	64.6	518.1			188.3	391.7	32.6	48.2	153.0	0.5	3799.1	12.5
G - Li/He/HT-9/-	103.1	827.0	143.7	1152.2	475.9	237.9			323.2	1.1	3915.0	12.9
H - Flibe/He/HT-9/Be	149.5	1198.8	194.7	626.4	41.2	82.4	68.4	126.6	286.3	0.9	4598.9	15.2
I - $LiAlO_2/H_2O$/HT-9/Be	110.4	885.4			188.9	427.2	58.3	93.9	166.8	118.8	575.0	409.7
Tokamak												
A - $LiAlO_2$/NS/HT-9/Be	48.0	384.8			252.5	558.0	79.4	102.9	60.2	111.4	644.4	1192.5
D - Li/Li/V/-	59.8	357.6	227.7	1826.4	See Coolant				341.6	170.8	826.7	413.4
E - Li_2O/He/HT-9/-	102.9	825.1			190.6	518.5			247.3	0.8	4956.9	16.4
F - $LiAlO_2$/He/HT-9/Be	88.2	707.1			322.5	399.3	35.0	51.9	243.9	0.8	4578.2	15.1
G - Li/He/HT-9/-	106.6	855.2	127.2	1020.3	359.7	179.9			380.3	1.3	5217.9	17.2
H - Flibe/He/HT-9/Be	126.7	1016.0	167.5	538.8	39.6	79.2	86.0	159.0	267.1	0.9	5673.3	18.7
I - $LiAlO_2/H_2O$/HT-9/Be	112.0	897.9			168.2	380.5	73.0	117.5	152.7	108.8	500.0	356.2

Note: This table does not include limiter volumes or masses.
Volumes shown in Table are obtained from zone thicknesses times material fractions
Masses are obtained by multiplying volumes, % theoretical density, and theoretical density.

to 60 m^3 of coolant volume. It is also the lightest coolant mass with the exception of the helium coolant. The heaviest coolant is the lithium-lead coolant. The total coolant mass, including the primary coolant in the blanket headers, plenums, piping, tanks and heat exchangers or steam generators was computed for a BCSS activation calculation.

All blanket and shield costs are shown in Table 3.2-2. The first wall costs range around $1M for tandem mirror (TM) and $2M for the tokamak (TOK). The V-15Cr-5Ti first wall options C, LiPb/LiPb/V, and D, Li/Li/V, are higher in cost than their HT-9 contemporaries. Also, the first wall option I, $LiAlO_2/H_2O$/HT-9/Be, which is water-cooled, has a higher cost due to a thicker wall and a higher metal volume fraction. In the structure, again the vanadium-based options are the more costly, being at least a factor of four times higher than the HT-9 based structures. The lightest and cheapest HT-9 structures are the nitrate salt options (A) in the TM and TOK, reflecting their smaller volumes shown in Table 3.2-1. The Flibe design had the heaviest HT-9 structure. The spread of the HT-9 structures is about a factor of 3 for the TM designs and a factor of 2 for the TOK designs.

The breeder costs show more variation due to the diversity of the forms (spherepac, clad sintered plates and liquids) and a range of enrichment (natural to 90% enriched). The solid breeders used the most volume and were the heaviest while Flibe had the least volume, mass, and cost. The breeder costs range from $3M to $81M. The Be neutron multipliers, if required, have costs which range from $20M to $70M, these being influenced both by the density and form (spherepac, rods, and balls). The Li, LiPb, and He-cooled Li designs use HT-9 reflectors. The He-cooled Li used approximately one-half the reflector mass (and cost) of the lithium or lithium-lead designs. The lighter weight SiC reflector for the Flibe design is approximately the same cost as the HT-9 in the He-cooled lithium design.

Not all designs required a separate plenum region. The nitrate salt concept designated 20 percent of the outermost region of the blanket as a plenum region and so was only a minor cost item. The lithium designs had no plenum regions defined. The nitrate salt is a relatively inexpensive coolant. Water and helium are low enough in cost not to be included in the analysis.

TABLE 3.2-2
BLANKET AND SHIELD COST ELEMENTS
(M$)

CODE	FW	STR	BRDR	MLTPR	RFLR	PLNM	CLNT	SHLD	TOTAL
Tandem Mirror									
A - $LiAlO_2$/NS/HT-9/Be	0.89	9.69	37.69	33.85		0.63	0.13	100.21	183.09
B - Li/Li/HT-9/-	0.36	13.41	5.98		24.59			84.60	128.94
C - LiPb/LiPb/V/-	2.29	104.64	20.89		29.83			79.19	236.85
D - Li/Li/V/-	2.29	86.20	5.98		24.59			84.72	203.76
E - Li_2O/He/HT-9/-	1.19	22.35	19.74			5.87		104.67	153.82
F - $LiAlO_2$/He/HT-9/Be	1.19	16.88	39.17	21.21		5.39		113.19	197.04
G - Li/He/HT-9/-	1.19	26.[illegible]0	9.52		13.83	7.77		122.14	180.94
H - Flibe/He/HT-9/Be	1.01	31.71	3.05	55.71	11.18	18.98		100.73	222.46
I - $LiAlO_2$/H_2O/HT-9/Be	1.93	13.76	81.18	41.31		1[illegible].20		91.64	244.02
Tokamak									
A - $LiAlO_2$/NS/HT-9/Be	2.66	11.56	42.41	45.26		0.69	0.17	100.39	203.12
D - Li/Li/V/-	11.80	77.59	6.83		21.92			96.99	215.13
E - Li_2O/He/HT-9/-	2.28	22.97	20.74			11.20		102.07	159.26
F - $LiAlO_2$/He/HT-9/Be	2.28	17.86	39.93	22.82		10.65		110.53	204.06
G - Li/He/HT-9/-	2.22	20.44	7.19		12.24	12.57		112.04	166.71
H - Flibe/He/HT-9/Be	2.22	26.97	2.93	69.98	9.70	15.12		99.74	226.67
I - $LiAlO_2$/H_2O/HT-9/Be	4.56	15.21	72.30	51.69		11.80		94.91	250.46

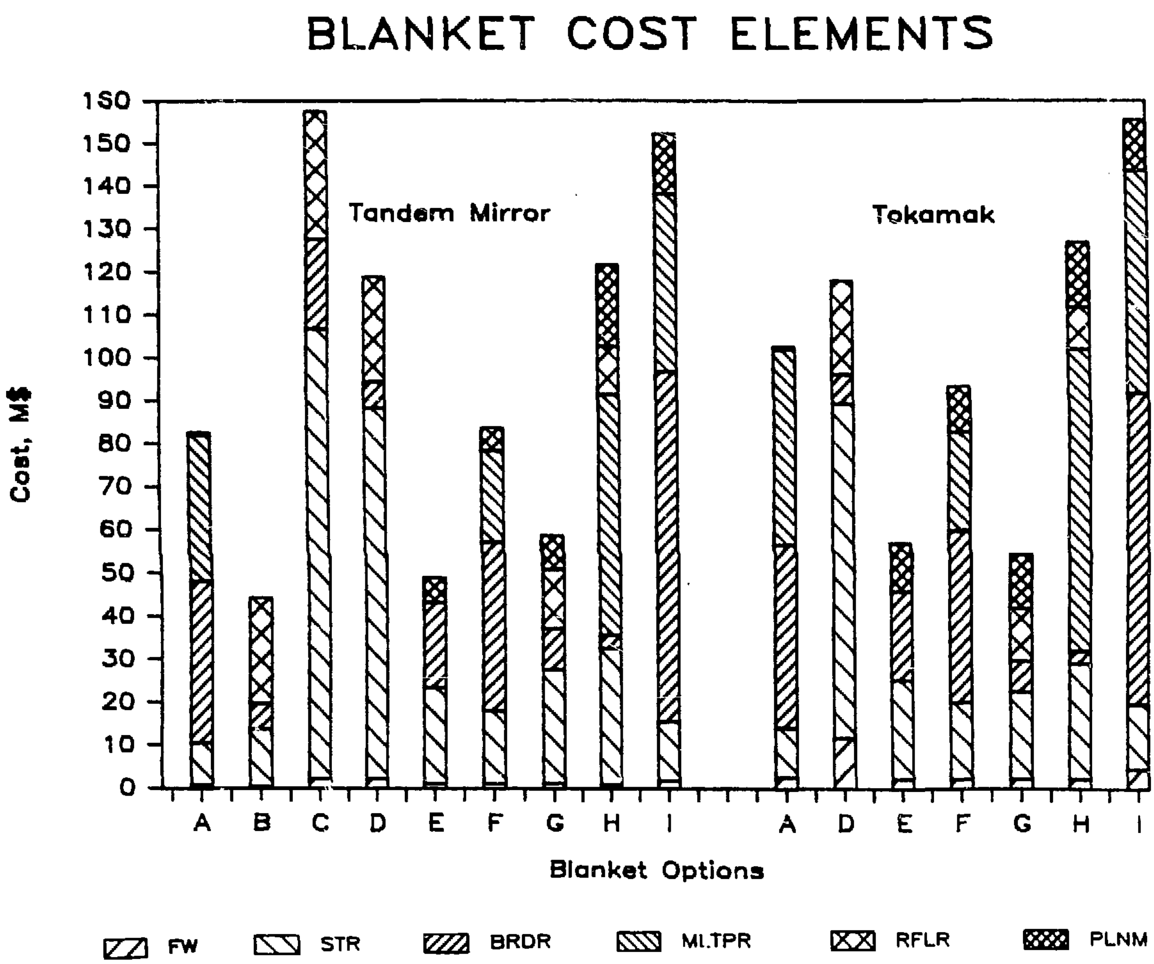
FW = First Wall
RFLR = Reflector
STR = Structure
PLNM = Plenum
BRDR = Breeder
CLNT = Total Coolant
MLTPR = Multiplier
SHLD = Shield

The shielding requirements were determined by a 1-D neutronics analysis. Composition and thicknesses were determined for each blanket concept. Then using the reactor configuration, the volumes, weights and costs were determined. The variation of the shielding immediately surrounding the blanket ranged from $79M to $122M for the TM and $94M to $112M for the TOK.

The sum of these cost elements are totaled in the last column on Table 3.2-2. In summary, the cost drivers for the blankets were the breeder enrichment, the beryllium multiplier, the vanadium structure, and the use of a massive reflector. The lowest total cost TM design is $128M for the Li/HT-9 and the highest is the $244M H_2O-cooled solid breeder. In the tokamak designs, the helium-cooled, solid Li_2O was the cheapest at $159M, followed closely by the helium-cooled, liquid lithium breeder at $166M. A Li/Li/HT-9 tokamak design might have been the cheapest blanket and shield but this design was not proposed. The water-cooled design was again the highest cost blanket and shield. Although the blanket and shield costs for the tokamak were somewhat higher as a group, there was more variation in the tandem mirror costs. Figure 3.2-1 graphically displays the various cost elements of the blanket portion.

Power Conversion and Transport - The blanket of a fusion reactor has only two functions: power conversion and tritium production. This subsection will discuss the power performance of the various blanket options and the system implications inherent in the various design decisions. Sec. 5.3.2 discusses the economic groundrules of this study, including the determination of the reactor power flows. Briefly, the fusion power for the tandem mirror is held constant at a value to produce a neutron wall loading of 5 MW/m^2 on a first wall surface identical to MARS. Variations in the thicknesses of the blanket and shield will only influence the magnet costs. Table 3.2-3 presents the tandem mirror power balance for the considered blanket options.

Each blanket has a blanket energy multiplication factor (EMF) determined by the BCSS Neutronics Task, based on the particular material composition. This value represents the useful energy recoverable from the blanket. Table 3.2-3 presents a complete listing of these data from the tandem mirror designs. The highest blanket EMF was 1.549 for the Flibe design (H) and the lowest was 1.228 for the Li_2O design (E). On Table 3.2-3, blanket output



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
I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-1

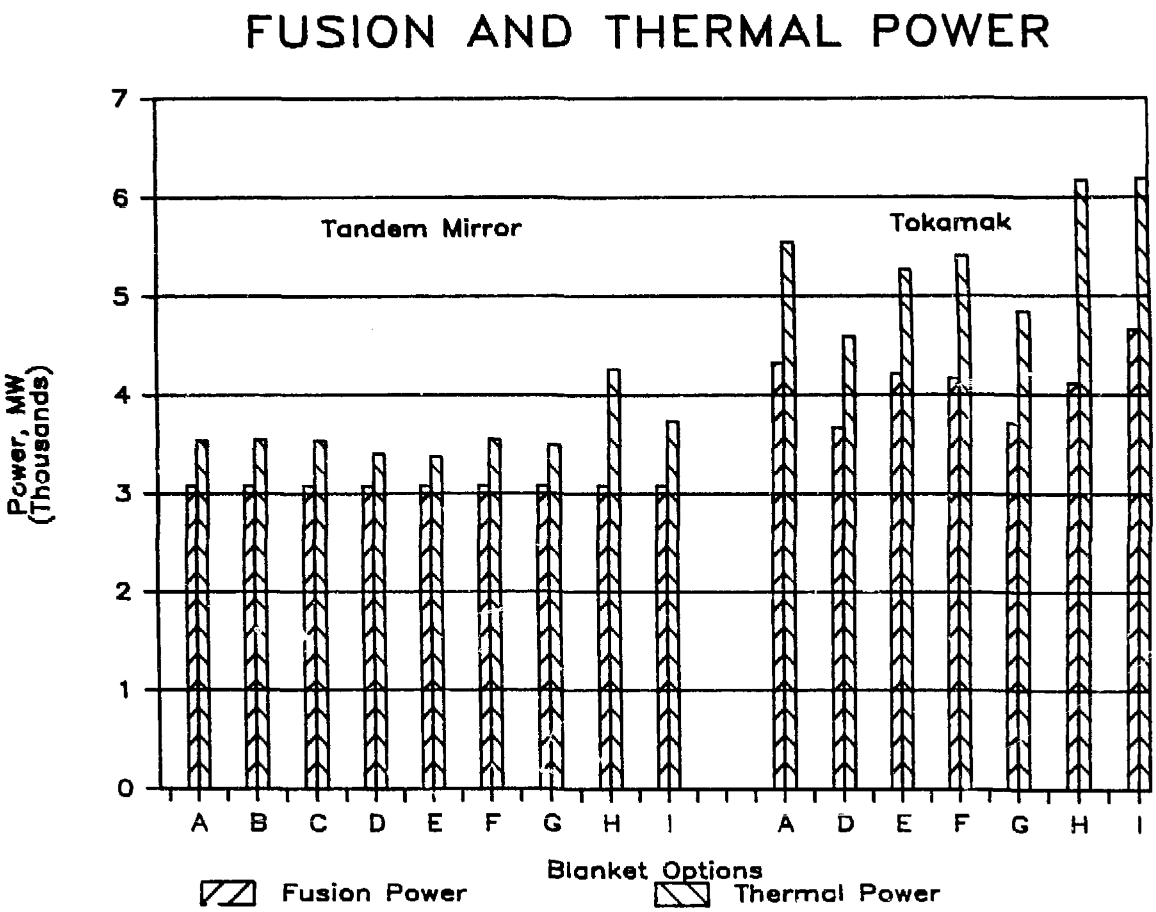
TABLE 3.2-3
POWER BALANCE DATA

Tandem Mirror

Blanket Option	A	B	C	D	E	F	G	H	I
BDR/CLT/STR/MULTR	$LiAlO_2$/ NS/HT-9/ Be	Li/Li/ HT-9/-	LiPb/ LiPb/V/-	Li/Li/ V/-	Li_2O/He/ HT-9/-	$LiAlO_2$/ He/HT-9/ Be	Li/He/ HT-9/-	Flibe/ He/HT-9/ Be	$LiAlO_2$/ H_2O/HT-9/ Be
Fusion Power, MW	3082.8	3082.8	3082.8	3082.8	3082.8	3082.8	3082.8	3082.8	3082.8
Neutron Power, MW	2465.0	2465.0	2465.0	2465.0	2465.0	2465.0	2465.0	2465.0	2465.0
Neutron Wall Load, MW/m^2	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0	5.0
Energy Multiplication Factor	1.316	1.313	1.294	1.254	1.228	1.291	1.270	1.549	1.386
Blanket Output, MW	3243.9	3236.5	3189.7	3091.1	3027.0	3182.3	3130.6	3818.3	3416.5
Alpha Power, MW	617.3	617.3	617.3	617.3	617.3	617.3	617.3	617.3	617.3
Heating Power to Plasma, MW	122.7	122.7	122.7	122.7	122.7	122.7	122.7	122.7	122.7
Power to Direct Convertor, MW	493.8	493.8	493.8	493.8	493.8	493.8	493.8	493.8	493.8
Electric Power from DC, MWe	317.5	317.5	317.5	317.5	317.5	317.5	317.5	317.5	317.5
Thermal Power from DC, MW	176.3	176.3	176.3	176.3	176.3	176.3	176.3	176.3	176.3
Halo Thermal Power, MW	123.5	123.5	123.5	123.5	123.5	123.5	123.5	123.5	123.5
Power to Low Temperature Turbine, MW	299.7	299.7	299.7	299.7	299.7	299.7	299.7	299.7	299.7
Pump Power Addition, MW	5.6	18.0	34.5	13.0	55.0	74.7	73.5	147.7	24.8
Total Thermal Power to Turbines, MW	3549.2	3554.3	3524.0	3403.8	3381.8	3556.8	3503.8	4265.7	3741.0
Gross Thermal Efficiency	.406	.405	.420	.423	.400	.400	.400	.389	.357
Gross Turbine Elect Power, MWe	1402.6	1420.9	1457.0	1415.8	1335.6	1405.6	1384.4	1645.6	1331.3
Total Gross Electric Power, MWe	1720.1	1738.4	1774.5	1733.4	1653.1	1723.2	1702.0	1963.1	1648.9
Pumping Power, MWe	6.2	20.3	38.4	14.7	64.7	87.9	86.5	173.7	27.5
Heating Power, MWe	193.0	193.0	193.0	193.0	193.0	193.0	193.0	193.0	193.0
Other Auxiliary Power, MWe	180.0	180.0	180.0	180.0	180.0	180.0	180.0	180.0	180.0
Net Electric Power, MWe	1340.0	1345.0	1363.0	1345.0	1215.0	1262.0	1242.0	1416.0	1248.0

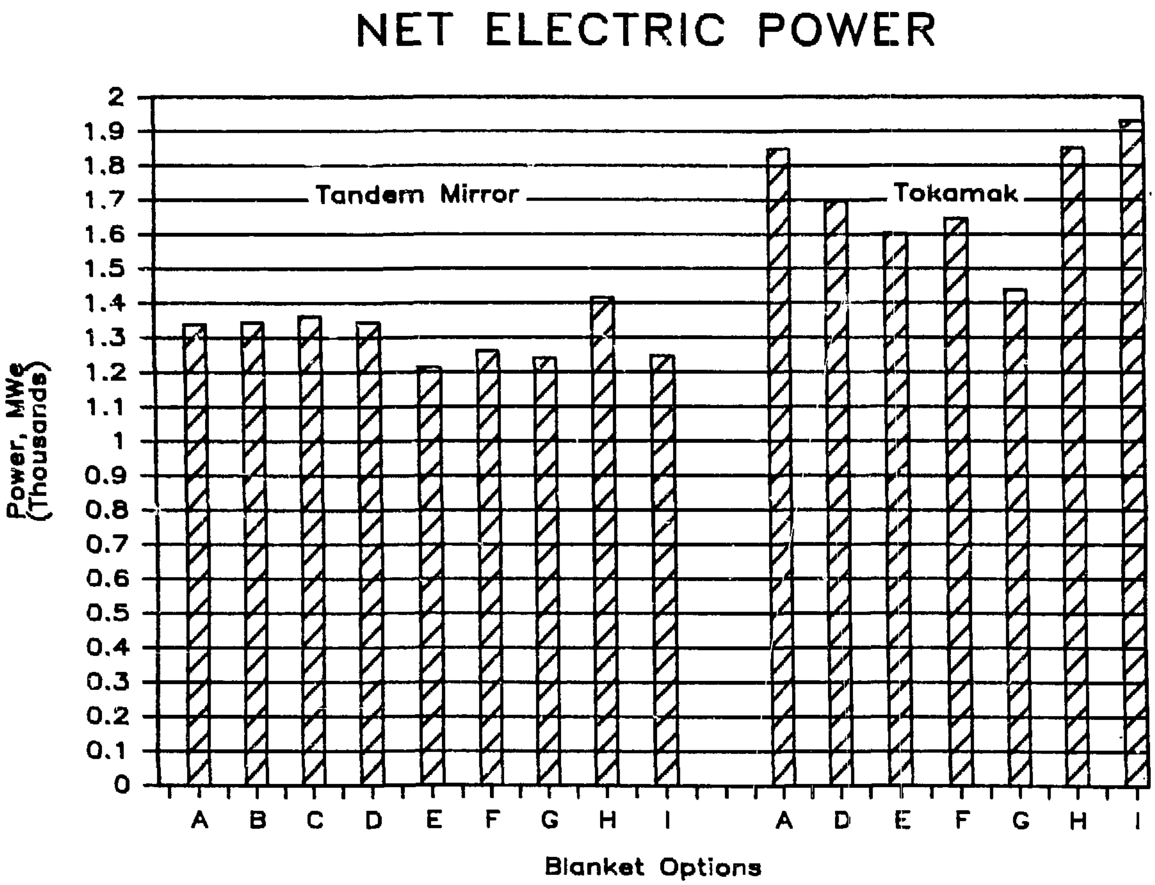
power is the product of the blanket EMF and the neutron power. The blanket EMF caused a range of blanket output power from a high of 3818 MW for the Flibe design with a high percentage of beryllium to the lowest power output of 3027 MW for the Li_2O design. Sixty-seven percent of the sum of the alpha power and the heating power is converted to electricity by the direct convertor. The remaining plasma power loss is collected by the halo scraper and is combined with the direct convertor reject heat and is then sent to a moderate temperature thermal energy conversion system. The electricity produced from the direct convertor and the moderate thermal conversion system will be added to the primary thermal conversion system. Meanwhile, the blanket thermal output is collected by the variety of primary coolants, each with its own set of operating conditions. Tables 5.3-8, Primary Coolant Data and 5.3-20, BCSS Thermal-Hydraulics Data, summarize those data. Since the pumps input pump work into the system, their contributions are additive to the overall turbine thermal power as shown in Table 3.2-3. Again the Flibe design has the highest power at 4266 MW and the Li_2O is the lowest at 3382 MW. Figure 3.2-2 also illustrates these thermal power results. Flibe is clearly producing more thermal power than any other design.

Within the limits of structural and coolant temperatures, the thermal conversion efficiencies of each TM system are shown in Table 3.2-3. The lithium coolants required an intermediate sodium coolant loop which lowered their efficiencies somewhat. The Flibe design still remained on top with the highest gross turbine electrical power. Conversely, the lower thermal efficiency of the pressurized water coolant cuased it to displace the Li_2O as the lowest electric production system. These electric powers are then added to the other electrical sources to produce a total gross value. The next step was to subtract off the power required to pump the primary coolant and the intermediate coolant, if required, and to provide plasma heating and other auxiliary power. The last line in Table 3.2-3 reflects the net electric power values. Figure 3.2-3 graphically presents these data. The Flibe design ended up with the highest net electric power (NEP) production, but the margin is less than it was in the comparison of the thermal power. On the basis of the TM NEP, Flibe is the leader, followed by the nitrate salt and the liquid lithium and lithium-lead designs with the helium- and water-cooled designs coming in with the least power.



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
I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-2



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

I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-3

In the tokamak reactor design, the thickness of the blankets and shields influence not only the costs of the magnets, but also determine the magnetic field on axis and in turn, the plasma fusion power. See Sec. 5.3.2 for a more complete discussion on this subject including the methodology and input design parameters. Table 3.2-4 presents the tokamak power balance. The combined thicknesses of the inner blanket and shield modify the radius of the peak magnetic field, the field on the plasma axis, and the fusion power. The H_2O-cooled design (I) had the thinnest blanket and shield and thus had the most fusion power. Flibe again had the highest energy multiplication, but could not displace the H_2O-cooled design for the maximum thermal power (which includes the alpha power contribution).

The heating and pump power addition contributed to the overall thermal power sent to the turbines. As in the TM cases, lithium still requires an intermediate sodium coolant loop. At the turbine input, Flibe and the H_2O designs are virturally identical in the total thermal power. The lithium-cooled (D) concept is the lowest thermal power option. Figure 3.2-2 compares the thermal power rankings for all the concepts.

When the thermal efficiency is considered, the lithium-cooled option improves. The lowest gross electrical power was produced by the helium-cooled, lithium breeder option. Since the Flibe design had better thermal efficiency, it has the highest gross power production option. This advantage for the Flibe is short-lived when the pumping power is subtracted for the overall net electric power production. Figure 3.2-3 clearly shows the H_2O-cooled design producing the most net electric power with the nitrate salt and Flibe tied for second place. The helium-cooled, lithium-breeder produces the least net electric power.

The costs for the heat transfer and transport (HTT) system are significantly influenced by the thermal power levels handled and the primary and intermediate coolants involved. Detailed assumptions and groundrules are discussed in Sec. 5.3.2. Using those groundrules and the aforementioned power data, the costs for the HTT system were developed and are shown in Table 3.2-5. The higher piping cost for the lithium-lead blanket concept was caused by the high density of the material whereas the higher Flibe cost was attributable to the larger power levels handled. Pumps were a high cost item except for the nitrate salt and H_2O design options. The H_2O-cooled design was the

TABLE 3.2-4
POWER BALANCE DATA

Tokamak							
Blanket Option	A	D	E	F	G	H	I
BDR/CLT/STR/MULTR	$LiAlO_2$/ NS/HT-9/ Be	Li/Li/ V/-	Li_2O/He/ HTR-9/-	$LiAlO_2$/ He/HT-9/ Be	Li/He/ HT-9/-	Flibe/ He/HT-9/ Be	$LiAlO_2$/ H_2O/HT-9/ Be
Peak Field Radius, m	3.553	3.410	3.530	3.520	3.420	3.510	3.620
Field on Plasma Axis, T	5.076	4.871	5.043	5.029	4.886	5.014	5.[illegible]71
Fusion Power, MW	4331.3	3675.0	4220.3	4172.6	3718.3	4125.4	4667.4
Neutron Wall Load, MW/m^2	4.4	3.7	4.2	4.2	3.8	4.2	4.7
Energy Multiplication Factor	1.323	1.272	1.223	1.280	1.279	1.511	1.372
Thermal Power, MW	5450.2	4474.5	4972.9	5107.0	4548.0	5811.4	6056.0
Heating Power, Ave, MWe	90.0	90.0	90.0	90.0	90.0	90.0	90.0
Pump Power Addition, MW	9.5	32.0	206.9	215.9	211.8	272.1	44.6
Total Thermal Power to Turbines, MW	5549.7	4596.4	5269.8	5413.0	4849.8	6173.6	6190.6
Gross Thermal Efficiency	.375	.423	.392	.392	.392	.389	.357
Total Gross Electric Power, MWe	2081.1	1944.3	2065.8	2121.9	1901.1	2401.5	2210.0
Pumping Power, MWe	10.5	35.6	243.4	254.0	249.1	320.2	49.5
Heating Power, MWe	150.0	150.0	150.0	150.0	150.0	150.0	150.0
Other Auxiliary Power, MWe	72.1	59.8	68.5	70.4	63.0	80.3	80.5
Net Electric Power, MWe	1848.0	1698.0	1603.0	1647.0	1438.0	1851.0	1930.0

TABLE 3.2-5
HEAT TRANSFER AND TRANSPORT POWER AND COSTS

CODE	T PWR (MW)	FLUID	PIPING+ (M$)	PUMPS (M$)	DT/P (M$)	CCMU (M$)	IHX/SG (M$)	TS (M$)	TOTAL (M$)	ICS (M$)	LCS (M$)	RHMS (M$)	TOTAL (M$)
Tandem Mirror													
A - $LiAlO_2$/NS/HT-9/Be	3549	NS	48	3	1	5	39	NA	95	0	0	1	96
B - Li/Li/HT-9/-	3554	Li	64	22	2	36	60	NA	183	130	0	1	314
C - LiPb/LiPb/V/-	3542	LiPb	132	27	2	35	70	NA	266	0	0	1	267
D - Li/Li/V/-	3404	Li	62	21	1	34	58	NA	177	125	0	1	302
E - Li_2O/He/HT-9/-	3382	He	84	30	0	6	95	NA	215	0	0	1	216
F - $LiAlO_2$/He/HT-9/Be	3557	He	87	31	0	6	100	NA	224	0	0	1	226
G - Li/He/HT-9/-	3504	He	87	31	0	6	98	NA	221	0	0	1	222
H - Flibe/He/HT-9/Be	4266	He	101	36	0	7	119	NA	263	0	0	1	264
I - $LiAlO_2$/H_2O/HT-9/Be	3741	H20	51	3	7	5	41	NA	107	0	0	1	108
Tokamak													
A - $LiAlO_2$/NS/HT-9/Be	5550	NS	52	4	1	7	61	8	133	0	8	1	141
D - Li/Li/V/-	4596	Li	59	26	2	42	78	11	219	162	19	1	400
E - Li_2O/He/HT-9/-	5270	He	90	41	0	8	148	31	317	0	8	1	326
F - $LiAlO_2$/He/HT-9/Be	5413	He	91	42	0	9	152	32	325	0	8	1	334
G - Li/He/HT-9/-	4850	He	83	38	0	8	136	29	293	0	19	1	313
H - Flibe/He/HT-9/Be	6174	He	102	47	0	10	173	35	367	0	8	1	376
I - $LiAlO_2$/H_2O/HT-9/Be	6191	H20	58	4	11	8	68	7	156	0	8	1	165

T PWR = Thermal Power to the Turbines
PIPING+ = Piping, Manifolds, Elbows & Valves
DT/P = Dump Tanks or Pressurizers
CCMU = Coolant Cleanup & Makeup
IHX/SG = Intermediate Heat Exchanger or Steam Generator

TS = Thermal Storage
ICS = Intermediate Coolant Sys
(Steam Generator, Piping, Pumps, Etc)
LCS = Limiter Coolant Sys
RHMS = Residual Heat Removal Sys

only system that required an expensive pressurizer. the remainder of the costs in this column were associated with dump tanks. The coolant cleanup and makeup were most costly for the lithium and lithium-lead systems.

The next column illustrates either the cost of a steam generator or, in the case of a lithium coolant, the cost of an intermediate heat exchanger. The H_2O and the nitrate salts cases have the lower cost steam generators. Thermal storage is required only by the pulsed tokamak systems. These costs are influenced by the power level and the coolant medium.

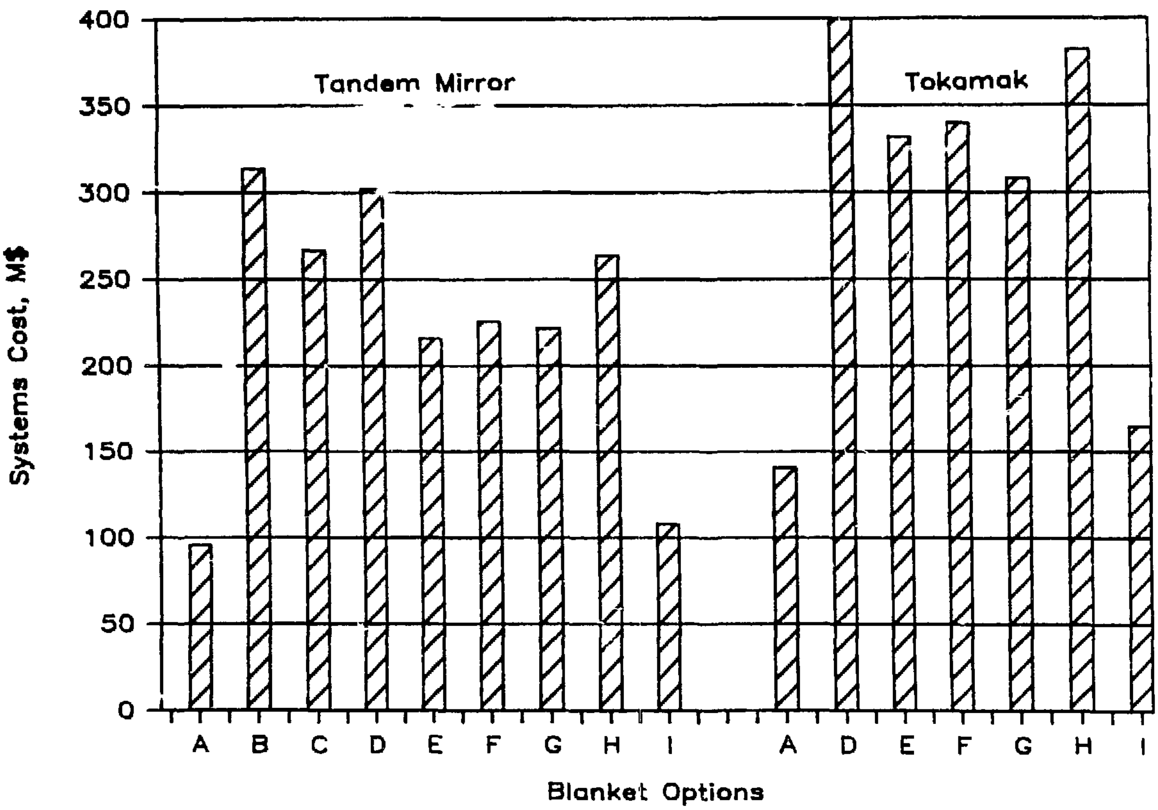
The values shown for the intermediate coolant system includes costs for the steam generator, piping, pumps, and all other required hardware. The limiter coolant system applied only to the tokamak systems and are either water- or lithium-cooled structures. The structure is either HT-9 or V-15Cr-5Ti specifically for the lithium-cooled blanket. The overall HTT system costs are tabulated in Table 3.2-5 and shown graphically on Fig. 3.2-4. The figure illustrates clearly that there is a large range of costs associated with the HTT system. The lithium and the Flibe designs are the most expensive and the nitrate salt and water are the least expensive.

Major Reactor Plant Costs - The structures and site facilities costs are constant except for modifications to the cost of the reactor building to accommodate the requirements of the coolants. Specifically, the overpressure requirements for the water coolant and a steel flooring for the liquid lithium, lithium-lead, nitrate salt and the Flibe coolants or breeder. These data are shown in Table 3.2-6.

The first wall, blanket and shield costs have already been discussed in previous paragraphs. The magnets are determined by the thicknesses of the blanket and shield. These magnet costs include all magnets including TM end cells and tokamak poloidal field coils. The TM magnets have a higher degree of cost variation than do the TOK magnets.

The limiter subsystem applies only to the tokamak designs. The only differences are a lithium-cooled limiter for the lithium-cooled and the lithium-breeder blankets or a water-cooled option for all other concepts. The HTT system costs have previously been discussed.

HEAT TRANSFER SYSTEM COST



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

I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-4

TABLE 3.2-6
MAJOR REACTOR PLANT COSTS
(M$)

CODE	SSF	FWB	SHLD	MAG	LMTR	HTT	FHS	TPE	EPE	S.MTLS	DCC	T CAP	$KWe
Tandem Mirror													
A - $LiAlO_2$/NS/HT-9/Be	256	83	123	562	NA	96	202	299	172	1	2452	4542	3390
B - Li/Li/HT-9/-	256	44	108	504	NA	314	197	301	176	9	2586	4790	3561
C - LiPb/LiPb/V/-	256	158	102	506	NA	267	213	394	181	44	2723	5044	3701
D - Li/Li/V/-	256	119	108	503	NA	302	207	298	174	9	2665	4937	3670
E - Li_2O/He/HT-9/-	250	49	128	493	NA	216	195	289	189	1	2472	4579	3769
F - $LiAlO_2$/He/HT-9/Be	250	84	136	492	NA	225	207	299	196	1	2564	4749	3763
G - Li/He/HT-9/-	256	59	145	564	NA	222	196	296	196	10	2625	4862	3914
H - Flibe/He/HT-9/Be	256	122	124	509	NA	264	235	335	222	4	2764	5120	3616
I - $LiAlO_2$/H_2O/HT-9/Be	262	152	115	479	NA	108	162	294	170	1	2403	4450	3566
Tokamak													
A - $LiAlO_2$/NS/HT-9/Be	382	103	195	297	4	141	211	393	142	1	2731	5059	2737
D - Li/Li/V/-	382	118	192	304	15	400	167	365	147	10	2980	5520	3251
E - Li_2O/He/HT-9/-	376	57	197	304	4	326	211	387	230	1	2974	5509	3437
F - $LiAlO_2$/He/HT-9/Be	376	94	205	304	4	334	227	394	235	1	3071	5688	3453
G - Li/He/HT-9/-	382	55	207	319	15	313	161	365	231	7	2919	5407	3760
H - Flibe/He/HT-9/Be	382	127	195	304	4	376	262	431	264	3	3281	6076	3283
I - $LiAlO_2$/H_2O/HT-9/Be	389	156	190	295	4	165	173	414	160	1	2830	5242	2716

SSF = Structures & Site Facilities
FWB = First Wall & Blanket
SHLD = Shield
MAG = Magnets
LMTR = Limiter
HTT = Heat Transfer & Transport

FHS = Fuel Handling and Storage
TPE = Turbine Plant Equipment
EPE = Electric Plant Equipment
S MTLS = Special Materials
DCC = Direct Capital Cost
T CAP = Total Capital

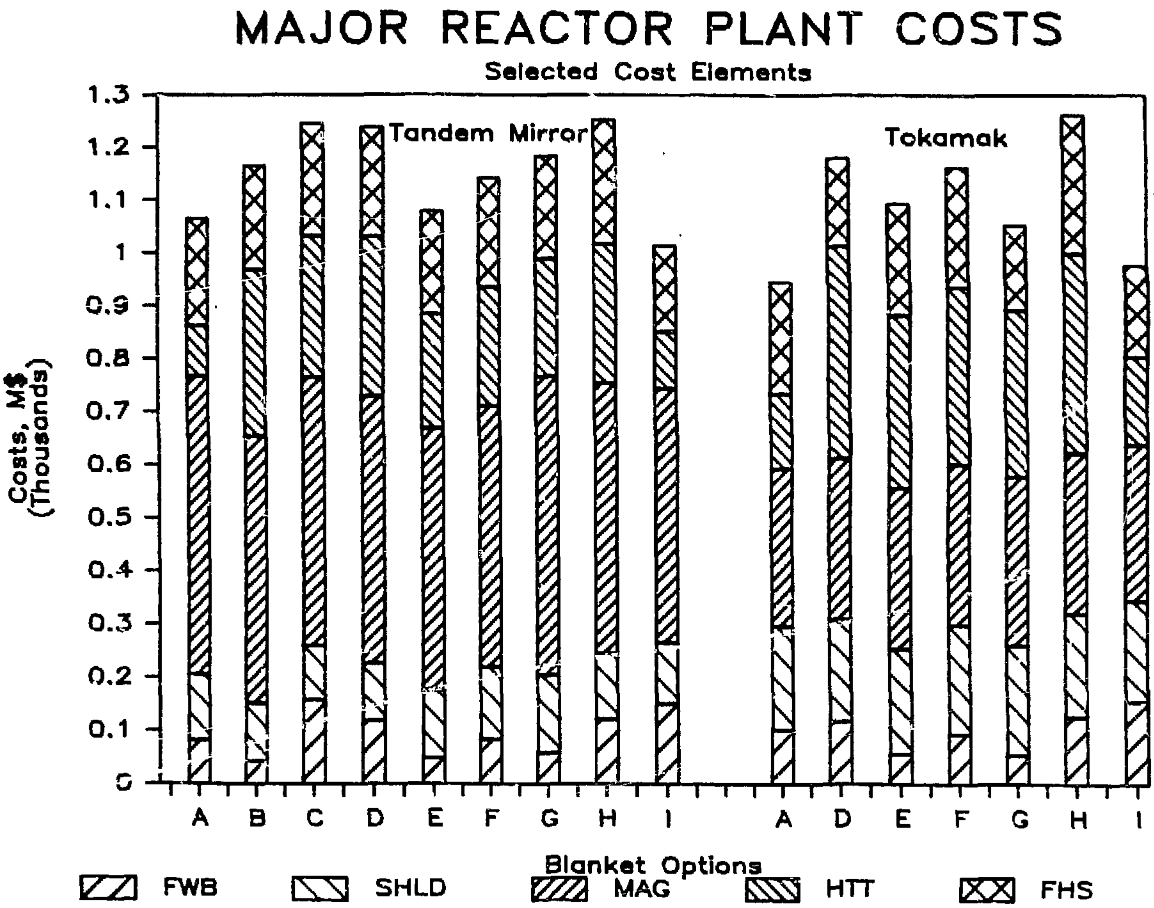
The fuel handling and storage (FHS) system costs are discussed in Sec. 5.3.3. The costs are the highest for the Flibe-cooled designs because of the high partial pressure of T_2. The lithium-cooled blanket option has the lowest FHS system costs. Figure 3.2-5 illustrates the Fuel System costs along with other Major Reactor Plant Cost factors.

The Turbine Plane Equipment and the Electric Plant Equipment costs shown in Table 3.2.1-6 are determined by the thermal power, electric power and the recirculating power requirements. Flibe is the highest cost option in this area but there is no overall general lowest cost option. Special Materials are low cost except for large quantities of the lithium-lead coolant and lithium for the coolant or breeder.

The next two columns in Table 3.2-6 compare the Direct Capital Costs and the Total Capital Costs. Figure 3.2-6 is also a comparison of Total Capital Costs. The lowest capital costs are the water-cooled TMR and the nitrate salt-cooled tokamak. The Flibe blanket concept is the most costly in both reactor types although by a lesser margin in the tandem mirror. The differences in the costs between the whole class of TM and TOK are representative of many factors specific to this analysis including the power output scaling of the plants. Again, it is important to stress that this study was structured to compare blanket concepts when employed in the two reactor types and not to compare the reactors themselves. The technical and economic bases and ground-rules implicit in the reactor definition differ significantly and any cross comparisons between tokamaks and tandem mirrors will lead to erroneous conclusions. The final column in Table 3.2-6 and Fig. 3.2-7 illustrate the normalized Cost of Capacity in terms of dollars/kWe. The TM reactors are clustered closer together around the $3500/kWe whereas the tokamak designs show more variation. The nitrate salt and H_2O designs are the lower cost designs and the helium-cooled, lithium-breeder is the highest cost option. The Flibe design is much improved when compared on this basis.

3.2.2 Overall Economic Evaluation

This subsection combines the previously discussed economic factors together in the economic evaluation parameter of Cost of Electricity. Table 3.2-7 lists these major factors. The Net Electrical Power and Total Capital Cost of the reactor plants are presented for all the blanket concepts. The



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

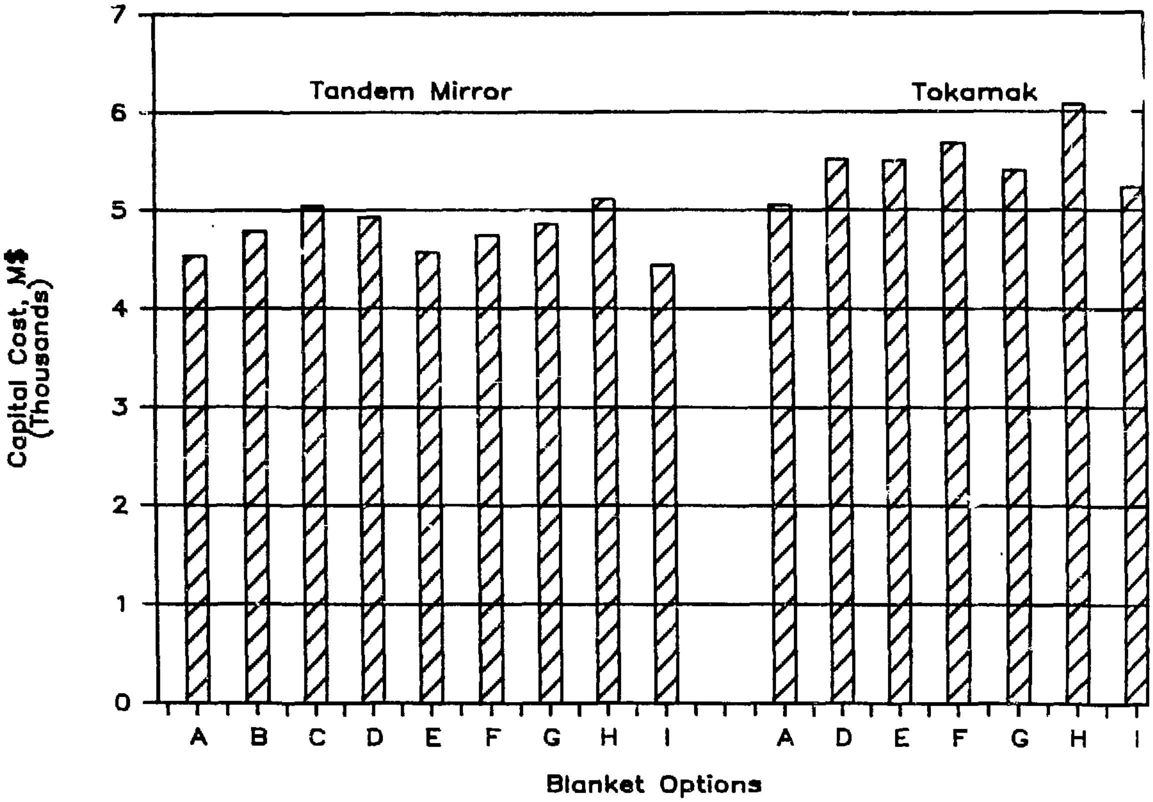
G – Li/He/HT-9

H – FLIBE/He/HT-9/Be

I – $LiAlO_2$/H_2O/HT-9/Be

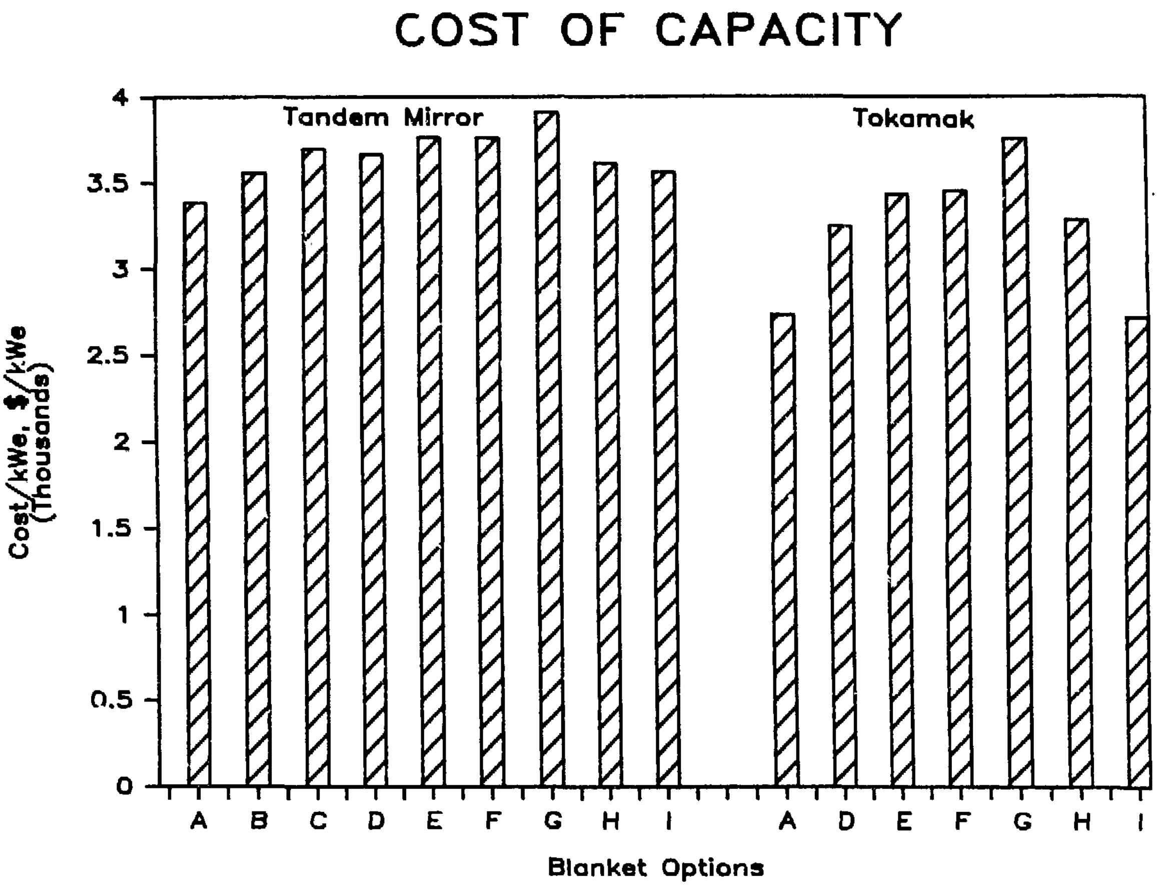
Figure 3.2-5

TOTAL CAPITAL COST



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
I – $LiAlO_2$ / H_2O / HT-9 / Be

Figure 3.2-6



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

I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-7

TABLE 3.2-7
OVERALL ECONOMIC EVALUATION

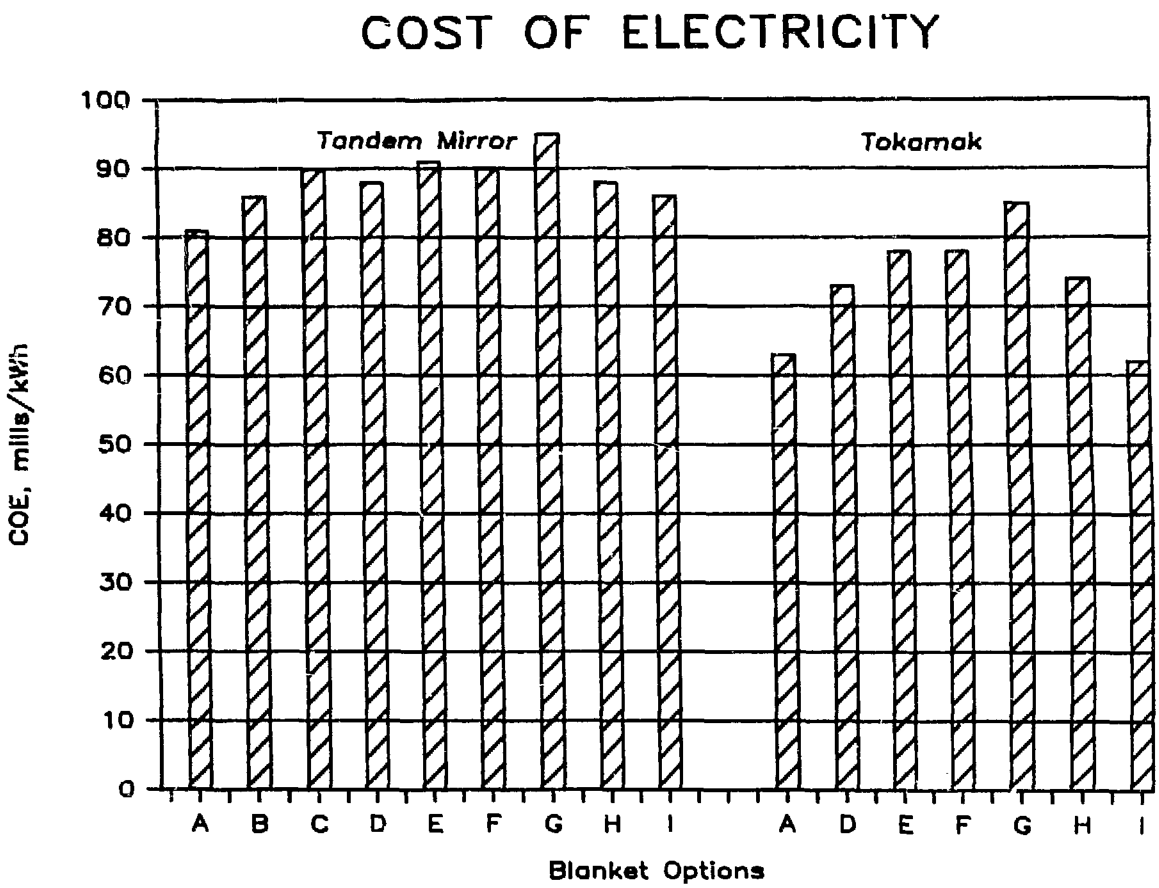
CODE	NEP (MW)	TCC (M$)	CAP (M$/Y)	O&M (M$/Y)	SCR (M$/Y)	FUEL (M$/Y)	TOTAL (M$/Y)	COE (Mills/kWh)	BCSS RANKING Lowest COE/COE
					Tandem Mirror				
A - $LiAlO_2$/NS/HT-9/Be	1340	4542	681.3	65.7	16.7	0.5	764.2	81	1.00
B - Li/Li/HT-9/-	1345	4790	718.5	69.3	18.3	0.5	806.6	86	0.94
C - LiPb/LiPb/V/-	1363	5044	756.6	73.0	26.8	0.5	856.8	90	0.90
D - Li/Li/V/-	1345	4937	740.5	71.4	13.5	0.5	825.9	88	0.92
E - Li_2O/He/HT-9/-	1215	4579	686.9	66.3	20.5	0.5	774.1	91	0.89
F - $LiAlO_2$/He/HT-9/Be	1262	4749	712.3	68.7	13.2	0.5	794.7	90	0.90
G - Li/He/HT-9/-	1242	4862	729.3	70.3	23.0	0.5	823.1	95	0.85
H - Flibe/He/HT-9/Be	1416	5120	768.0	74.1	26.5	0.5	869.0	88	0.92
I - $LiAlO_2$/H_2O/HT-9/Be	1248	4450	667.6	64.4	20.4	0.5	752.8	86	0.94
					Tokamak				
A - $LiAlO_2$/NS/HT-9/Be	1848	5059	758.8	31.9	20.0	0.5	811.2	63	0.98
D - Li/Li/V/-	1698	5520	828.0	31.9	12.5	0.5	872.9	73	0.85
E - Li_2O/He/HT-9/-	1603	5509	826.4	31.9	16.3	0.5	875.2	78	0.79
F - $LiAlO_2$/He/HT-9/Be	1647	5688	853.2	31.9	15.1	0.5	900.7	78	0.79
G - Li/He/HT-9/-	1438	5407	811.1	31.9	15.7	0.5	859.3	85	0.73
H - Flibe/He/HT-9/Be	1851	6076	911.5	31.9	16.2	0.5	960.1	74	0.84
I - $LiAlO_2$/H_2O/HT-9/Be	1930	5242	786.3	31.9	22.1	0.5	840.8	62	1.00

NEP = Net Electric Power
TCC = Total Capital Cost
CAP = Capital Cost (Annual)
O&M = Operations & Maintenance
SCR = Scheduled Component Replacement

maximum net power output is usually the nitrate salt-, Flibe-, or the H_2O-cooled blanket options. To compare these blanket options on a consistent basis, the annual cost for each were developed. Subsection 5.3.1 presents the methodology for the development of the annual costs. The Total Capital Costs are converted to an annual basis by application of a fixed charge rate. The O&M costs for the TMR were determined using the relationship defined in the MARS study, namely 2% of the Total Direct Costs escalated to the initial operating year. The STARFIRE O&M costs were fixed at \$31.9M. The Scheduled Component Replacement Costs considered the life expectancy of the blanket. Some blankets require more expensive parts and others require more costly remote maintenance operations. The groundrules for replacement and refurbishment costs are discussed in Subsection 5.3.2. The Fuel Cost is assumed to cost a fixed \$0.5M for all designs. The Total Annual Costs are presented in Table 3.2-7. These Annual Costs are combined with the Net Electrical Power production and other constants to arrive at the Cost of Electricity shown in the table and in Fig. 3.2-8. The final column in Table 3.2-7 is a relative ranking of the COE. The blanket concept with the lowest COE is rated 1.00 and all other options within a reactor type are compared to that concept on a relative basis. Figure 3.2-9 graphically presents these relative COE data as the economic figure of merit.

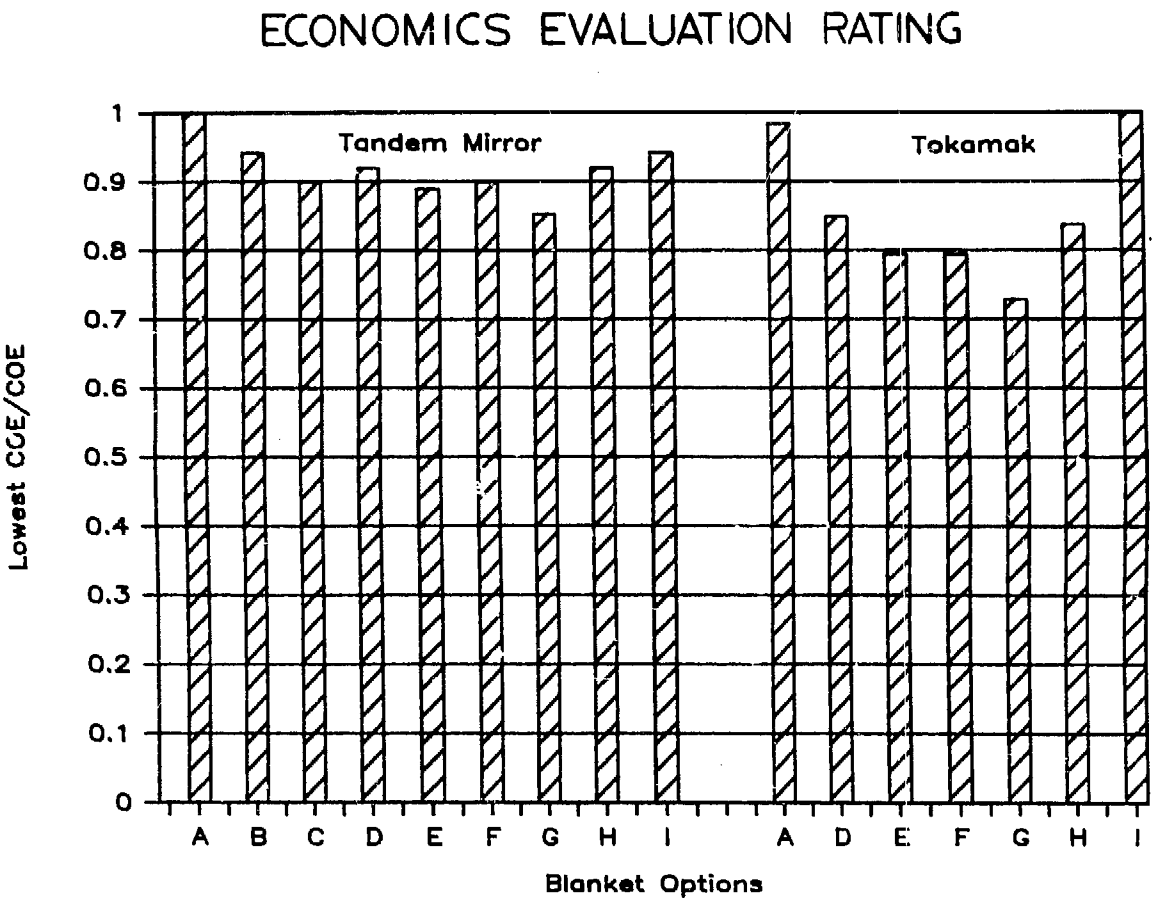
The nitrate salt is rated best in the tandem mirror and second best in the tokamak. The best blanket in the tokamak is the H_2O-cooled option. Again there is a narrow spread of economic evaluation data for the TMR. The tokamak options showed a much larger spread of evaluation data. In general, the water and the nitrate salt designs did best. The lithium and lithium-lead cooled designs were in the middle range with the helium designs faring least well in the economic area.

Again, it should be stressed that it is hard to evaluate the economics of an evolving technology. We can hope that these data will fairly represent trends and illustrate technical and economic strong and weak points about these generic blanket concepts. There may be a degree of uncertainty about the absolute values of the costs or performance, but it is reasonable to assume that the relative rankings are a true representation of the blanket's economic performance.



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
I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-8



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

I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-9

3.2.3 Economic Sensitivity Studies

The previous baseline economic evaluations were predicated upon nominal, most-likely design conditions, performance parameters, and cost estimates. Most of these influences had a narrow band of uncertainty which, in turn, had a minimal influence on the final economic evaluation outcome. However, several parameters in the evaluation were assessed to have an unresolvable high degree of uncertainty at this point in time. To assess the impact of these uncertainties, a set of sensitivity studies were conducted to evaluate the consequences. Some of these studies were in the area of the unit cost of materials and were applied on the complete spectrum of blanket concepts. These studies were applied to the tandem mirror as being representative of both tokamaks and tandem mirrors, in regard to these effects.

Economic sensitivities were also conducted to assess the economic benefits of enhancements in the blanket performance parameters. This would aid future investigations as to the relative impact of stressing performance in given areas. Areas investigated were:

- Blanket Energy Multiplication
- Blanket Lifetime
- Gross Thermal Efficiency
- Blanket/Shield Thickness
- Coolant Pumping Power

These parameters were assessed on a single blanket concept for both reactor types. The $LiAlO_2$/He/HT-9/Be concept was picked as the test concept for these investigations. It was usually somewhere in the middle of the range on most evaluation criteria. When the parameters are modified, the effect can be measured on how the concept improved on the relative ranking system. The effect may be interpolated or extrapolated to other values in a linear basis except for the tokamak Blanket/Shield Thickness which is a higher-order effect.

Material Unit Costs - In Section 5.3.2, Economic Groundrules, the baseline material unit costs used in the evaluation were defined based upon the most current information. The HT-9 structural and the shielding materials have some usage in other non-fusion applications and have a degree of historical basis to establish a reasonable cost estimate. The vanadium alloy, V-15Cr-5Ti, is an exception having little fabrication experience. Also the materials used in breeding and neutron multiplication have a limited data base. That data base is limited because of small usage of the materials in general and no usage of the materials in this particular fusion-related application. So the baseline cost estimates for these materials are formulated on the basis of an educated conjecture, considering similarity to other applications and/or other materials, likely learning-curve effects, projected new sources and anticipated process or fabrication techniques. In most cases considered in this materials sensitivity study, the more adverse assumptions were investigated, considering the impacts if the more pessimistic conditions prevailed. Usually this was assumed to be a projection of present market prices with current fabrication techniques.

The V-15Cr-5Ti structure for the baseline was estimated at $150/kg for the material and $100/kg for the fabrication, inspection, assembly, installation, and checkout. The material cost assumed a sizeable discount for large quantity purchase over a time period for a maturing economy. This would likely hold for this material if chosen to be used. However, the fabrication and inspection may be more difficult than expected which could cause the fabrication cost to significantly increase. Thus an upper limit of $200/kg was assumed for fabrication plus the $150/kg for the material resulting in a $350/kg material unit cost. These assumptions are shown on Table 3.2-8. This sensitivity of the vanadium material cost is evaluated on the blanket options LiPb/LiPb/V and Li/Li/V, tandem mirror design. Table 3.2-8 illustrates that this additional fabrication induces a cost penalty of $40M to $50M of direct capital cost and an increase in the cost of electricity of 1.3 to 1.6 mills/kWh. This single change in a material cost produces a significant effect on the system economics.

The costs for the lithium and lithium compounds used in the basecase analysis are documented in Table 5.3-3. Lithium cases have a high degree of uncertainty when applied to a commercial reactor a number of years distant. Again sizeable discounts from present prices were assumed based upon the mature market assumption, namely large purchases, development of cheaper processing methods, and more pricing leverage. The assumed price of $40/kg is

TABLE 3.2-8
COST SENSITIVITY TO MATERIAL COSTS
TANDEM MIRROR REACTOR

MATL, ENRCHMT, FORM	UNIT COST, $/kg BASELINE	POSSIBLE
V-15Cr-5Ti	250	350
Lithium, Natural	40	72
Li2O, Natural	40	72
$LiAlO_2$, 50%, S Pac	76	108
$LiAlO_2$, 60%	100	132
$LiAlO_2$, 90%, S Pac	190	222
Flibe, Natural	37	48
LiPb, 30%	6.25	6.25
Be, Spherepac	440	150
Be, Rods	440	1000
Be, Balls	440	1000

Option	Matl Cost Changed	Delta Direct Cost (M$)	Delta Annual (M$)	Delta COE (Mills/kWh)	COE	Sensitivity Ranking*	Baseline Ranking
A - $LiAlO_2$/NS/HT-9/Be	$LiAlO_2$,50% Enrichment	18.6	7.6	0.8	82	0.99	1.00
	Be Spherepac	-26.2	-9.8	1.0	80	1.01	1.00
B - Li/Li/HT-9/-	Li, Natural	12.0	5.1	0.5	86	0.94	0.94
C - LiPb/LiPb/V/-	V-15Cr-5Ti	50.3	15.3	1.6	91	0.89	0.90
D - Li/Li/V/-	Li, Natural	12.0	4.8	0.5	88	0.92	0.92
	V-15Cr-5Ti	41.6	12.7	1.3	89	0.91	0.92
E - Li_2O/He/HT-9/-	Li2O	18.6	10.4	1.2	92	0.88	0.89
F - $LiAlO_2$/He/HT-9/Be	$LiAlO_2$,60% Enrichment	14.7	6.2	0.7	91	0.89	0.90
	Be Rods	31.7	2.7	0.3	90	0.90	0.90
G - Li/He/HT-9/-	Li, Natural	16.6	7.2	0.8	95	0.85	0.85
H - Flibe/He/HT-9/Be	Flibe, Natural	2.0	0.9	0.1	88	0.92	0.92
	Be Balls	83.3	28.2	2.8	90	0.90	0.92
I - $LiAlO_2$/H_2O/HT-9/Be	$LiAlO_2$, 90% Enrichment	16.1	5.5	0.6	87	0.93	0.94
	Be Spherepac	-32.0	-11.7	-1.3	85	0.95	0.94

*Ranking derived from Baseline Lowest COE/COE = 81/COE in line with the DOE recommended value(1) for standard unit costs. However, lithium is being used in large quantities for battery applications and thus considerably weakening the argument of quantity discounts. Therefore for a sensitivity study, the cost of 99.8% pure lithium was assumed to cost the current price of \$72/kg. Since there was no fabrication cost for the liquid lithium, this is also the installed price. All other lithium compounds, Li_2O, $LiAlO_2$, and Flibe, were scaled upward by the same delta cost of \$32/kg for the lithium component. Table 3.2-8 illustrates these costs increases. The baseline cost of the enrichment was assumed to be correct. The change in the lithium cost in the lithium-lead mixture did not change the cost of the mixture enough to warrant a sensitivity run on this parameter. When evaluated in the tandem mirror rector design, the postulated unit cost of lithium up to the market value increased the direct costs only by \$12M to \$16M and the COE by a half to three-quarters of a mill/kWh. The lithium oxide has slightly higher capital cost increase but the annual cost had more effect. This was because the liquid lithium needed only to be periodically purified and enriched whereas the lithium oxide must be replaced. The direct cost and COE increases associated with higher $LiAlO_2$ unit costs are intermediate to the lithium/lithium and the Li_2O cases. The Flibe had only a very modest \$2M capital cost increase and a \$0.1 mill/kWh COE increase which indicated the market price of lithium has very little influence on this coolant and breeder material.

The cost of beryllium is also highly uncertain. The source of raw material is controlled by a single company and there is a limited demand for the material. TRW has investigated the fabrication of beryllium for several fusion applications. Their data indicates the raw material would cost in the range of \$350/kg with rods and balls being fabricated using powder metallurgy methods for a cost of \$440/kg (with minimal losses and inspection). See Section 5.3.2 for more information. These data were used for the baseline cost. The beryllium spherepac would also be assumed to cost only slightly over the cost of the powder form, basically screening for proper sizing of the granular mix. The problem with the baseline costs is the variability of the base costs and the fabrication methods. Most current quotes on the raw and scrap metal range from \$400/kg to \$500/kg in large quantities. The powder metallurgy process is also thought not to be applicable to the manufacture of 1-cm diameter balls because of the buildup of material in the dies. Instead the balls will have to be ground to a rough spherical shape at a higher cost,

$3000/kg to $7000/kg. The rods may be able to be made by extruding a powder metallurgy part but the cost is uncertain. Costs were quoted for a drawn rod of a similar size (2-cm diameter) in the $2000/kg range. Thus it is hard to predict exactly what the eventual cost (or design) may be. To estimate the impact of the formed beryllium, the costs were assumed to be $1000/kg for the balls and rods. If the eventual costs are higher than these values, the impact can be scaled from these values. When these unit costs were evaluated in the TMR, the 2-cm diameter beryllium rods increased the capital cost $31.7M but the annual cost raised only 0.3 mill/kWh because the beryllium can be reprocessed at a cost considerably less than the original price. Although the capital cost is considerable, it is not a real detriment for the Cost of Electricity! The 1-cm diameter beryllium balls had an even higher capital cost increase, $83M, because of the considerable usage of beryllium on the Flibe design. Although the balls are also remanufactured at a reasonable cost, the annual cost also contributed a significant amount toward the cost increase. This resulted in a COE increase of 2.8 mills/kWh, which is the highest COE sensitivity shown. The general results to be concluded is that if reasonable amounts of beryllium are utilized in the design, sizeable unit costs can be tolerated without adversely affecting the economics. If a high usage of beryllium is contemplated, unit costs in excess of the anticipated range will result in serious economic consequences.

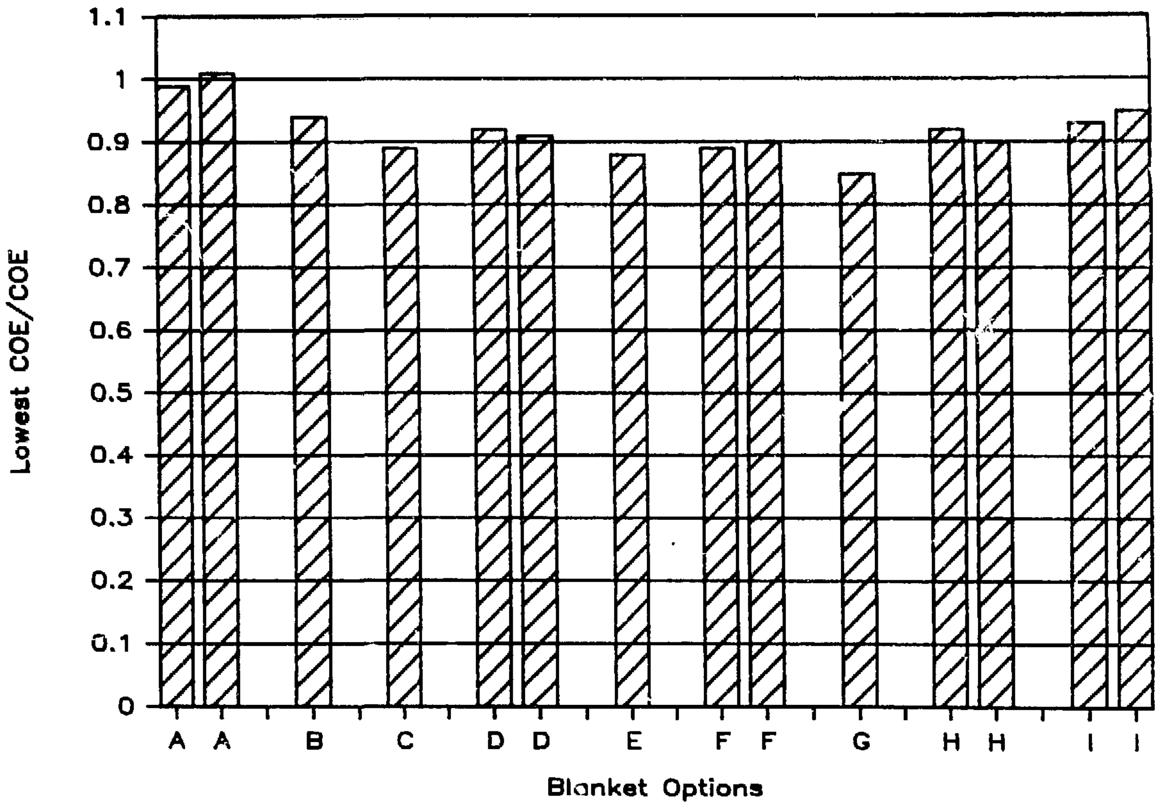
The sensitivity to the cost of the beryllium spherepac presents another interesting possibility. In the course of investigating the cost of beryllium, a breakthrough was forecast for the extraction and initial processing of beryllium. If the beryllium can be atomized and condensed as relatively pure beryllium powder, the costs of raw material would drastically fall to perhaps to the range of $60/kg. The cost of beryllium spherepac would fall to the range of $150/kg. This reduces the direct capital costs to the neighborhood of $30M with a COE reduction of a mill/kWh or more. If the arguments for the difficulty of fabrication of the beryllium rods and balls still hold, this material cost reduction may allow those forms to keep their cost near their baseline values.

All of the above sensitivities were based on unit costs still in the realm of possibility and application of these values produced COE variations of a percent or two. However, the relative rankings did not change appreciably, as shown in Figure 3.2-10. Even if all other concepts were evaluated with baseline values and only a single material cost were increased, usually no or only a minimal reordering of the concepts occurred. Thus it can be concluded that the economic evaluation can be considered as reasonable, even if the unit costs of some materials have a high degree of uncertainty.

Performance Parameters - One of the more important performance parameters throughout the BCSS design process was the energy multiplication factor (EMF). Just how important is it to optimize the materials from a neutronics standpoint to achieve the maximum EMF? The Flibe designers chose to use a high beryllium content to maximize the EMF among other factors, whereas other designers deliberately chose to not have a multiplier. To assess this question, Blanket Option F ($LiAlO_2$/He/HT-9/Be), which uses a beryllium multiplier, was used as a test case. The EMF was increased by 10%, up to 1.42. The better part of this 10% increase was carried throughout the thermal, gross electrical and net electrical power for both the tandem mirror and tokamak. The actual values can be seen in Table 3.2-9. The attendant cost increase, mainly in the heat transport system, is in the range of 2 to 3%. The COE was reduced 6 to 7 percent, which is a significant improvement to the COE and the BCSS rankings. If this could be accomplished, it would be a beneficial design change on both the tandem mirror and the tokamak designs.

The blanket lifetime was a design parameter which should be greater than some minimum value but no maximum value or goal was established. Again Option F was used to evaluate the benefit of doubling the blanket life from 4-5 years to the range of 8-10 years. The effect as shown in Table 3.2-10 is to reduce the Scheduled Component Replacement Costs. This effect is not linear as blanket costs are not the only costs in this category. The COE was reduced by approximately 0.5 mill/kWh for both the tandem mirror and tokamak. The engineering effort to double the lifetime would likely be very costly but this effect is not considered in this sensitivity study. Thus if this cost effect were included, the net benefit of additional life beyond the current postulated values may be negative, or at least, be of marginal benefit. On the other hand, a longer lifetime may achieve a higher reliability value which will increase the system availability and lower the COE. Also shown in this trade study is the effect of a possible reduction of the helium-cooled tandem

MATERIAL COST SENSITIVITY



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

I – $LiAlO_2$/H_2O/HT-9/Be

Figure 3.2-10

TABLE 3.2-9. COST SENSITIVITY TO ENERGY MULTIPLICATION FACTOR

Option "F" - LiAlO2/He/HT-9/Be

Increased EMF by 10%

	Tandem Mirror		Tokamak	
	(baseline)	(new)	(baseline)	(new)
Fusion Power, MW	3083	3083	4173	4173
Energy Multiplication Factor	1.291	1.42	1.28	1.408
Blanket Thermal Output, MW	3182	3500	5107	5534
Total Thermal Power, MW	3557	3882	5413	5857
Gross Electric Power, MW	1723	1853	2122	2296
Net Electric Power, MW	1262	1383	1647	1795
Total Direct Cost, M$	2564	2607	3071	3144
Annual Cost, M$	795	808	901	921
COE, Mills/kWh	90	83	78	73
BCSS Ranking	0.9	0.97	0.79	0.85

TABLE 3.2-10. COST SENSITIVITY TO BLANKET LIFETIME

Option "F": Doubled Blanket Life

	Tandem Mirror			Tokamak	
	(baseline)	(new)	(new)	(baseline)	(new)
Blanket Life, y	4.4	8.8	2.0	4.2	8.4
Annual Sch Repl Costs, M$	13.2	9.7	21.6	15.1	10.9
Annual Cost, M$	795	791	803	901	897
COE, Mills/kWh	89.8	89.5	90.8	78.0	77.7
BCSS Ranking	0.902	0.905	0.892	0.794	0.798

NOTE: The above data do not include the influence of blanket life upon the availability of the plant.

mirror blanket lifetime to approximately two years. The effect is an increase in COE of 1 mill/kWh or approximately a 1% change. There are two other effects which could not be assessed at this time. If the life is shortened, either the availability may be reduced and/or the maintenance equipment must be increased. Either of these factors would tend to increase the COE.

The gross thermal efficiency has a high visibility in the BCSS because of the available choices of primary coolants. What is the potential benefit if additional thermal efficiency can be obtained from a particular coolant? The helium coolant had some promise of increasing the efficiency. A one percent increase in gross thermal efficiency was postulated which resulted in a 2 1/2 to 3% increase in Net Electrical Power with a minimal increase in cost. This same percentage is translated into a 2 1/2 to 3% decrease in COE. These effects are shown in Table 3.2-11. Thus if the efficiency can be increased by raising the coolant temperatures or using more reheats in the turbine at reasonable cost, this would be an area worthy of consideration.

In the tandem mirror design, the thickness of the blanket and shield under the coils determine the size and the cost of the central cell coils. If the thickness of the blanket and shield were decreased by 10 cm out of 120 cm or more, the cost of the central cell magnets are reduced from $195M to $178M, a reduction of 8.7%. See Table 3.2-12 for more details. The radiation protection for the magnet must remain constant. This thickness reduction translates into a 1% change in capital cost, annual cost and COE. This would rank as a moderately beneficial change but would not be classed as a high priority item for a tandem mirror.

A change in the inner blanket and shield thickness on a tokamak has a much more pronounced effect. The peak field radius would be modified, which also would change the fusion power to the fourth power. A 10 cm thickness decrease produces a 11% increase in fusion power and a 13% increase in net power, see Table 3.2-12. However, magnet costs decreased only slightly. The overall Total Direct Costs increased because of the additional power handled. The tokamak COE decreased by approximately 7 percent due to a 10 cm thinner blanket and shield. This design change has a very powerful leverage on the COE. This reduction on the thickness should be investigated as a

TABLE 3.2-11. COST SENSITIVITY TO GROSS THERMAL EFFICIENCY

Option "F": Increased gross thermal efficiency by 1 percentage point

	Tandem Mirror		Tokamak	
	(baseline)	(new)	(baseline)	(new)
Fusion Power, MW	3083	3083	4173	4173
Total Thermal Power, MW	3557	3557	5107	5107
Gross Thermal Efficiency	0.4	0.41	0.392	0.402
Gross Electric Power, MW	1723	1755	2122	2176
Net Electric Power, MW	1262	1294	1647	1701
Total Direct Cost, M$	2564	2567	3071	3076
Annual Cost, M$	795	796	901	902
COE, Mills/kWh	90	88	78	76
BSCC Ranking	0.9	0.92	0.79	0.82

TABLE 3.2-12. COST SENSITIVITY TO BLANKET AND/OR SHIELD THICKNESS

Option "F": Reduced Shield Under Coil (Tandem Mirror) or Inner Shield (Tokamak) Thickness by 10 cm

	Tandem Mirror		Tokamak	
	(baseline)	(new)	(baseline)	(new)
Peak Field Radius, m	-	-	3.52	3.62
Field on Axis, T	-	-	5.029	5.171
Fusion Power, MW	3083	3083	4173	4667
Total Thermal Power, MW	3557	3557	5413	6043
Net Electric Power, MW	1262	1262	1647	1857 (+13%)
CC or TF Magnet Cost, M$	195	178 (-8.7%)	248	244 (-1.6%)
Cost of Coil or Inner Shield	28	23	13	11
Cost of Heat Transport, M$	225	225	334	369
Total Direct Cost, M$	2564	2537	3071	3167
Annual Cost, M$	795	787 (-1.1%)	901	929 (+3%)
COE, Mills/kWh	90	89	78	71
BCSS Ranking	0.9	0.91	0.79	0.87

priority item. Figure 3.2-11 extends the thickness variation over a larger range, further illustrating the importance of a thinner blanket and shield on the inner leg of a tokamak reactor.

Another facet of the choice of the primary coolant was the pumping power required. There was much discussion over the lithium and lithium-lead MHD effects and the helium pumping power. The Option F with helium coolant was again used as the test case to evaluate the sensitivity to pumping power. Table 3.2-13 illustrates an increase in pumping power of 40% on both the tandem mirror and tokamak. Although 85% of the helium pumping power is recovered as useful thermal energy, only approximately 40% of that thermal energy is converted back into electrical energy. In this case, the net electrical power was decreased by 2% on the tandem mirror and 4% on the tokamak, with a similar percentage point change in the COE. This change also has a high leverage, but it is doubtful if the pumping power can be reduced much from the present levels. However, all efforts should be concentrated to keep the pumping power from increasing above current levels. For example, if the minimum operating temperature of HT-9 caused an increase in the helium inlet temperature, the pumping power would rise significantly.

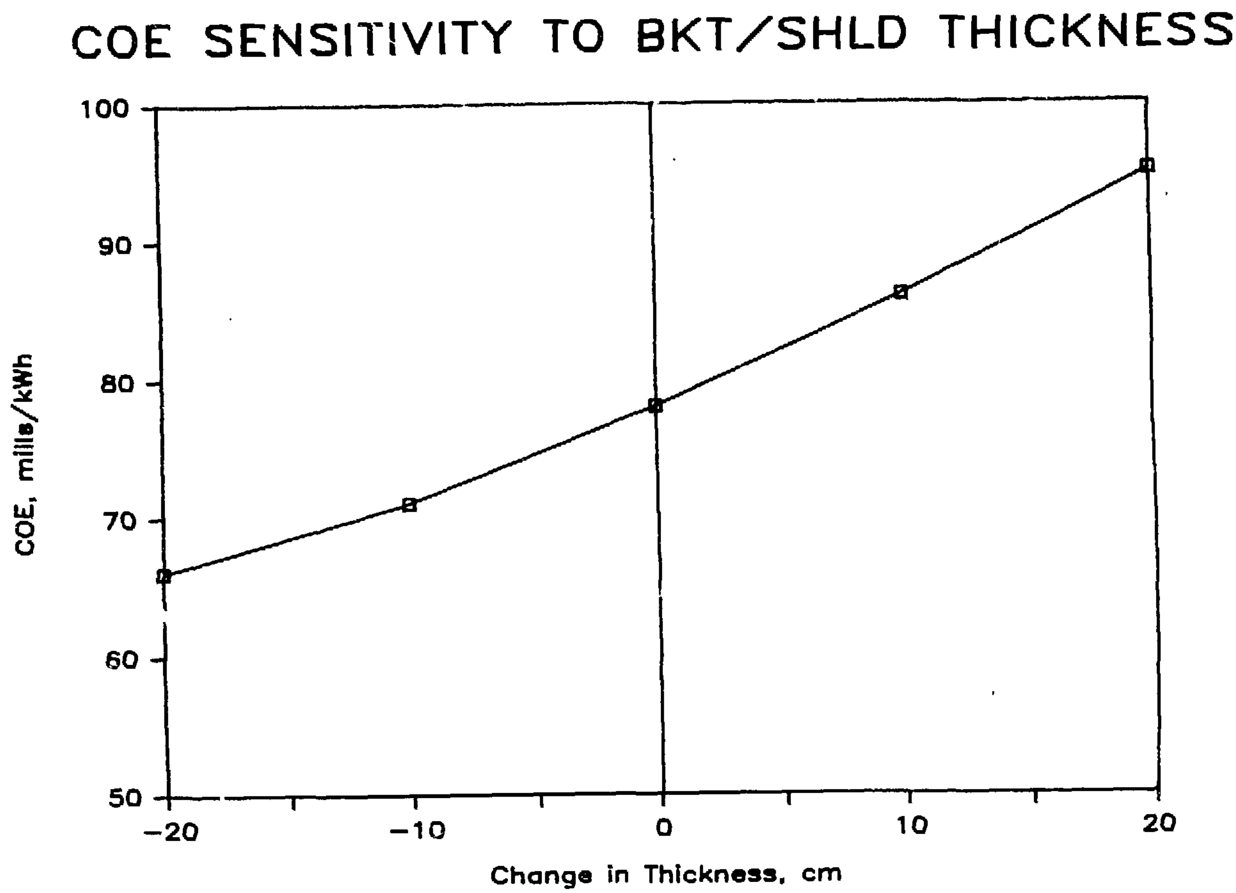


Figure 3.2-11

TABLE 3.2-13. COST SENSITIVITY TO PUMPING POWER

Option "F": Increased pumping power by 40%

	Tandem Mirror		Tokamak	
	(baseline)	(new)	(baseline)	(new)
Pumping Power MW	87.9	124.3	254	356.3
Pump Power Addition, MW	74.7	105.6	215.9	302.9
Gross Electric Power, MW	1723	1736	2122	2156
Net Electric Power, MW	1262	1238	1647	1578
Total Direct Cost, M$	2564	2578	3071	3120
Annual Cost, M$	795	799	901	914
COE, Mills/kWh	90	92	78	83
BCSS Ranking	0.9	0.88	0.79	0.75

3.3 Safety

Fusion's societal acceptance and its ultimate potential as an energy source will depend, in part, on how successful the public and plant workers are protected from potential harm. Just as the choice of blanket should decrease costs and enhance fusion's economic attractiveness, so should the choice of blanket decrease the potential risk to the public and enhance fusion's safety and environmental attractiveness. The Safety Evaluation is therefore intended to measure the relative safety and environmental attractiveness of the final group of blanket options.

A complete probabilistic risk assessment of the entire fuel and facility cycle would produce an integrated value of total risk to the public, a single figure-of-merit including all significant risks in proportion to their contribution to the total. However, such an analysis is beyond current resources and knowledge. Instead, eleven individual safety indices were defined to compare blanket attractiveness in specific areas. Then, each index was weighted to reflect an estimate of their importance to total risk. Thus, the Safety Evaluation is intended to approximate, to the extent possible, a relative risk assessment comparison by focusing on various specific areas of possible differences among blankets.

Readers are referred to Section 5.4 for a full description of the methodology, technical basis for comparison, and results. The following subsections are brief summaries of the results, summaries of some sensitivity cases, discussion of the results, and overall safety conclusions.

3.3.1 Methodology and Results

Each blanket concept received a score for each index, I_i, between 0.0 and 1.0. Individual indices are mixtures of quantitative and/or qualitative information. The quantitative information was used in the form of certain figures-of-merit. The range of the various figures-of-merit among designs for a given issue varied from a factor of 3, e.g., blanket lifetime, to a factor of 7 orders of magnitude, e.g., radiation field around coolant piping.

In a sense, the entire evaluation was conducted twice, once for mirror versions of blankets and once for Tokamak versions of blankets. No cross comparison between mirror blankets and Tokamak blankets was desired or attempted.

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:

$$SFM = \sum_{i=1}^{11} I_i W_i \quad .$$

Overall, 76% of the total SFM is based on quantified figures-of-merit; 24% is based on engineering judgement.

The eleven indices are listed in Table 3.3-1. They fall into four general areas: accident source term characterization, accident fault tolerance, effluent control, and maintenance and waste management. A balance (60-40%) was established between accident issues and nonaccident issues, which is a compromise between the general public perception that accidents should be weighted high and the actual low weighting for accidents that result from total fuel/facility cycle risk studies for other energy technologies. The 60% accident weighting was divided equally between accident source term characterization (30%, Indices 1-3) and accident fault tolerance (30%, Indices 4-8). The 40% nonaccident weighting was divided equally between effluent control (20%, Index 9) and maintenance and waste management (20%, Indices 10-11). These four areas will now be briefly discussed. The resulting weighted scores for the reference blankets in these four areas are listed in Tables 3.3-2 and 3.3-3.

TABLE 3.3-1. SAFETY EVALUATION INDICES

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 [illegible]ioactive Effluents	20
10	Occupational Exposure	10
11	Waste Management	10

TABLE 3.3-2. SAFETY EVALUATION RESULTS FOR MIRROR BLANKETS

Blanket	Accident Source Term Characterization. Indices 1-3	Accident Fault Tolerance. Indices 4-8	Effluent Control. Index 9	Maintenance and Waste Management. Indices 10-11	Total
Li/Li/V	11.9	19.5	17.0	14.6	63.0
Li/Li/HT-9	7.5	21.4	17.0	7.9	53.8
LiPb/LiPb/V	21.8	23.5	10.0	9.4	64.7
Li/He/HT-9	16.6	22.6	10.0	11.3	60.5
FLIBE/He/HT-9/Be	16.4	24.7	0.0	9.5	50.6
Li_2O/He/HT-9	21.0	23.9	7.4	11.5	63.8
$LiAlO_2$/He/HT-9/Be	18.4	24.0	3.8	12.3	58.5
$LiAlO_2$/NS/HT-9/Be	4.8	24.0	0.0	6.9	35.7
$LiAlO_2$/H_2O/HT-9/Be	8.1	19.9	2.2	8.3	38.5

TABLE 3.3-3. SAFETY EVALUATION RESULTS FOR TOKAMAK BLANKETS

Blanket	Accident Source Term Characterization. Indices 1-3	Accident Fault Tolerance. Indices 4-8	Effluent Control. Index 9	Maintenance and Waste Management. Indices 10-11	Total
Li/Li/V	11.9	17.5	15.2	15.1	59.7
Li/He/HT-9	16.4	19.7	8.0	11.2	55.3
FLIBE/He/HT-9/Be	16.4	21.8	0.0	10.1	48.3
Li_2O/He/HT-9	20.9	20.9	6.2	11.8	59.8
$LiAlO_2$/He/HT-9/Be	17.8	21.0	3.6	11.7	54.1
$LiAlO_2$/NS/HT-9/Be	4.0	21.0	0.0	5.8	30.8
$LiAlO_2$/H_2O/HT-9/Be	8.9	16.9	1.2	8.7	35.7

3.3.1.1 Accident Source Term Characterization

This first category (Indices 1-3) relates to the component of accident risk from the radioactive and chemical toxicity source term common to accident initiators. The three indices measure the source term for the structure, breeder/multiplier, and coolant, respectively. In each case, the index is divided equally between the hazard from activation products and the hazard from tritium inventory. The activation hazard measurement starts with the activation inventory (see Section 6.12) translated into Biological Hazard Potential (BHP) for breeder and coolant materials or Public Health Effects (PHE) for structural materials. The PHE is an actual calculation of public health effects from an accidental release of one m^3 of first wall; had sufficient data been available, they would have been used for breeder and coolant materials also. Both the BHP and PHE were adjusted by the relative volatility of the elements involved. One effect is to penalize liquid lithium because the prime activation species, Na^{22} and Na^{24}, would be highly volatile in the event of lithium combustion. The chemical toxicity of beryllium was added into the breeder BHP calculation for those blankets including beryllium. However, the impurities in $LiAlO_2$ produce sufficient activation so that beryllium toxicity had only about a 30% impact on $LiAlO_2$/Be BHP. The tritium hazard was measured by the vulnerable tritium inventory. The entire tritium inventory was considered vulnerable, except for $LiAlO_2$, where only 10% was considered vulnerable.

The various resulting figures-of-merit generally varied by 3 or 4 orders of magnitude among designs. A logarithmic utility function was used for each one to compress the range of values to that portion of the 0.0-to-1.0 index score allotted for that particular figure-of-merit. Therefore, a 1% change in SFM translates into more than a 1% change in total risk. The total accident source term characterization score, with a possible range of 0 to 30, is listed in Tables 3.3-2 and 3.3-3 for the reference blankets. Details may be found in Section 5.4.2.

3.3.1.2 Accident Fault Tolerance

The response of each blanket design to specific accident initiators (and their likelihood), such as loss of power, loss of coolant, and coolant tube failures, constitute the fault tolerance Indices 4 to 8. The five indices measure the success of a blanket with regard to passive resistance to cooling tube breaks, cooling transients, external events, and near-blanket component failures, and in minimizing the impact on the containment function of the reactor building. Scores were determined by comparing the anticipated blanket response to each transient with a priorly established design guideline. Thus, design teams had a target. Limitations of resource and knowledge prevented most responses from being directly determined by transient calculations. Unlike the other indices, the 30% of the SFM associated with fault tolerance was based primarily on engineering judgment. Thus, together the five fault tolerance indices attempt to measure the inherent or passive safety of each design with respect to specific fault conditions.

Two key design decisions were made that dramatically improve the safety attractiveness of lithium and vanadium blankets. First, water-cooled limiters appeared unacceptable from a safety point of view with lithium or vanadium blankets. Therefore, lithium or helium-cooled limiters were assumed for those cases. It was not possible for the BCSS to perform sufficient analysis to firmly establish the feasibility of these limiter concepts, although sufficient analysis was done to establish the possiblity of using non-water-cooled limiters. Second, nitrogen building atmospheres were used for lithium blankets to substantially reduce fire concerns. Air and carbon dioxide can produce substantial lithium fires. The lithium-nitrogen reaction appears sufficiently benign so as to present only a minor risk when nitrogen atmospheres are used. Nitrogen building atmospheres were used for vanadium blankets to substantially reduce oxidation concerns. Air apparently can produce rapid vanadium oxidation in temperatures above 650°C and partial pressures over about 10^{-4} atmospheres. It is emphasized that the favorable safety ranking of blankets with lithium or vanadium, and to a lesser extent 17Li83Pb, requires these two design decisions and assumptions.

These two design decisions substantially reduce potential chemical reaction problems. In addition, a steel liner is used to prevent concrete reactions. In the final evaluation, lithium was still penalized because water was still available for combustion in shields, resistive choke coils, direct converters, and halo scrapers, and because the steel liner over concrete could fail. Vanadium was also penalized because of the fear of water-induced oxidation of the metal.

The resulting total fault tolerance scores are listed in Tables 3.3-2 and 3.3-3. One result is that the overall fault tolerance score was not a major discriminator among the blankets. Although significant differences occurred for specific fault tolerance issues, overall they tended to balance out. Further details are found in Subsection 5.4.3.

3.3.1.3 Effluent Control

With the exception of the Nitrate Salt designs, the effluent control problem of blankets appears to be dominated by tritium, and was judged on that basis. The potassium in Nitrate Salt leads to production of copious amounts of ^{39}Ar (269 yr), sufficient to require about 99% capture efficiency for the noble gas that is generated. For both tritium and ^{39}Ar, the basis for judgment is the currently proposed U.S. EPA standard of 10 mrem/yr maximum exposure for an individual in the public.

The index score for tritium-dominated designs was determined by the following utility function:

$$
\begin{aligned}
I_9 &= 1.0 && R < 1.0 \text{ Ci/day} \\
&= 0.5 \log (100/R) && 1.0 < R < 100 \text{ Ci/day} \\
&= 0.0 && R > 100 \text{ Ci/day}
\end{aligned}
$$

where

I_9 = effluent score, Index 9

R = tritium release rate to the steam generator in Ci/day.

The Nitrate Salt concepts were scored 0.0 because of the ^{39}Ar problem.

Of all parts of the safety evaluation, the tritium control area is the most uncertain. The tritium release calculations (see Section 6.6) are based on the following:

- o Substantial tritium enters the coolant via the first wall
- o All tritium leaving the solid breeders is in the highly permeable elemental form, partially because hydrogen is added to the purge stream
- o Hydrogen is added to the solid breeder purge or 17Li83Pb counter-flow separator streams to facilitate tritium leaving the breeder surface

- Hydrogen mixed with tritium reduces tritium permeation by a dilution effect

- Permeation through oxidized HT-9 is a factor of 100 lower than classically predicted for clean metal, i.e., an "oxide barrier factor" of 100 is used

- Tritium is not oxidized as it passes through oxidized metal walls

- Tritium oxidation kinetics in helium streams are sufficiently slow so that oxidation of tritium does not occur (a) between solid breeder surface and tube walls or (b) between coolant tube walls and steam generator.

Reality is unknown. However, mounting evidence points in the general direction of this picture. Some of these conditions make tritium control more difficult; some make it easier.

It is emphasized that several of the above statements are different from past design studies, especially the addition of hydrogen to helium purge streams and the release of tritium from solid breeders in the permeable elemental form. The above assuptions are based on more recent experimental results not available to past studies. The net effect is to make tritium control for the solid breeders much more difficult than heretofore believed, reducing the safety attractiveness of those concepts. Future results could prove tritium control for most concepts to be either more or less favorable. Current tritium control calculations are uncertain by at least an order of magnitude.

The results are listed in Tables 3.3-2 and 3.3-3. Further details are found in Subsection 5.4.4.

3.3.1.4 Maintenance and Waste Management

Maintenance/Occupational Exposure, Index 10, was judged by three quantified figures-of-merit. First, the exposure from cooling system/steam generator maintenance was measured by the Remote Maintenance Rating (RMR) of the coolant alone, evaluated one day after shutdown. The RMR is the contact dose rate in mR/h for an infinite slab of material. Second, the blanket maintenance exposure was judged by the number of blanket changeouts over the reactor lifetime. Third, the tritium exposure was measured by the total blanket tritium inventory.

Waste Management, Index 11, was also judged by three quantified figures-of-merit. First, the ability for blanket materials to meet near-surface burial requirements was measured by the Waste Disposal Rating (WDR) averaged over the blanket. The WDR is the ratio of activity divided by near-surface burial isotope concentration limits, per 10CFR61. Second, the exposure during waste handling and processing was measured by the RMR averaged over the blanket evaluated at 10 years after shutdown. Third, the difficulty and risk of handling and transporting waste material was measured by the total waste volume, integrated over the reactor lifetime.

The results are listed in Tables 3.3-2 and 3.3-3. A key finding was that $LiAlO_2$ and 17Li83Pb do not meet near-surface burial requirements whereas Li, Li_2O, and FLIBE do. This did not particularly hurt the various $LiAlO_2$/HT-9 or 17Li83Pb/HT-9 concepts because HT-9 did not meet the near-surface requirements either. The 17Li83Pb/V15Cr5Ti design is penalized because V15Cr5Ti meets near-surface burial whereas 17Li83Pb does not. If future work continues on ternary solid breeders, a silicon-based ceramic would probably be better than $LiAlO_2$ from the long-term activation standpoint. In the limit of very low impurities in a silicon-based ceramic, its SFM score would approach that of Li_2O. A zirconium-based ceramic would be much less favorable. Further details are explained in Subsection 5.4.5.

3.3.1.5 Overall Safety Figure-of-Merit

Tables 3.3-2 and 3.3-3 also list the total SFM for the reference blankets. These are graphically displayed in Figures 3.3-1 and 3.3-2. Overall, the most attractive blankets are LiPb/LiPb/V (TMR only), Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9. The range of mirror SFM scores is

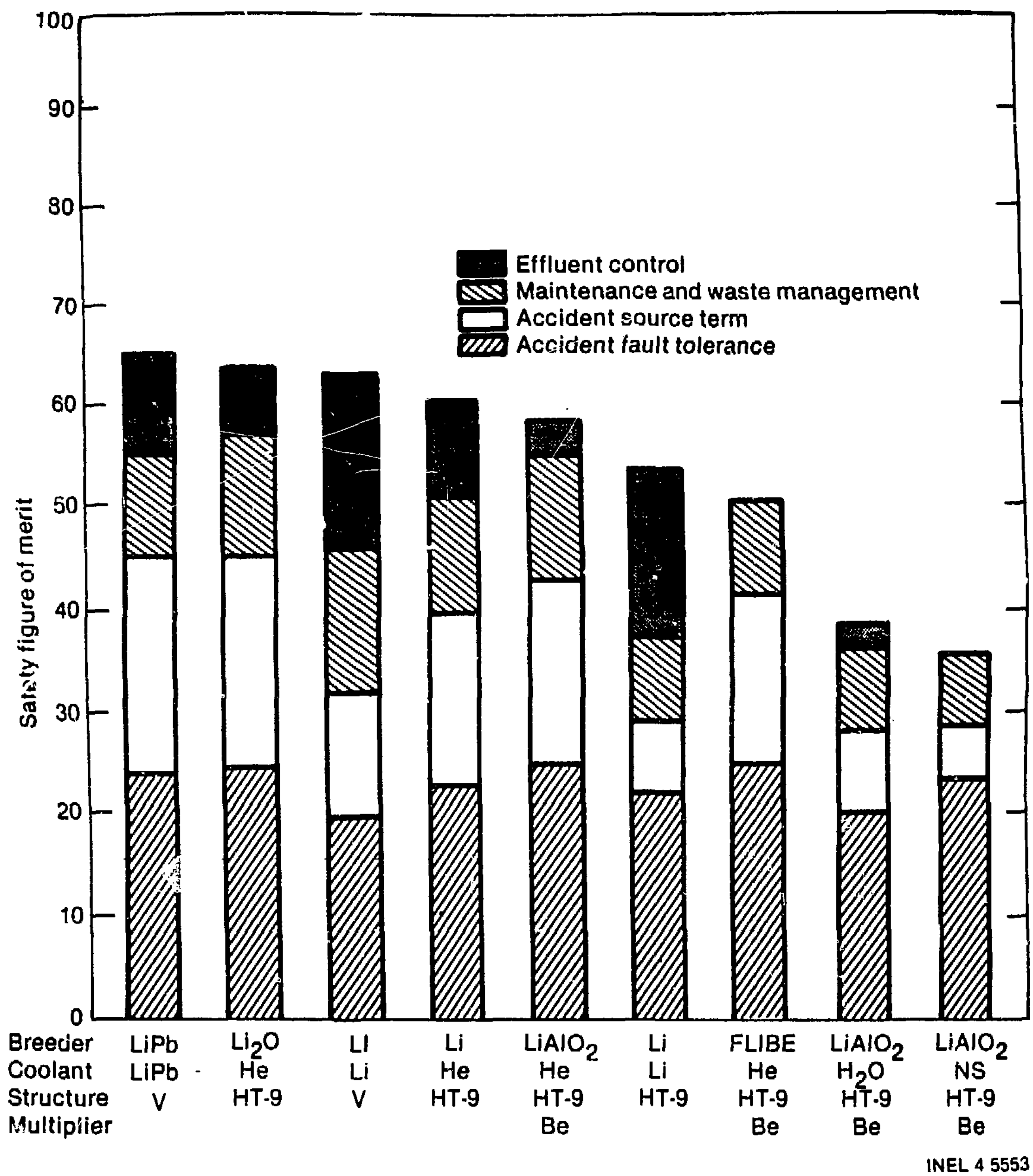


Figure 3.3-1. Safety Evaluation Results for Mirror Blankets

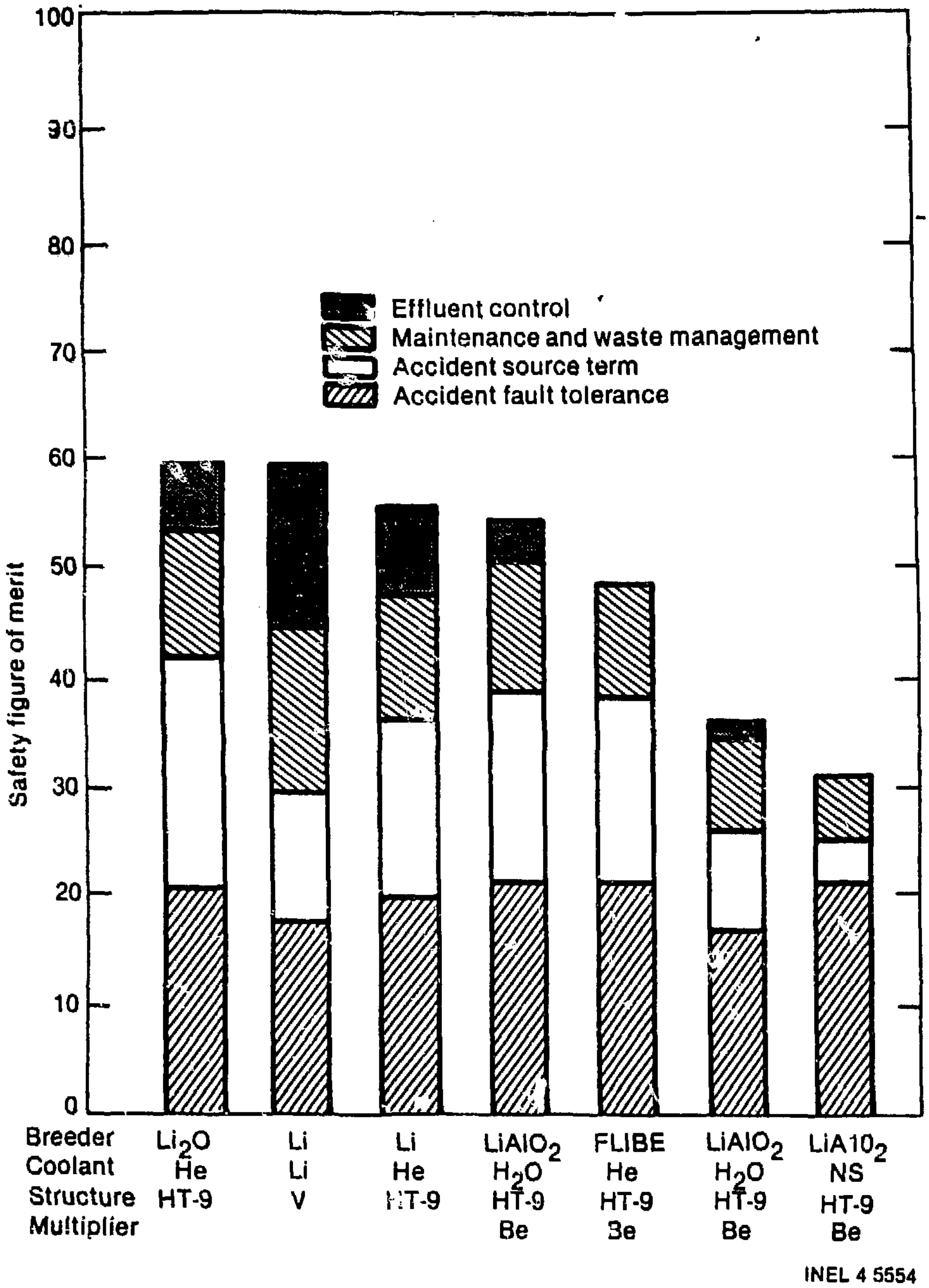


Figure 3.3-2. Safety Evaluation Results for Tokamak Blankets

64 to 36. The range of tokamak SFM scores is 60 to 31. The rank ordering among mirror concepts was the same as among tokamak concepts. In interpreting the numerical results, it is emphasized that a 1% change in the SFM score would translate into much more than a 1% change in total safety risk because of the use of logarithmic utility functions. Also, there is not a go/no-go cut-off score for safety attractiveness in this evaluation. Higher scores indicate blankets with a higher probability of public acceptance and licensing approval and a lower probability of needing costly and complicated special safety systems. The following sections include discussion of the sensitivity of the results and a brief discussion for each blanket concept.

3.3.2 Sensitivity Cases

Several sensitivity cases were examined. Only a few are mentioned here; all can be found in Section 5.4. The following cases are discussed here: "low-activation" steels, a more risk-based SFM, optimistic effluent control, pessimistic effluent control, optimistic chemical reaction control, and pessimistic chemical reaction control. In addition, two hypothetical cases were briefly examined, the impact of changing the reference 5 MW/m^2 neutron wall loading and other physics concepts.

3.3.2.1 "Low-Activation" Steels

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 proposed "low-activation" steels solve the waste disposal problem by eliminating elements that give rise to long-term (>10-yr) isotopes and replacing them with elements, tungsten and manganese, that give 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 a negative one. On balance, the proposed "low-activation" steels are not necessarily a net safety improvement.

It appears that Modified HT-9 is a net improvement over HT-9 even though the tungsten in Modified HT-9, gives rise to several 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. V15Cr5Ti is better from the activation standpoint. A very low activation material like SiC would be even better.

In terms of SFM points, the use of Modified HT-9 versus reference HT-9 could range from lowering the SFM score of a concept by 2.3 to raising the score by 5.5. For Tenelon versus PCA, the impact could range from lowering the SFM by 5.8 points to raising it by 1.4. In both cases, the range is caused by the variation of the impact of the substitution among concepts and by the uncertainty of the impact for a given activation area. The beneficial impact is masked in most designs by use of breeder or coolant materials, 17Li83Pb, $LiAlO_2$, Nitrate Salt, that already do not meet near-surface burial requirements. Thus, the beneficial aspect of the "low-activation" steels is much lower for concepts using a higher activation coolant or breeder. The main drawback of the proposed "low-activation" steels appears to be an increase in afterheat levels, making passive tolerance to cooling transients more difficult, especially for Tenelon. The afterheat drawback was not directly quantified because resource limitations prevented cooling transient calculations for "low activation" steel blankets.

Whereas it is agreed that hands-on maintenance of BCSS blanket structures appears unlikely, limited hands-on maintenance of tritium purge and coolant systems, e.g., the steam generator, may be possible. Use of helium appears to be the best way to achieve these worthwhile goals. Also, one safety advantage of lithium coolant is the use of an intermediate loop so that the steam generator does not have an activated fluid, though the lithium-to-sodium heat exchanger would have activated corrosion products and impurities.

For the reference-composition structural materials and activation areas studied, impurities were generally not found to be important. The main exception was that niobium content causes waste management problems.

Impurities in several breeder materials are definitely important, especially sodium and potassium in all lithium-bearing materials except FLIBE. Economical reduction of impurities in lithium and Li_2O is a worthwhile goal. The reference level of lithium-related impurities, specifically sodium and potassium, may be too high for those breeder materials that use isotopically enriched lithium. If so, 17Li83Pb and $LiAlO_2$ would still be of higher-activation than Li and Li_2O.

3.3.2.2 Risk-Based Safety Figure-of-Merit

The reference SFM divides the 60% weighting for accidents into source term characterization and fault tolerance, ultimately added together. An alternate SFM was defined whereby the total source term characterization score was multiplied by the total fault tolerance score. This would be a closer approximation of risk. With one exception, the use of the alternate SFM did not change the rank ordering of designs. The one exception was reversal of Li/Li/HT-9 (6th in the reference case) with FLIBE/He/HT-9/Be (7th) among the 9 TMR concepts.

3.3.2.3 Optimistic Effluent Control

The tritium effluent calculations could be too conservative. A limiting sensitivity case was defined where all blankets got the maximum score, 20 points for effluent control. The resulting change in rankings is listed in Table 3.3-4. The actual numerical SFMs for this and other sensitivity cases can be found in Subsection 5.4.6. As seen in the table, the liquid metal cases go down, whereas some of the solid or FLIBE breeder cases go up. The reason is simply that the reference liquid metal breeder cases have the best tritium control for the base case and are therefore helped least by the optimistic case. The reference top blankets of LiPb/LiPb/V, Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9 would become Li_2O/He/HT-9, LiPb/LiPb/V, and $LiAlO_2$/He/HT-9. Thus the Li/Li/V and Li/He/HT-9 designs would fall out of the top group. <u>In other words, the high ranking of lithium-breeder concepts in the reference case is a consequence of poor tritium control for most of the other blankets.</u>

TABLE 3.3-4. SENSITIVITY CASES FOR REFERENCE MIRROR BLANKETS FOR SAFETY RANKINGS[a]

Blanket	Base Case	Optimistic Effluent Control	Pessimistic Effluent Control	Optimistic Chemical Reaction Control	Pessimistic Chemical Reaction Control
Li/Li/V	3	6	1	1	--
Li/Li/HT-9	6	7	2	6	--
LiPb/LiPb/V	1	2	--	2	--
Li/He/HT-9	4	5	3	4	--
FLIBE/He/HT-9/Be	7	4	--	7	3
Li_2O/He/HT-9/Be	2	1	--	3	1
$LiAlO_2$/He/HT-9	5	2	--	5	2
$LiAlO_2$/NS/HT-9/Be	9	9	--	9	5
$LiAlO_2$/H_2O/HT-9/Be	8	8	--	8	4

a. Ranking for Tokamak cases are the same after deleting the Li/Li/HT-9 and LiPb/LiPb/V cases. Circled numbers refer to top group of blankets.

3.3.2.4 Pessimistic Effluent Control

The tritium effluent assumptions and base case results could be too optimistic. A limiting sensitivity case was defined where the lithium self-cooled blankets received full credit for tritium control because only they are fairly immune to the various tritium control uncertainties. The Li/He/HT-9 design was given a score of 0.0 for tritium control, but it still might be capable of adequate tritium control. Because Li/He/HT-9 uses a lithium breeder that operates with very low tritium pressure, the only effluent problem for Li/He/HT-9 is tritium getting into the coolant via the first wall. The remaining helium-cooled concepts and water-cooled concepts might become unacceptable as more tritium entered the coolant from either the first wall or breeder zone. The LiPb-cooled blankets already depend on flowing all the coolant through a counter-flow tritium separator;

further unfavorable data for the LiPb case might make it unacceptable or require an intermediate loop (an important economic penalty) for adequate tritium control. The Nitrate Salt may have favorable tritium control but has other severe problems, e.g., ^{39}Ar. A Nitrate Salt intermediate loop, potentially used for tritium control for other blankets, would not have the ^{39}Ar problem.

In the pessimistic effluent case, the top blankets of LiPb/LiPb/V, Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9 would be replaced by Li/Li/V, Li/Li/HT-9, and Li/He/HT-9, with most other blankets perhaps becoming unacceptable. Note that the optimistic tritium control case favors LiPb/LiPb/V and Li_2O/He/HT-9 among the reference top group, whereas the pessimistic tritium case favors Li/Li/V and Li/He/HT-9.

3.3.2.5 Optimistic Chemical Reaction Control

Chemical reaction control could be even better than the reference case, which allows water in shields, choke coils, direct convertors, and halo scrapers but assumes adequate isolation between water and reactive metals. If the water were replaced, then the remaining lithium and vanadium fault tolerance penalties for water chemical reactions (Index 7) would be removed. In this case, the reference four top blankets would probably not change, but the Li/Li/V and LiPb/LiPb/V cases would rise to the top of the list.

Actually, the full impact of the optimistic chemical reaction control case is even higher, since the high volatility assigned to alkali elements in liquid lithium because of the possibility of combustion would be lowered. In this instance, Li/Li/HT-9 might displace Li_2O/He/HT-9 in the top group of blankets.

3.3.2.6 Pessimistic Chemical Reaction Control

As discussed above, the favorable ranking of lithium and vanadium concepts depends on the various design decisions and assumptions that largely eliminate chemical reaction concerns. Some of these assumptions could be too optimistic. For example, nonwater-cooled limiters may not be credible. Nonair building atmospheres or adequate passive fire prevention techniques may prove too expensive. In these cases, the concepts bearing lithium or vanadium may prove to be unacceptable. If so, the reference top blankets, LiPb/LiPb/V, Li_2O/He/HT-9, Li/Li/V, and Li/He/HT-9 would be replaced by Li_2O/He/HT-9, $LiAlO_2$/He/HT-9/Be, and FLIBE/He/HT-9/Be.

3.3.2.7 Impact of Deviation from 5 MW/m^2 Neutron Wall Loading

The exact value of the wall-loading value will change design details. In general, the relative ranking and attractiveness of concepts in the Safety Evaluation would not change if the 5 MW/m^2 value were lowered or raised. However, two specific issues might change, which might provide an impact on the overall Safety Evaluation. The first issue is the relative ability of concepts to survive cooling transients. Afterheat levels will vary up or down with the wall loading. An increased wall loading will harm the liquid breeder concepts since some may no longer be able to passively deal with cooling transients. (It is already assumed that solid breeder issues have troubles in this regard.) A decreased wall loading would probably have little effect since it is not likely that the afterheat level could fall sufficiently to allow solid breeder concepts to passively handle these transients. The second issue is whether a change in neutron wall loading makes it harder or easier to use non-water-cooling for near-plasma components, e.g., limiter. Assuming that the surface heat flux would scale roughly as the neutron wall loading, it seems likely that increased wall loading would make it more difficult to replace water cooling, hence significantly lowering the safety attractiveness of lithium and/or vanadium concepts. Similarly, reduced wall loading seems likely to improve chances of replacing water cooling, hence raising the attractiveness of reactive metal blankets.

In summary, raising the 5 MW/m^2 value would tend to decrease the safety attractiveness of liquid-metal and/or vanadium blankets because of increased difficulty in passively handling cooling transients and in replacing water cooling in near-plasma components. Decreasing the

5 MW/m^2 value would tend to increase the safety attractiveness of lithium and/or vanadium blankets because of increased ability to replace water cooling.

3.3.2.8 Other Physics Confinement Concepts

The Safety Evaluation found near-identical relative rankings among concepts independent of whether the physics concept was TMR or tokamak. Although the rank ordering would have to be examined for other physics concepts on a case-by-case basis, it does not appear that the safety rank ordering would necessarily change. The most likely issues that could lead to changes are (a) differences in the chemical reaction risk of water and (b) presence of significant amounts of copper. If a given physics concept had special requirements for water-cooled components, it could severely harm the vanadium and/or lithium concepts. On the other hand, if a given physics concept had no requirements for water-cooled components, the blankets with reactive metals would be helped. If significant amounts of copper were present, the Li/Li/V concept relative ranking would decrease. This is because copper grossly fails 10CFR61 and its presence would mean no blanket, even Li/Li/V, would score well in the Waste Management Index. If a physics concept required copper coils imbedded in the blanket, for example, one could forget about fusion meeting the near-surface burial goal.

3.3.3 Discussion of Results by Blanket

The following discussion is intended to give a brief description of why each of the reference blankets ranks where it does. The order is best to worst, recognizing that only the "top" final group of blanket concepts were evaluated.

The Li_2O/He/HT-9 blanket generally does very well in all areas. The largest uncertainty is tritium control: if tritium control becomes significantly easier, this blanket becomes the unquestioned best choice; if tritium control is significantly less favorable, the blanket is far less attractive and may become unacceptable if tritium control gets two orders of magnitude worse.

The LiPb/LiPb/V blanket (TMR only) scores about the same as Li_2O/He/HT-9. This blanket also does well across the board, generally a little better than Li_2O/He/HT-9 in tritium control and worse in occupational exposure and waste management. The latter is caused by the high activity of 17Li83Pb. This blanket has two key uncertainty areas: how well will the tritium control scheme work, and how well will LiPb and V be protected from water and air? Highly unfavorable outcomes in either area might make the blanket unacceptable. There is no obvious area of significant safety improvement.

The Li/Li/V design scores about the same as LiPb/LiPb/V. The lithium advantages in less radioactivity and better tritium control are offset by chemical reaction concerns and higher tritium inventory in lithium. This is the only blanket with all materials passing the 10CFR61 near-surface burial goal. This blanket could be significantly hurt, to the point of being unacceptable, if air and water reactions were not controlled to the extent assumed in this study. On the other hand, if air and water reaction risk is further lowered (e.g., allowing neither water or air in the reactor building), the blanket would be even more attractive and would be the best overall choice.

Next comes the Li/He/HT-9 blanket. The use of HT-9 (higher radioactivity) and helium (worse tritium control) outweighs its advantages over Li/Li/V in the area of helium's low radioactivity. Tritium control is the major uncertainty, which could raise or lower the attractiveness of this blanket. However, the overall safety attractiveness is significantly less than the cases of Li_2O/He/HT-9 and LiPb/LiPb/V.

The fifth blanket is $LiAlO_2$/He/HT-9/Be. All of the higher ranked blankets have significantly lower activation: Li_2O/He/HT-9 is lower because of Li_2O versus $LiAlO_2$/Be; LiPb/LiPb/V is lower because of V versus HT-9; Li/Li/V is lower because of Li versus $LiAlO_2$/Be and V versus HT-9; and Li/He/HT-9 is lower because of Li versus $LiAlO_2$/Be. All of the higher blankets also do better in the area of tritium control. The $LiAlO_2$/He/HT-9/Be blanket is as close to the others as it is because of the lack of chemical reaction concerns. The blanket would be significantly more attractive if tritium control were 1 to 2 orders of magnitude better, but not better than Li_2O/He/HT-9 because of the activation and chemical toxicity difference between Li_2O and $LiAlO_2$/Be.

The sixth blanket, Li/Li/HT-9 (TMR only), is very similar to Li/Li/V, except for higher activation from HT-9. The V to HT-9 activation difference is the largest for this pair of blankets, because lithium is fairly low activation (does not mask the structure's activation), and because there is no difference in tritium control. The latter is predicated on using steel for the Li/Li/V loop for the nonblanket parts. Overall, the activation disadvantage for HT-9 is sufficient to drop Li/Li/HT-9 into the middle of the pack. Better chemical reaction control would help this blanket, as for Li/Li/V, and raise its attractiveness, but never to more than Li/Li/V.

The seventh blanket, FLIBE/He/HT-9/Be, does very poorly in tritium control and is heavily penalized as a result. The chemical toxicity of beryllium is a distinct disadvantage, because otherwise FLIBE would compare favorably with Li_2O in terms of BHP. The blanket would be helped if tritium control were improved by two orders of magnitude or if it were found that beryllium toxicity is not as bad as assumed here (see Subsection 5.4.2.3). However, it does not appear that this blanket could be more attractive than Li_2O/He/HT-9 or $LiAlO_2$/He/HT-9/Be.

The eighth blanket is $LiAlO_2$/H_2O/HT-9/Be. The blanket does poorly in all four major safety areas. Its biggest problems are tritiated control and pressure. The pressure is inherently high enough so that tritium water may leak from the primary to steam side. The high pressure makes entrained activation products and tritium very mobile. The two-phase high-pressure nature allows for significant pressurization of whatever chamber the water would leak into. The blanket does avoid chemical reaction problems. This blanket scores at the bottom for all sensitivity cases studied.

The ninth blanket is $LiAlO_2$/NS/HT-9/Be. This blanket does poorly in all safety areas except fault tolerance, where the low operating pressure is an advantage. This advantage is based on the questionable assumption that salt decomposition is not a problem during transients. If that assumption is not made, the blanket appears potentially unacceptable. The very high activity and high tritium inventory in the salt outweigh its low pressure advantages. Rather than tritium, ^{39}Ar would be a major effluent control problem. The blanket scores at the bottom for all the sensitivity cases studied. The major way to improve Nitrate Salt would be replacement of its sodium and potassium with something else, a form of elemental tailoring. A low-activation nitrate salt with good thermal stability would rank much better in the safety evaluation.

PCA versions of several of the HT-9 blankets were also given a Safety Evaluation. In all cases, the PCA version scored significantly lower than the HT-9 version, i.e., 7-8 SFM. The difference is caused by the higher radioactivity in PCA.

3.3.4 Conclusions

Given the reference assumptions that

(a) some tritium control ideas will work,

(b) air chemical reaction problems are largely solved by use of nitrogen cover gas, and

(c) water chemical reaction problems are largely solved by elimination of water-cooled limiters and adequate separation of water and reactive metals (Li,V),

then the top blankets are:

LiPb/LiPb/V

Li_2O/He/HT-9

Li/Li/V

Li/He/HT-9.

These top choices are a mixture of blankets that are especially attractive in terms of tritium control, i.e. elemental lithium-bearing, and those most attractive in terms of chemical reaction control, i.e. Li_2O/He/HT-9.

This is most easily seen by looking at two alternative sets of assumptions for sensitivity cases. First, if one believes that

(a) adequate tritium control is economically credible for all designs,

(b) cooling and pressure transients can be passively protected against, and

(c) chemical reaction problems (Li, 17Li83Pb, NS, V, Be-powder) are not solvable,

then the preferences are (in order):

Li_2O/He/HT-9 (appears on reference top list)

$LiAlO_2$/He/HT-9/Be

FLIBE/He/HT-9/Be;

and the blankets with either lithium or vanadium may not be acceptable. In other words, if passive control of lithium or vanadium reactions is not sufficient to effectively eliminate these accident concerns, then He/solid breeders/HT-9 concepts are the most attractive.

Second, if one believes that

(a) nonair, noncarbon dioxide building atmospheres or protection schemes are economically credible and solve air-metal chemical reaction problems,

(b) nonwater-cooled components are technically credible and are used to reduce water-metal chemical problems, and

(c) tritium control is extremely difficult,

then the preferences are (in order):

Li/Li/V (appears on reference top list)

Li/Li/HT-9

Li/He/HT-9 (appears on refere e top list),

and most other designs may not be acceptable. In other words, if tritium control of 17Li83Pb and solid breeder designs is not adequate to meet social safety standards, then elemental lithium-bearing designs are the most attractive.

Therefore, it is seen that the top blanket preferences depend on some optimism in tritium control and chemical reaction control. Pessimism in both areas produces an empty set of acceptable blanket choices. That is, the combination of tritium/effluent control and chemical reaction control concerns is a <u>fusion</u> feasibility issue. None of the blankets studied avoids both major problem areas. The Nitrate Salt blanket appears to avoid tritium control and lithium chemical reaction problems but has ^{39}Ar effluent and potential chemical decomposition problems. Given the current understanding and analysis of the impact of various uncertain issues, the most attractive blankets from the safety standpoint are

- Li_2O/He/HT-9 (<u>best</u> if tritium control better and chemical control worse than the reference case)

- Li/He/HT-9 (similar to Li_2O/He/HT-9 except for more difficult chemical control and easier tritium control--more of a compromise)

- o Li/Li/V (best if chemical control better and tritium control worse than the reference case)

- o LiPb/LiPb/V (similar to Li/Li/V except for more difficult tritium control, easier chemical control, and higher activation--more of a compromise).

Two of these are helium-cooled HT-9. Two are liquid-metal self-cooled V15Cr5Ti. The preferred breeder for helium-cooled HT-9 is Li_2O if tritium effluent control is favorable; a lithium-breeder would be the backup. The preferred liquid-metal for the liquid-metal-cooled V15Cr5Ti concept is lithium if air and water chemical reactions are adequately controlled; 17Li83Pb would be the backup. The Safety Evaluation results show that water and Nitrate Salt coolant and PCA structure score poorly relative to the other concepts studied. Finally, it should be mentioned that some blankets that could be extremely attractive from the safety standpoint, e.g. helium-cooled, Li_2O or Li_3SiO_3 breeder, SiC, were not examined in the BCSS.

3.4 Research and Development Concept Evaluation Results

The evaluation methodology for research and development (R&D) aspects of the BCSS blanket options is presented in Sec. 5.5. In summary, the methodology provides for an overall figure of merit (RDFM) which is given by

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

where RDR is a parameter that assesses the "risk" in carrying out the R&D for a particular blanket option and RDI is a 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.)

3.4.1 R&D Investment Evaluation Results

The R&D investment parameter (RDI) 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) (see Sec. 5.5.1) such that

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

Table 5.5-1 is reproduced here as Table 3.4-1 for convenience to the reader.

Table 3.4-1. RDI CATEGORIES

Time Scale	Score X_1	Average Annual Operating Cost	Score X_2	Required Facilities	Score X_3
< 10 years	(1)	< \$5M	(1)	No New Facilities > \$10M	(1)
10 years → 20 years	(2)	\$5-20M	(2)	New Facilities \$10-\$50M	(2)
> 20 years	(3)	> \$20M	(3)	New Facilities > \$50M	(3)

The results of the RDI evaluation are shown in Table 3.4-2. No significant difference was identified between tokamak and tandem mirror reactors.

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 ferritic steels could be developed to that point in less than 10 years, thus $X_1 = 1$. This assumed that the blanket would use an existing ferritic steel alloy such as HT-9. The development of a more advanced ferritic steel (e.g., a low activation improvement) would require more time. Vanadium alloy blankets would require 10 to 20 years, thus $X_1 = 2$. No blanket option was judged to take longer than 20 years, given adequate funding.

All blankets were judged to require annual operating R&D costs of >$5M. The blanket concepts employing Li and Li_2O without neutron multipliers were judged to be in the range of $5M to $20M per year, thus $X_2 = 2$. LiPb was judged to require more resources than Li. Similarly, ternary ceramics (TC) blankets with Be 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 ferritic steel blankets but to require a longer time as indicated on the X_1 scores.

TABLE 3.4-2. RDI EVALUATION RESULTS

Concept	X_1 Time Scale	X_2 Annual Cost	X_3 New Facilities	RDI R&D Investment
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 3.4-1 for definition of categories for X_1, X_2, and X_3.

The facility scores (X_3) are similar to the scores for X_2. The BCSS did not consider facility cost related to integrated testing in some type of fusion-based test reactor[a], except to note that integrated testing of blankets will be 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 fission reactor and ion irradiation 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 investment score (RDI) indicates that the ferritic steel 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 ternary ceramics with a Be neutron multiplier have somewhat less resource requirements than LiPb.

3.4.2 R&D Risk Evaluation Results

The R&D Risk parameter (RDR - see Sec. 5.5.2) 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). Thus, the ratings can range from 1 x 1 to 3 x 3. In practice only ratings of 1 x 2 or 2 x 1 or larger are retained.

The first step in this process by the BCSS was to develop an independent list of key issues for each blanket option. The spirit of this exercise was to develop a list of issues which had or could have a reasonably important impact on the blanket. There was an attempt to focus only on key issues. A total of 29 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 3.4-3) and each issue was given the $C_i \times P_i$ rating for each blanket concept to which it applies.

[a] This question is being addressed in detail by the FINESSE Project during FY1984-1985.

TABLE 3.4-3. BLANKET CONCEPT KEY ISSUES RATING
[R = C x P, TOK
TMR]

Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He FS/Be	$LiAlO_2$/H_2O/ FS/Be	$LiAlO_2$/NS/ FS/Be	FLIBE/He FS/Be
1. Unsatisfactory weld/fabrication of structural materials	3x2 3x2	-- 3x2	-- 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2
2. Excessive embrittlement of structure by hydrogen		--	-- 2x1		2x1 2x1	2x1 2x1	2x1 2x1	2x1 2x1	2x2 2x2
3. Unacceptable radiation-induced embrittlement of structure including DBTT concerns	3x1 3x1	-- 3x1	-- 3x1	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x1 3x1	3x2 3x2
4. V-blanket requires non-V balance of plant (BOP)	2x1 2x1	--	-- 2x1						
5. Risk from reactivity of structure with environment (inability to use inert atmosphere)	3x1 3x1	--	-- 3x1						
6. Risk from reactivity of coolant and breeder with environment (inability to use inert atmosphere)	3x1 3x1	-- 3x1	-- 2x1	2x1 2x1					
7. Corrosion worse than expected (includes non-V BOP in V designs)	2x2 2x2	-- 2x2	-- 2x2	2x2 2x2	1x2 1x2			2x2 2x2	2x1 2x1
8. MHD effects substantially worse	3x2 2x1	-- 2x1	-- 2x1	2x1					
9. Insulators not developed for liquid metal blanket	3x1 2x1	-- 2x1	-- 2x1	2x1					
10. Inability to develop non-water cooled near plasma components	3x2 3x1	-- 3x1	-- 2x1	2x2 2x1					
11. Difficult to meet seismic requirements		--	-- 2x2						

TABLE 3.4-3. BLANKET CONCEPT KEY ISSUES RATING (cont.)
[R = C x P, TOK
TMR]

	Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He FS/Be	$LiAlO_2$/H_2O/ FS/Be	$LiAlO_2$/NS/ FS/Be	FLIBE/He FS/Be
12.	Inadequate tritium breeding	2x1 2x1	-- 2x1	--	2x1 2x1	3x2 3x2	3x1 2x1	3x1 2x1	2x1	3x1
13.	Inability to accommodate breeder swelling		--	--		3x2 3x2	2x1 2x1	2x1 2x1	2x1 2x1	
14.	Temperature range for tritium release much less than predicted		--	--		3x2 3x2	2x2 2x2	2x2 2x2	2x2 2x2	
15.	Unacceptable temperature predictability of breeder (e.g., breeder-to-structure gaps are created)		--	--		3x1 3x1	3x1 3x1	3x2 3x2	3x1 3x1	
16.	Unacceptable power variation capability		--	--		2x1 2x1	2x1 2x1	2x2 2x2	2x1 2x1	
17.	Fabrication/refabrication of solid breeder		--	--		2x2 2x2	2x1 2x1	2x2 2x2	2x2 2x2	
18.	Tritium recovery/leakage/control worse than predicted		--	-- 3x2	3x1 3x1	3x2 3x2	3x2 3x2	3x2 3x2	3x2 3x2	3x3 3x3
19.	Loss of Be integrity is a major problem		--	--			1x3 1x3			1x2 1x2
20.	Inability to reprocess Be in efficient manner		--	--			2x1 2x1	2x2 2x2	2x2 2x2	2x1 2x1
21.	T-release from Be to primary coolant		--	--			2x2 2x2			2x2 2x2
22.	Excessive chemical reactivity of Be with salt		--	--					3x2 3x2	
23.	Salt stability/decomposition worse than predicted		--	--					3x2 3x2	

TABLE 3.4-3. BLANKET CONCEPT KEY ISSUES RATING (cont.)
[R = C x P, TOK
TMR]

	Key Issue	Li/Li/V	Li/Li/FS	LiPb/LiPb/V	Li/He/FS	Li_2O/He/FS	$LiAlO_2$/He FS/Be	$LiAlO_2$/H_2O/ FS/Be	$LiAlO_2$/NS/ FS/Be	FLIBE/He FS/Be
24.	Inadequate performance of grooved first wall (tokamaks only)	2x2	--	--	3x2	3x2	3x2	2x2	2x2	3x2
25.	Excessive coolant leakage to plasma	2x1 2x1	-- 2x1	-- 2x1	2x2 2x2	2x2 2x2	2x2 2x2	2x2 2x2		2x2 2x2
26.	Difficult coolant/breeder cleanup after spill		--	-- 2x2						2x2 2x2
27.	Coolant containment reliability of double tubed wall less than predicted		--	--				3x1 3x1		
28.	Excessive activation products, difficult to control		--	-- 2x2					2x3 2x3	
29.	Electromagnetic effects worse than assumed	3x1 2x1	-- 2x1	--	2x1	2x1	2x1	2x1	2x1	2x1
<u>TOTALS</u>	Tokamak	47	---	---	43	61	57	60	66	54
RDR	TMR	34	32	48	29	53	48	53	58	43

In general, the issues are grouped in Table 3.4-3 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, electromagnetic effects - 29) apply to almost all blanket concepts.

There are some important differences between tokamaks and tandem mirrors; thus an entry is made in Table 3.4-3 for both concepts (the upper entry is for tokamaks). The overall score (RDR) for each blanket concept is then a sum down the column of all the C_i x P_i values separately for blanket options for tokamaks and tandem mirrors. The results are shown at the bottom of Table 3.4-3.

The following is a brief summary of the rationale for the ratings for the various issues. (Chapter 4 contains a more detailed dicussion of each key issue.)

1. Unsatisfactory Weld/Fabrication of the Structural Materials

This issue could be of very serious consequence (C_i = 3) for both ferritic steel and vanadium alloy blankets. The probability was judged to be moderate, or about "50/50," so that P_i = 2. Obviously the issue is different for ferritic steel (mainly the need for complex post-weld heat treatments) and vanadium alloys (need for welding in an inert atmosphere).

2. Excessive Embrittlement of Structure by Hydrogen

Embrittlement of ferritic steel by hydrogen was judged to be of moderate consequence (C_i = 2) but low probability (P_i = 1) because the hydrogen partial pressures in most of the blankets would result in hydrogen concentrations in the ferritic steel below the threshold values for embrittlement (tens of ppm). The FLIBE blanket has a significantly higher hydrogen partial pressure and was thus rated P_i = 2. Vanadium was not rated because its threshold is an order of magnitude higher than that of ferritic steel.

3. Unacceptable Radiation-Induced Embrittlement of Structure Including DBTT Concerns

Radiation embrittlement would be of high consequence (C_i = 3) for all blankets. It was judged to be of moderate probability (P_i = 2) for ferritic steels and somewhat lower probability (P_i = 1) for vanadium because of the expected better radiation resistance of vanadium. The nitrate salt (NS) blanket was given a probability rating of 1 because it operates at a higher coolant inlet temperature and thus provides more margin with respect to DBTT problems.

4. V-Blanket Requires Non-V Balance of Plant (BOP)

The vanadium alloy blanket designs employ a ferritic steel for the piping of the heat transport system outside of the reactor, mainly for economic reasons. This could lead to problems of impurity transport between the ferritic pipes and vanadium blanket ducts. This was judged to be of moderate consequence and low probability.

5. and 6. Risk from Reactivity of Structure and Coolant/Breeder with Environment

These two issues were judged to be a concern for self-cooled Li and Li-Pb designs with vanadium structures. The consequence is high (C_i = 3) but the probability is low (P_i = 1) because of the use of inert atmospheres in the reactor building. It is also an issue for static lithium designs but with only a moderate consequence.

7. Corrosion Worse than Expected (Includes Non-V BOP in V Designs)

The consequence of corrosion worse than assumed in the BCSS was rated moderate (C_i = 2) for almost all designs and would most likely result in a reduction in allowed structure/coolant interface temperatures and thus a lower thermal efficiency. The probability was generally rated as moderate (P_i = 2); except for the FLIBE design which was rated low (P_i = 1). (Data with austenitic steel indicate very little corrosion with FLIBE; however, no data exists with ferritic steels.)

8. MHD Effects Substantially Worse

MHD effects more severe than anticipated in the BCSS would have a severe impact (C_i = 3) on the self-cooled lithium design for tokamaks; probability is judged to be moderate (P_i = 2). The self-cooled designs for tandem mirrors would have lower consequences (C_i = 2) and probabilities (P_i = 1) because of the lower magnetic fields and first wall heat fluxes. The "static" lithium design for the tokamak would also have a lower probability and consequence than does the self-cooled tokamak case.

9. Insulators Not Developed for Liquid Metal Blankets

Insulators were assumed to be used in the inlet and outlet regions of these designs, but not in the blankets themselves. Consequences would be moderate (C_i = 2) except for the self-cooled tokamak which would be high (C_i = 3). The probability of not being able to use such insulators was judged to be low (P_i = 1).

10. Inability to Develop Non-Water Cooled Near Plasma Components

If non-water cooled near-plasma components cannot be developed for self-cooled lithium designs, the consequence would be high (C_i = 3). The consequence for LiPb and "static" lithium blankets (which have an extra containment boundary compared to self-cooled designs) is rated C_i = 2. The probability of not being able to develop non-water coolants is related to the severity of the heat extraction issue; being rated P_i = 2 for tokamaks (e.g., limiters and divertor plates) and P_i = 1 for mirrors (e.g., choke coils, direct-convertor plates).

11. Difficult to Meet Seismic Requirements

This is an issue only for the LiPb blanket. It is judged to be of moderate consequence and probability.

12. Inadequate Tritium Breeding

The tritium breeding issue is most severe (C_i = 3) for the lithium oxide design and for the tokamak versions of the ternary ceramic (TC) and FLIBE designs which use beryllium neutron multipliers with water and helium coolants. These designs have very little margin for increasing the tritium breeding ratio (TBR). On the other hand, the consequence is judged to be moderate (C_i = 2) for all liquid metal blanket designs and for the TC tandem mirror designs. The probability is judged to be low (P_i = 1) for all designs except the lithium oxide design which is judged to be moderate (P_i = 2). The reason for this is that small increases in the structural volume fraction in the Li_2O design would likely reduce the TBR to less than 1. The TBR used in Section 3.2 for the $LiAlO_2$/He/Be/FS tokamak design is low because the design team opted not to place beryllium in the inside blanket. This could be corrected by adding beryllium in this region. The nitrate salt and FLIBE designs are not rated for the tandem mirror because of their relatively large TBR's.

13. Inability to Accommodate Breeder Swelling

Swelling of the solid breeder is judged to be a serious consequence (C_i = 3) in the lithium oxide design and is rated a moderate probability (P_i = 2). The ternary ceramics are expected to be more resistant to swelling and are therefore rated at a lower consequence and probability.

14. Temperature Range for Tritium Release Much Less Than Presently Predicted

If the temperature range acceptable for solid breeder operation is reduced (e.g., by radiation effects) this would be a serious consequence (C_i = 3) for the lithium oxide design. The consequence is judged to be moderate (C_i = 2) for the ternary ceramics because of a wider predicted temperature window. All solid breeder designs are rated with a probability of P_i = 2.

15. Unacceptable Temperature Predictability of Breeder (e.g., Breeder-to-Structure Gaps are Created)

Difficulty in obtaining the temperature distribution in a solid breeder (e.g., due to changes in the heat transfer between the breeder and coolant) would have a serious consequence on all solid breeder designs (C_i = 3). This is judged to be a low probability (P_i = 1) except for the water-cooled design which is rated P_i = 2 because of the lower coolant inlet temperature.

16. Unacceptable Power Variation Capability

Solid breeder designs are in general less able to operate at lower than rated nominal power, especially for the water-cooled designs. The consequence is rated as moderate (C_i = 2) and the probability is rated low (P_i = 1) except for the water-cooled design.

17. Fabrication/Refabrication of Solid Breeder

Difficulties in fabrication and refabrication (which will require remote reprocessing techniques) of the solid breeders are judged to have moderate consequence (C_i = 2). The probability of having difficulty in fabrication and refabrication is judged moderate for lithium oxide and for the sphere-pac form of lithium aluminate (P_i = 2); the probability is rated low (P_i = 1) for the sintered-product form of lithium oxide in the helium cooled design.

18. Tritium Recovery/Leakage/Control Worse Than Predicted

The consequences of a more severe problem regarding tritium recovery/leakage/control is rated severe (C_i = 3) on all blanket concepts except liquid lithium. This is due primarily to the relatively low tritium partial pressure in liquid lithium compared to either lithium lead or the solid breeders. The probability of this being an issue is generally moderate (C_i = 2) except for the FLIBE concept which has a significantly higher tritium partial pressure and is rated P_i = 3.

19. Loss of Be Integrity is a Major Problem

For the designs using either beryllium rods or large balls, loss of beryllium integrity is judged to be moderate to high probability but of a relatively low consequence. This is because the beryllium is not a structural material and -- in the case of rods -- probably could be easily canned.

20. Inability to Reprocess Be in an Efficient Manner

Difficulty in the efficient reprocessing of beryllium is judged to have a moderate consequence (C_i = 2). The probability is rated moderate for the water-cooled and nitrate salt designs because the Be is mixed in with the $LiAlO_2$ sphere-pac material. On the other hand, the probability is rated low for helium-cooled designs because the Be is separate from the $LiAlO_2$ or FLIBE as either rods or balls.

21. Tritium Release from Be to Primary Coolant

Tritium release from beryllium is a problem only for those designs in which beryllium is in direct contact with the primary helium coolant. It is judged to be of moderate consequence and moderate probability.

22. and 23. Excessive Chemical Reactivity Between Be and the Molten Salt and Molten Salt Decomposition

These issues obviously apply only to the nitrate salt design. Both issues are of potentially severe impact (C_i = 3) and are judged to be of moderate probability (P_i = 2).

24. Inadequate Performance of Grooved First Wall (Tokamaks Only)

This issue applies only to tokamaks because of the higher first wall surface heat flux compared to tandem mirrors. The issue rating depends primarily on the coolant; the consequence is condsidered severe for helium designs (C_i = 3) and moderate for the other coolants (C_i = 2) due to lower margins in the helium designs with respect to allowable first wall stresses and temperature gradients. The probability is judged to be moderate (P_i = 2) for all concepts.

25. Excessive Coolant Leakage to Plasma

The problem of first wall/blanket coolant leakage into the plasma chamber through small cracks is judged to be of moderate consequence (C_i = 2) for all concepts. The probability is judged to be moderate (P_i = 2) for helium and water cooled concepts. The self-cooled liquid metals were judged to be of low probability (P_i = 1) because of lower predicted leak rates for given small crack sizes. The nitrate salt cooled design was also rated a low probability due to its relatively low pressure.

26. Difficult Cleanup of Coolant or Breeder after Spill

Problems related to coolant or breeder cleanup were judged to be a key issue only for the LiPb and FLIBE salt designs; each was given a moderate consequence and probability rating. It was judged that lithium and nitrate salt spills could be cleaned up using water dissolution techniques.

27. Coolant Containment Reliability of Double Walled Tube Less Than Predicted

The reliability of coolant containment is a special issue for high pressure water cooled designs due to the very large number of small tubes in the blanket. The consequence is high but the probability is low because of reasonable confidence that double walled tubes would solve the problem.

28. Excessive Activation Products, Difficult to Control

Activation product release, primarily due to gaseous radioactivity mixed with tritium spills and leakage, is a key issue only for LiPb and nitrate salt (NS) designs. The consequence is judged to be moderate but with a high probability for the NS design.

29. Electromagnetic Effects Worse than Assumed

Electromagnetic effects (EM) due to disruptions or rapid plasma loss are important in all tokamak designs and in tandem mirrors with liquid lithium (LiPb is not a major concern due to its higher resistivity compared to Li). The probabilities are rated low in all cases because of the expectation that the time scales for disruption/plasma losses will be long enough (~ 100 msec) compared to the blanket characteristic EM time scale. The consequence is moderate in solid breeder blankets and in tandem mirrors with liquid lithium; the consequence could be high in tokamaks with self-cooled liquid lithium blankets.

Summary

The Li/He/FS blanket represents the minimum R&D risk (lowest total of Σ C x P) for both tokamaks and tandem mirrors. The Li/Li/FS and Li/Li/V designs are the next lowest risk designs for mirrors while Li/Li/V is the second lowest risk design for tokamaks. The highest risk designs are the TC concepts with Be multipliers, particularly with a nitrate salt coolant, for both tandem mirrors and tokamaks.

3.4.3 Composite R&D Evaluation Results

The composite score for tokamaks is shown in Table 3.4-4 and for tandem mirrors in Table 3.4-5. The relative ranking of the blanket concepts is also indicated in each table. It is clear that the risk factor $[\Sigma (C \times P)]^{-1}$ is more important in determining the overall R&D evaluation than the investment factor (RDI).

The Li/He/FS ranks number one for both tokamaks and tandem mirrors, i.e., this blanket has the best overall combination of minimum R&D resource requirements and minimum risk. The Li/Li/FS and Li/Li/V designs rank two and three for tandem mirrors while the Li/Li/V and Li_2O/He/FS ranks two and three for tokamaks. All the ternary ceramic designs rank low with the nitrate salt ranking last in both tokamaks and tandem mirrors. The LiPb/V design for mirrors also ranks very low. Basically liquid lithium does well because the designs are somewhat simpler compared to solid breeder designs and the data base for lithium is much better compared to LiPb or solid ceramics. Solid breeder designs with ternary ceramics have added risks due to the need for Be.

TABLE 3.4-4 R&D EVALUATION FOR TOKAMAKS

Blanket Concept	$\text{RDFM} = \frac{30}{\sum_i C_i \times P_i}$ $\sum_i C_i \times P_i$ (1)	$+ \frac{1}{\text{RDI}}$ RDI (2)	RDFM	Rank
Li/Li/V	47	2.0	1.14	2
-	-	-	-	
-	-	-	-	
Li/He/FS	43	1.7	1.29	1
Li_2O/He/FS	61	1.7	1.08	3
TC/He/FS/Be	57	2.3	0.97	5
TC/H_2O/FS/Be	60	2.3	0.93	6
TC/NS/FS/Be	66	2.3	0.89	7
FLIBE/He/FS/Be	54	2.0	1.06	4

(1) From Table 3.4-3

(2) From Table 3.4-2

TABLE 3.4-5 R&D EVALUATION FOR MIRRORS

$$RDFM = \frac{30}{\sum_i C_i \times P_i} + \frac{1}{RDI}$$

Blanket Concept	$\sum_i C_i \times P_i$ (1)	RDI (2)	RDFM	Rank
Li/Li/V	34	2.0	1.38	3
Li/Li/FS	32	1.7	1.52	2
LiPb/LiPb/V	48	2.7	1.00	8
Li/He/FS	29	1.7	1.62	1
Li_2O/He/FS	53	1.7	1.16	5
TC/He/FS/Be	48	2.3	1.06	6
TC/H_2O/FS/Be	53	2.3	1.00	7
TC/NS/FS/Be	58	2.3	0.95	9
FLIBE/He/FS/Be	43	2.0	1.20	4

(1) From Table 3.4-3

(2) From Table 3.4-2

3.5 Overall Evaluation

The previous sections present the results of the engineering feasibility, economic, safety, and R&D evaluations. Tables 3.5-1 and 3.5-2 summarize the relative ratings of the tokamak and tandem mirror 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 (Tables 3.5-1 and 3.5-2). 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. However, because performance cannot be guaranteed for any of the blankets, 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 3.5-1 and 3.5-2 to identify the other top rated blanket concepts that should provide a focus for the R&D effort.

- 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.

- 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.

- Numerical averaging of the individual evaluations with either the three factor or four factor approaches above, but with nonuniform weighting factors. Two specific weighting factors proposed and con

TABLE 3.5-1.
TOKAMAK BLANKET RANKING
SEPTEMBER 10, 1984

	Engineering	Economics	Safety	R&D
Li/Li/V 1.000	.85	.998	.886	
Li/Li/FS				
LiPb/LiPb/V				
Li/He/FS .750	.73	.925	1.000	
Li_2O/He/FS	.719	.79	1.000	.840
$LiAlO_2$/He/FS/Be	.611	.79	.904	.754
$LiAlO_2$/H_2O/FS/Be	.682	1.00	.597	.723
$LiAlO_2$/NS/FS/Be	.849	.98	.515	.692
FLIBE/He/FS/Be	.658	.84	.807	.824

TABLE 3.5-2.
TMR BLANKET RANKING
SEPTEMBER 10, 1984

	Engineering	Economics	Safety	R&D
Li/Li/V .999	.92	.974	.852	
Li/Li/FS .917	.94	.832	.944	
LiPb/LiPb/V	.959	.90	1.000	.617
Li/He/FS .939	.85	.936	1.000	
Li_2O/He/FS	.927	.89	.987	.716
$LiAlO_2$/He/FS/Be	.884	.90	.905	.654
$LiAlO_2$/H_2O/FS/Be	.811	.94	.595	.617
$LiAlO_2$/NS/FS/Be	1.000	1.00	.552	.586
FLIBE/He/FS/Be	.994	.92	.782	.741

sidered were 30-30-30-10 and 25-50-25-0 for the engineering feasibility, economic, safety, and R&D evaluations, respectively.

- A more qualitative comparison of the evaluation results including either the three factor 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 comparative results obtained by the various methods are summarized in Tables 3.5-3 and 3.5-4. The three-factor approach that provides equal weighting to the engineering feasibility, economic, and safety evaluations is generally favored. The results obtained by this method of comparison with some additional judgemental considerations have been used to identify the four leading blanket concepts that should provide the focus for the R&D effort (Table 3.5-5).

Clearly the Li/Li/V concept is the top rated blanket in this study. In the three factor approach used as a basis, the Li/Li/V concept rates far 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 non-water-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 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

TABLE 3.5-3. COMPARISON OF OVERALL TOKAMAK EVALUATIONS OBTAINED BY VARIOUS METHODS: RATING (RANK)

	THREE FACTOR APPROACH (Eng. Feasibility, Economics, Safety)			FOUR FACTOR APPROACH (Eng. Feasibility Economics, Safety, R&D)		
	Numerical Average			Numerical Average		
	33-33-33-0 (Reference)	25-50-25-0	Qualitative	25-25-25-25	30-30-30-10	Qualitatative
Li/Li/V	1.000 (1)	1.000 (1)	2.5 (1)	1.000 (1)	1.000 (1)	3.0 (1)
Li_2O/He/FS	.878 (2)	.892 (3)	1.0 (2)	.897 (3)	.894 (2)	1.0 (2)
Li/He/Fs	.842 (3)	.848 (6)	-0.5 (5)[a]	.912 (2)	.873 (3)	0.5 (3)
$LiAlO_2$/NS/FS/Be	.831 (4)	.898 (2)	0.5 (3)	.813 (6)	.818 (6)	-0.5 (5)
FLIBE/He/FS/Be	.809 (5)	.850 (5)	0 (4)	.838 (4)	.822 (4)	0 (4)
$LiAlO_2$/He/FS/Be	.806 (6)	.837 (7)	-0.5 (5)[a]	.819 (5)	.821 (5)	-1.0 (6)
$LiAlO_2$/H_2O/FS/Be	.805 (7)	.886 (4)	-0.5 (5)[a]	.804 (7)	.802 (7)	-1.5 (7)

[a]More than one concept received same score.

TABLE 3.5-4. COMPARISON OF OVERALL TANDEM MIRROR EVALUATIONS OBTAINED BY VARIOUS METHODS: RATING (RANK)

	THREE FACTOR APPROACH (Eng. Feasibility, Economics, Safety)			FOUR FACTOR APPROACH (Eng. Feasibility Economics, Safety, R&D)		
	Numerical Average			Numerical Average		
	33-33-33-0 (Reference)	25-50-25-0	Qualitative	25-25-25-25	30-30-30-10	Qualitatative
Li/Li/V	1.000 (1)	1.000 (1)	2.5 (1)	1.000 (1)	1.000 (1)	3.0 (1)
LiPb/LiPb/V	.982 (2)	.986 (2)	1.5 (2)[a]	.928 (5)	.966 (2)	0.5 (6)
Li_2O/He/FS	.970 (3)	.969 (3)	1.5 (2)[a]	.940 (4)	.958 (4)	2.5 (2)
Li/He/FS	.943 (4)	.938 (7)	0.5 (7)[a]	.995 (2)	.965 (3)	1.5 (4)[a]
$LiAlO_2$/He/FS/Be	.927 (5)	.941 (6)	0.5 (7)[a]	.893 (7)	.916 (7)	0 (7)[a]
FLIBE/He/FS/Be	.924 (6)	.948 (5)	1.5 (2)[a]	.918 (6)	.929 (6)	1.5 (4)[a]
Li/Li/FS	.922 (7)	.952 (4)	1.0 (5)[a]	.970 (3)	.946 (5)	2.0 (3)
$LiAlO_2$/NS/FS/Be	.863 (8)	.932 (8)	1.0 (5)[a]	.838 (8)	.824 (8)	0 (7)[a]
$LiAlO_2$/H_2O/FS/Be	.793 (9)	.862 (9)	-0.5 (9)	.791 (9)	.806 (9)	-1.5 (9)

[a]More than one concept received same score.

TABLE 3.5-5. R&D IMPLICATIONS FOR LEADING BLANKET CONCEPTS

Concept	Comments	R&D/Feasibility Implications
Li/Li/V	Top Concept in both Tokamak and TMR	• Development of advanced structural alloy • Resolution of MHD and corrosion problems • Nitrogen environment provides adequate safety • Development of non-water-cooled limiter/divertor
LiPb/LiPb/V	High Rating in TMR Only	• Development of advanced structural alloy • Resolution of MHD and corrosion problems (more severe than for Li/Li/V) + Reduced concern for reactivity with environment • Resolution of tritium recovery/containment issue
Li_2O/He/FS	High Rating in Both Tokamak and TMR	• Resolve tritium recovery/containment issues • Resolve tritium breeding capability • Resolve solid breeder issues (swelling) + Concept avoids MHD issues
Li/He/FS	High Rating in Tokamak Good Rating in TMR	• Improve economic performance • Resolve structural material issues + reduced MHD problem compared to self-cooled concepts + reduced tritium recovery/containment compared to solid breeder + avoids solid breeder issues (e.g., swelling)

factor. Although these two liquid metal systems are quite similar with respect to key issues, important differences relate 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 judgemental 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 either 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 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:

- Each of these four concepts has a unique set of key issues. Therefore, serious negative results associated with the key feasibility issues will mostly likely apply to no more than two concepts, leaving two potentially viable options. Table 3.5-5 provides a summary of the R&D implications associated with each concept.

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

- 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.

- 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.

BLANKET COMPARISON AND SELECTION STUDY

CHAPTER 4 - R&D ASSESSMENT

CHAPTER 4

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4. R&D ASSESSMENT

Introduction

An assessment was carried out of the key R&D issues of the leading blanket concepts evaluated in the BCSS. The approach was to identify the key issues which would affect the feasibility and basic performance for each of the seven tokamak and nine tandem mirror blankets. A total of 29 issues were identified; these were combined into a single table (Table 3.4-3) which indicates the severity of the issue (in terms of a consequence times probability rating) as well as how the issue applies to the different blanket options.

This chapter contains a more detailed description of each issue (see Sections 4.1 to 4.29) in a common format as follows:

- -- description of issue,
- -- required data,
- -- status of data base,
- -- required resources.

No attempt was made to estimate the required total resources to completely resolve all the issues for the various blanket options. Rather the discussion of "required resources" focuses on the resources required to judge the feasibility of resolving the issue. Furthermore, the discussion of resources focuses on approximately the next five years. No attempt was made to formally rank the R&D issues in terms of their relative priority. However, a sense of the relative priority of the issues can be gleaned by examining Table 3.4-3. For example, the C x P rating indicates the severity of the issue for a particular blanket concept (vertical columns) while the number of entries horizontally across the table indicates how many blanket concepts are affected by a particular issue.

Based on the approach outlined above, the following general observations can be made:

- The most important structural material R&D issues are welding/fabrication and radiation induced embrittlement concerns for both ferritic steels and vanadium alloys. Chemical reactivity of vanadium is also an important issue.

- Major issues for liquid metal blankets include MHD effects and corrosion concerns. MHD research should include the testing of insulators, particularly for tokamak applications. Lithium (and to some extent LiPb) chemical reactivity is a key issue. Development of non-water cooled near-plasma components will be necessary, particularly for tokamak blankets that contain lithium.

- Tritium recovery/control is a major issue for all breeders except those using liquid lithium as a breeder and coolant. The form of the released tritium (T_2/HT or T_2O/HTO) and the chemical form of tritium in various fluid streams are important issues for tritium control for solid breeders.

- Tritium breeding is a key issue because of uncertainties in reactor configuration, nuclear data and tritium system performance. It is a particularly critical issue for blankets with a Li_2O breeder and no neutron multiplier. In general, adequate tritium breeding is of greater concern for tokamaks than for tandem mirrors. Tritium breeding is of the least concern for LiPb blankets and for blankets that have a sufficiently high ratio of beryllium to structure.

- The key issues for solid breeders (in addition to those discussed above) include the temperature limits for tritium release, heat transfer control between the lithium ceramic and coolant, difficulty of handling power variations and the radiation induced swelling of the ceramic (particularly Li_2O). Initial fabrication of sphere-pac breeder and beryllium and refabrication of all forms by remote handling techniques are also areas of concern. The BCSS has emphasized Li_2O and $LiAlO_2$.

- The most important concern related to first wall issues is the verification of the capability of a stress relief structure (orthogonally grooved first wall) for tokamaks to handle simultaneously heat and particle fluxes.

- Additional items include the thermal and radiation stability of molten salts; Be reprocessing efficiency; Be chemical interaction with molten salts; activation of LiPb and molten salts; and electromagnetic effects in tokamaks such as large pressures and torques due to plasma disruptions.

4.1 Welding/Fabrication of Structural Alloys

4.1.1 Issue

The serious consequences of leaks or failures in the weld regions of the blanket structure require that reliable welds be achieved. The complex geometries also result in special fabrication issues for the structural materials. Special, but different, weld procedures are required for ferritic steels (HT-9/9Cr-1Mo) and vanadium base alloys. Pre- and post-weld heat treatments are necessary to provide satisfactory weld performance for the ferritic steels. Although there is considerable experience with welding ferritic steels, the complex configuration of fusion blankets may make controlled heat treatments difficult. Compositional and microstructural constraints dictated by radiation damage considerations may further exacerbate the problem. Welding/ fabrication experience with vanadium alloys is very limited. Inert or vacuum environments are required for welding vanadium base alloys. Further work is required to demonstrate that highly reliable welds can be routinely attained with both materials for the complex geometries of a fusion reactor blanket. A special capability is required to provide high purity heats of selected vanadium alloys and ferritic steels for experimental programs.

4.1.2 Required Data

Further effort is required for ferritic steels to assure that satisfactory weld reliability is attainable for alloy compositions, thermal-mechanical treatments, and microstructures that provide adequate radiation damage resistance and mechanical properties. Weld procedures that are compatible with weld joint and blanket geometry must be verified. Specifically, any limits on heat treatment imposed by the unique blanket geometries must be identified. The primary focus of this effort is not whether ferritic steels can be welded but to demonstrate that acceptable weld performance can be routinely attained for the appropriate alloy microstructures and the complex geometries anticipated. Because of the known sensitivity of composition and heat treatment to the weld characteristics of ferritic steels considerable effort will be required to demonstrate satisfactory weld performance of the low activation versions of the ferritic steels.

Further effort is required to establish which vanadium alloys can be readily welded and to develop optimum weld procedures for these alloys. Any alloy that cannot be readily welded should be eliminated from other parts of the development program. Investigations should determine the required purity of the weld environment (for Argon and vacuum), limits on weld geometry (0.5 mm to 2 cm thickness), and weld parameters for TIG and EB methods. The microstructures, extent of contamination, and weld properties (DBTT) for alloy compositions of interest (V-Cr-Ti alloys) should be evaluated. Any special treatments, e.g., pre- or post-weld heat treatments, required for satisfactory welds should be identified for candidate alloy systems. This effort should include an assessment of the feasibility of applying the specified weld procedures to the proposed blanket geometries.

4.1.3 Status of Data Base

The experience with welding ferritic steels is fairly extensive. The ferritic steels are known to be susceptible to cracking unless weld procedures are carefully controlled. Pre- and post-weld heat treatments have proven effective in reducing this susceptibility for cracking. The optimum post-weld heat treatments are generally dependent on the composition, the thermal-mechanical treatment of the alloy and the geometry of the weld (thickness, etc.). The microstructures that are more weldable are not the most radiation damage resistant. Current experience indicates that the duplex microstructures are more weldable while the pure martensitic microstructure provides better high temperature strength and better radiation damage resistance. The Alloy Development for Irradiation Performance (ADIP) program has developed recommended welding procedures for fusion applications of HT-9 ferritic steels. The Ferritic Steel Task of the Alloy Development for Irradiation Performance (ADIP) program has developed recommended welding procedures for fusion application of HT-9 ferritic steel. No data have been reported for the low activation versions of the ferritic steels.

The welding characteristics of vanadium have, in general, been evaluated only in limited exploratory investigations. Vanadium is known to be very reactive with oxygen and air at elevated temperatures. Therefore, all welding must be conducted under vacuum or inert environments. The limited data indicate that certain alloys are readily welded under controlled conditions while other alloys exhibit significant grain growth. Satisfactory weld microstructures with adequate ductility have been obtained in some alloys under limited test conditions. However, further effort is required to establish which alloys can be reliably welded and to develop optimum weld procedures for these alloys.

4.1.4 Required Resources

The required resources for this effort are about 4 my-y for 4 - 5 years. Half of the effort would be for ferritic steels and half for vanadium alloys. The ferritic steel support would provide a modest effort (~ 1/4) on welding of existing alloys (HT-9 and 9Cr-1Mo) with the remainder focused on the preparation of low activation alloys in the early phases and weld characterization in the later stages. The effort on vanadium alloys would be split between alloy preparation and welding.

4.2 Embrittlement of Structural Materials by Hydrogen

4.2.1 Issue

Many structural alloys exhibit a large reduction in ductility when the hydrogen concentration exceeds a certain level. Both ferritic steels and vanadium alloys are known to be susceptible to this hydrogen embrittlement under certain conditions. The extent of embrittlement depends on a number of factors including temperature, hydrogen pressure/concentration, and alloy microstructure, e.g., weld or heat affected zones. Hydrogen or tritium partial pressures will vary greatly depending on the blanket system. Energetic deuterium and tritium ions from the plasma will also be injected into the first wall. The projected hydrogen isotope (H, D, or T) concentrations in some blanket concepts may be sufficient to compromise the integrity of the structure. Therefore, the conditions for excessive hydrogen embrittlement must be more precisely defined for the candidate alloys.

4.2.2 Required Data

The dependence of temperature and hydrogen concentration on the ductility of ferritic steel weldments must be more precisely determined. Particular attention should be given to effects of weld method, post weld heat treatment, compositional variations, and temperature history. Later stages of this effort should evaluate potential syngeristic effects of hydrogen and radiation induced defects. The tests should concentrate on the lower temperature range of interest (25–300°C).

Scoping tests should be conducted on vanadium alloy weldments to evaluate the potential for enhanced embrittlement. This effort should focus on compositional effects relating to higher titanium concentrations and the lower operating range of interest (< 400°C).

4.2.3 Status of Data Base

The conditions for hydrogen embrittlement of base-metal ferritic steels and vanadium alloys are reasonably well known. The major uncertainties relate primarily to weld regions, which are known to be more susceptible to embrittlement, and to possible synergistic effects that result from radiation induced defects. Enhanced cracking of weld regions containing only a few weight parts per million hydrogen has been observed in ferritic steels. The hydrogen concentrations in vanadium will be much higher than in ferritic steels at the same partial pressure; however, vanadium alloys are less susceptible to embrittlement because of their very high hydrogen solubility. Hydrogen concentrations of ~ 10^{-3} wppm (> 100 Pa) are required to embrittle vanadium. Insufficient data exist on vanadium alloy weldments to establish the importance of this problem.

4.2.4 Required Resources

Only a modest effort is required to resolve these issues. A 1 my-y effort integrated into other phases of the alloy development program should provide an adequate data base.

4.3 Radiation-Induced Embrittlement of the Structural Alloys

4.3.1 Issue

The mechanical properties of structural materials are, in most cases, significantly affected by neutron radiation. A major concern is the loss of ductility or fracture toughness of the ferritic steel or vanadium alloy to the extent that the mechanical integrity of the structure would be compromised under projected operating conditions. The potential embrittlement problems include: (1) increases in the ductile-to-brittle transition temperature (DBTT), (2) radiation hardening, and (3) embrittlement caused by helium generation/segregation. The extent of embrittlement or loss of fracture toughness is dependent on temperature, alloy composition, microstructure, and neutron fluence. The severity of this problem is currently perceived to be more serious for the ferritic steel than for the vanadium alloy. However, the data base for vanadium is more limited.

4.3.2 Required Data

Further effort is required to determine what factors contribute to loss of ductility/fracture toughness of ferritic steels and vanadium alloys. Specific data is required to determine the compositional and microstructural variations (including weld regions) that are more resistant to the embrittling phenomena. The factors affecting the increase in the DBTT should be given high priority since these tests can be done at nominal fluences (< 20 dpa). The temperature range of interest is 25-300°C. The impact of helium must also be evaluated.

The temperatures at which helium embrittlement become severe need to be determined since, in many cases, this will set the maximum operating temperature. These tests should focus on temperatures of 0.4-0.6 T_m and should include weldments. Higher damage levels (> 100 dpa) are required for these tests. The residual ductility or fracture toughness at intermediate operating temperatures (0.3-0.5 T_m) must be more accurately determined since this may affect the design constraints and performance characteristics of the alloys, particularly during possible thermal transient conditions. These tests should focus on irradiations with He/dpa ratios of 5-10 and damage levels > 50 dpa. Compositions and microstructures that provide higher ductility/fracture toughness need to be identified.

4.3.3 Status of Data Base

Sufficient data have been generated from fission reactor irradiations to clearly show that severe embrittlement and loss of fracture toughness occurs in most structural alloys under certain conditions. Significant increases (above room temperature) in the DBTT have been observed in several neutron irradiated ferritic steels. The extent of this increase, which is dependent on microstructure and composition, normally saturates at relatively low fluences ($\leq$ 10–30 dpa). The performance of weld metal and weld heat-affected zones generally differs from the base metal. The extent of the reduction of fracture toughness is less-well defined but is generally observed over the projected operating temperature range. Although the presence of helium generated by the high energy neutrons is expected to exacerbate these two effects, only limited data exist. The phenomenon of helium embrittlement is generally observed at higher temperatures (typically > 0.5 T_m) where helium segregation occurs more readily. This effect is dependent on the helium concentration and is, therefore, fluence dependent.

Although the data for vanadium alloys are more limited, available results indicate that the problem for selected vanadium alloys may be less critical than for the ferritic steels. Earlier tests indicated significant residual ductility after irradiation. In more recent tests where larger effects have been observed, it is not clear whether the embrittlement is caused by radiation or chemical effects, viz., sulfur, since the material used was less pure. In either case, additional effort is required to better define the extent and conditions under which radiation induced embrittlement occurs.

4.3.4 Required Resources

A major effort will be required to resolve these issues. Since a high flux, high volume 14 MeV neutron source does not exist, various test methods will have to be used to obtain the desired data. In addition to a manpower level of at least 5 my-y for over 5 years, substantial facility or neutron costs will be required.

4.5 Risk from Reactivity of Structure with Environment (Inability to Use Inert Atmosphere)

and

4.6 Risk from Reactivity of Coolant and Breeder with Environment (Inability to Use Inert Atmosphere)

4.5.1 Issue

Vanadium and lithium, and to a lesser degree 17Li-83Pb, are reactive to water, air, and concrete. If serious consequences can be anticipated from a particular materials combination, even under accident conditions, then design modifications to avoid those combinations of the materials may be necessary. At present, data on these reactions are insufficient to model realistic accidents.

4.5.2 Required Data

Data on thermal and pressure transients for these reactive materials are required for realistic operating conditions. The temperature excursions, pressure buildup, and material phase changes for possible accident scenarios will have to be determined and the impact on components, the reactor vessel and reactor building have to be evaluated. The effects of radiation release and poison gas release to the public, if any, have to be determined.

Moderate to large scale reactivity experiments are required to simulate the possible accident conditions. Important scale parameters which will affect reactions are: surface to volume ratios, initial temperature, mode and rate of heat transport away from the reaction, quantity of air or water available and moisture content of air, and degree of agitation. The importance of the parameters can be seen from test results with lithium mass-to-area ratios of 50 kg/m^2 and 215 kg/m^2, which resulted in temperature increase rates of 100°C/min and 10°C/min, respectively.

If an accident consequence is serious and not acceptable, one must determine if design modifications can be made to alleviate the problem. Because the limiter/divertor, which is a major source of high temperature and high pressure water, is a major concern, studies and experiments to verify the applicability of either gas or liquid metal cooled limiters/divertors are needed.

In the BCSS it was assumed that N_2-Li, N_2-LiPb and N_2-V reactions pose little risk and thus a nitrogen cover gas was specified for all liquid lithium and vanadium designs. Further experiments are needed to verify this.

4.5.3 Status of Data Base

Temperature excursions, pressure buildup, and hydrogen release rates have been studied between lithium and 17Li-83Pb with air, water, concrete, etc. under the direction of the Fusion Safety Program. The experiments are on a small to moderate scale. The interaction of 17Li-83Pb and air/water has also been studied by ANL and Ispra. Reaction kinetics studies between N_2-Li and H_2O-17Li-83Pb have been initiated at MIT and U. of Wis., respectively. Air interaction with V alloys has been performed at EG&G.

4.5.4 Required Resources

Moderate to large scale accidents have to be studied under realistic conditions and these results used to determine the response of the rest of the reactor to determine the overall consequences of the accident. Alternative design concepts have to be developed and verified.

It is estimated that 3 years will be needed with 3 m-y/y to carry out the experiments and analysis to study the lithium and lithium-lead accident situations, 3 more years with 2 m-y/y will be needed to determine the consequence of the accident to the rest of the reactor. It is not possible to determine the efforts required to formulate and verify the applicability of an alternative design. Similar resources would be needed to study the vanadium chemical reaction issues.

4.7 Liquid Metal Corrosion

4.7.1 Issue

Liquid metal corrosion is an important consideration in large heat transport systems. The two major compatibility concerns arising from the use of liquid metals are: (1) corrosion/mass transfer and (2) degradation of the mechanical properties of the containment. The projected corrosion rates of candidate structural materials for Li and LiPb are orders of magnitude higher than for sodium at the same temperature. The corrosion rates depend on alloy composition, velocity, temperature, temperature variation, impurity level and velocity profile (which may be dominated by MHD effects). In most cases the wall thinning is not the limiting process. The problems associated with corrosion product transport and deposition will most likely limit the allowable corrosion rates. Corrosion/mass transfer limits provide the most restrictive operating temperature constraints for several concepts. Non-metallic element and dissimilar metal (ferritic steel ex-reactor) are the primary concerns for vanadium alloys systems. Low temperature liquid metal embrittlement is a concern for LiPb/FS systems.

4.7.2 Required Data

The corrosion rates and deposition mechanisms must be more accurately determined for variables of temperature, temperature gradient, velocity, impurity levels and magnetic interaction parameters for systems of interest. The priorities among these groups are to investigate temperature, temperature gradient, and impurity effects and then to investigate the effects of velocity and magnetic fields. Investigations on the ferritic steel with lithium and LiPb should include tests (> 2000 h) to 600°C with emphasis on velocity (0.1 - 5 m/s) and deposition kinetics. The effort should include investigations of low temperature (235 - 300°C) liquid metal embrittlement of ferritic steels in LiPb. For the vanadium-alloy in lithium and LiPb, the more important problem is the transport of impurities and its effect on the mechanical properties of the structural material. Vanadium alloy tests at temperatures of 400 - 750°C are needed. These tests should include bimetallic systems to provide data relevant to the use of ferritic steels outside the blanket. In all cases control and monitoring of impurity elements in the liquid metal is important. Nitrogen in lithium should be controlled at levels below 100 ppm. Eventually dynamic corrosion tests in a magnetic field (> 1 T) will be required to evaluate velocity profile effects. As more data become available for guidance, corrosion inhibition and corrosion product removal will be investigated.

4.7.3 Status of Data Base

Only a limited data base exists for corrosion of candidate structural alloys in lithium. Much of these data have been obtained in capsule tests and thermal convection loops. In most cases impurity element concentrations have not been well characterized and the effects of velocity have not been adequately determined. The temperature dependence for corrosion of austenitic and ferritic steels has been determined over a limited range. Data indicate that nitrogen in lithium affects both the corrosion rate and the fatigue properties of iron-based alloys. There is some experimental evidence that the addition of aluminum in lithium will inhibit corrosion, possibly due to the reduction of the nitrogen activity. Deposition and corrosion product cleanup has received very little study. The effects of magnetic fields on corrosion have been evaluated analytically, but have not been investigated experimentally.

Preliminary results on the corrosion of austenitic and ferritic steels in LiPb have recently been reported. These results indicate that corrosion rates are much greater (by an order of magnitude) in LiPb compared to lithium. The temperature dependence has been evaluated over only a limited range and the effects of impurities have not been established. Also, no results have been reported on velocity effects, inhibition, deposition, corrosion product cleanup, and magnetic field effects. The potential for liquid metal embrittlement of ferritic steels in LiPb has not been adequately addressed.

Limited data indicate that vanadium is highly resistant to attack by lithium; however, the effects of impurity element transfer have not been established. Experiments have not been performed to evaluate the effects of bimetallic, e.g., vanadium-ferritic steel, systems.

4.7.4 Required Resources

There are a limited number of corrosion loops available in the U.S. These facilities have been designed primarily for investigating effects of temperature and impurities on the corrosive attack of austenitic and ferritic steels in Li or LiPb. However, these facilities have not been designed to study corrosion product deposition mechanisms, corrosion product trapping, or the effects of magnetic fields.

Additional facilities will be required to study the corrosion of iron-based alloys by either lithium or 17Li-83Pb at temperatures to 600°C. Emphasis will be placed on facilities for investigation of the influence of velocity (0.1 - 5 m/s), temperature gradients (Δt = 50 - 200°C), and liquid metal chemistry. Both corrosion and deposition kinetics will be evaluated. Similar facilities with the added capability for evaluating the effects of a magnetic field on the corrosion/mass transport processes will also be required. Additional capability will be required to evaluate environmental effects on the mechanical properties of structural alloys, e.g., liquid metal embrittlement of ferritic steels in LiPb. It is estimated that 5 years is required to study this phase with an average manpower of 6 m-y/y.

Another type of facility will be required to study the effects of impurities on the corrosion and mechanical properties of vanadium in either Li or 17Li-83Pb. This facility will include a vanadium test section with ferritic steel in the balance of the loop. It is estimated that 5 years is required for this effort with a manpower of 4 m-y/y.

4.8 MHD Effects Substantially Worse
and
4.9 Insulators Not Developed for Liquid Metal Blanket

4.8.1 Issue

The effects of high magnetic fields on heat, mass, and momentum transfer in conducting fluids are not clear at this time. The theory for fully developed flow in circular and rectangular straight ducts is fairly well understood and, although experimental confirmation of the theory has been limited to Hartmann numbers of $M < 500$, the theory is expected to apply to the reactor region of $M \simeq 10^5$. Expansion of the theory and development of calculational techniques to treat bends and field gradients and the complex configurations in fusion reactors is far from straightforward. If, as our understanding improves, presently conceived designs appear infeasible, then solutions such as electrically insulated walls may be necessary.

4.8.2 Required Data

The uncertainties of MHD are centered around 3-D effects that result from the variations of $\vec{B}$, $\vec{V}$, cross-sectional area, wall (electrical) conductance and interactions with adjoining channels. All of these variations will probably be present in fusion reactors. Anticipated 3-D effects include increased pressure drop and modified velocity profiles. The latter may cause severe problems in both heat transfer (first wall cooling) and mass transfer (corrosion). Thus, the primary focus for initial investigations should be experimental data and calculational techniques that can be used to describe and predict local velocities in typical "3-D configurations".

The most important parameters in MHD are the Hartmann number, M, and the interaction number, N. In order to be extrapolated to reactor conditions, experiments should be designed with M and N between 10^3 and 10^4. Initially, the flow configuration can be simple in order to understand the problem and to check computer models. However, realistic configurations should be used later in the experimental program. It will also be necessary to eventually examine transient effects of time varying magnetic fields.

Development of insulated walls may also be desirable or necessary. Some insulators (e.g., Y_2O_3) may have potential as coatings in direct contact with liquid metal. Parallel development of fabrication methods for sandwich-type walls would also be prudent.

4.8.3 Status of Data Base

As can be seen in Fig. 4-1, the few available data are located around both M and N ~ 100. A liquid metal MHD loop is being constructed at ANL which will be operated with M and N ~ 10^3 to 10^4. This is expected to provide a sufficient data base for MHD pressure drop to be extrapolated to reactor conditions and to provide information on flow redistribution in strong fields (2.5 T).

No experimental facility is available to study the effects of MHD on heat and mass transfer. The differing requirements for test times (10's of hours for heat transfer tests versus 1000's of hours for corrosion tests) suggest that preliminary tests should be conducted in separate facilities.

Very limited investigations at ANL of oxidation of yttrium in lithium indicate that oxide layers form but the data are as yet inconclusive regarding the chemistry of the oxide and the insulating protection along discontinuities such as edges.

4.8.4 Required Resources

Upon completion of the Argonne facility, the preliminary round of liquid metal MHD pressure drop experiments on simple configurations can be completed in about 3 years. Follow-on tests for manifolding effects, etc., would require an additional 2 years. The annual cost of operation for the experimental program and sufficient parallel analysis to design experiments

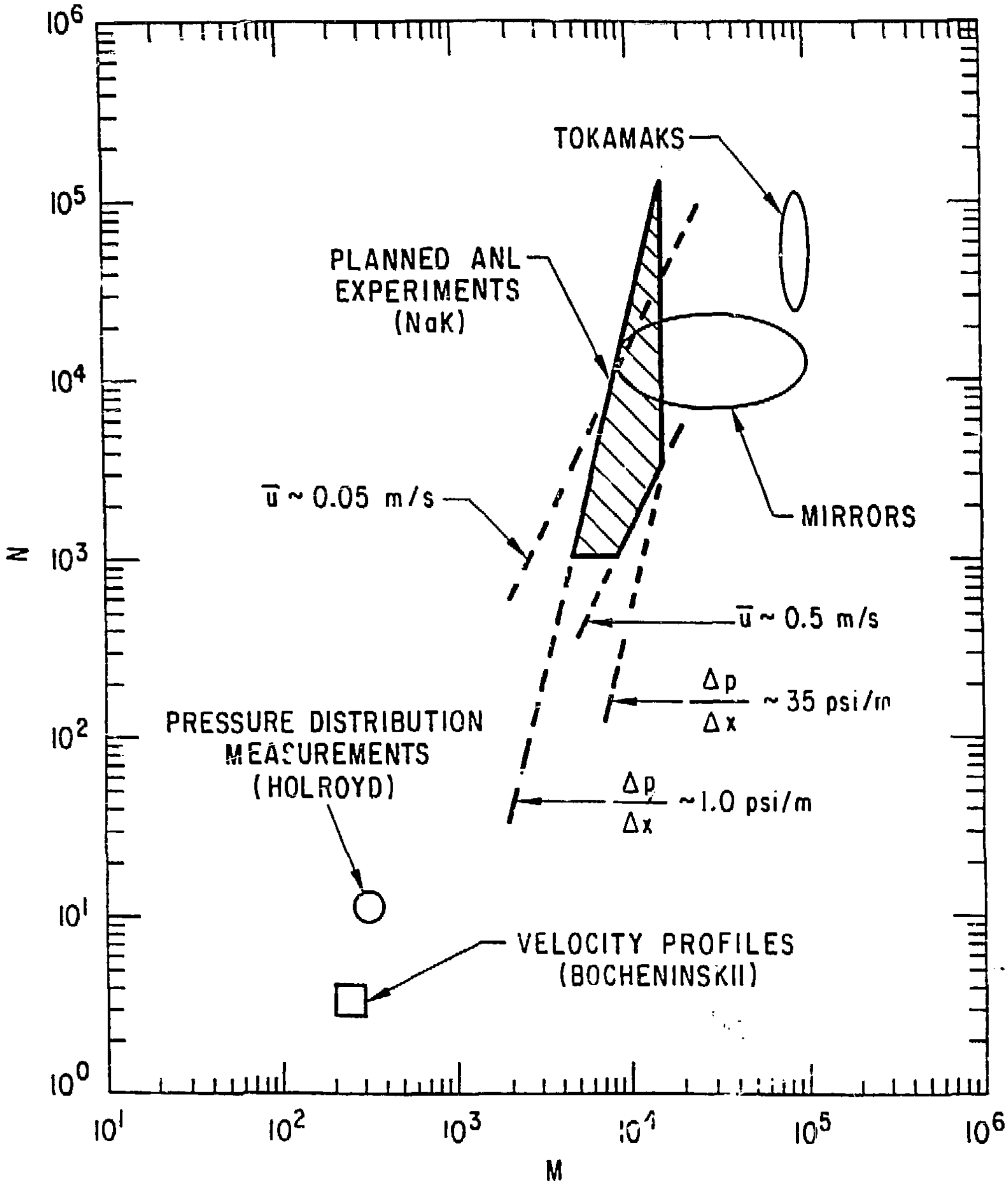

Fig. 4-1 Hartmann Number[(1)] and Interaction Parameter Ranges for Proposed Experiments.

(1) B. Picologlou, C. Reed, R. Nygren and J. Roberts, "Magneto-Fluid-Dynamic Issues for Fusion First Wall and Blanket Systems," Presented at the Fourth Beer-Sheva International Seminar on Magnetohydrodynamic Flows and Turbulence, Ben Gurion University at the Negev, Beer Sheva, Israel, Feb. 27 - Mar. 2, 1984.

and reduce data is 3.5 m-y/y for technical staff, plus $250K/y for equipment and services.

No facility is available for studying the effects of LMMHD on heat and mass transfer. Preliminary heat transfer tests could be combined with LMMHD tests and done in the regime of M above 1000 and N of 200-400, where flow will be laminar (turbulence suppressed) and reasonable flow velocities (~0.1 m/s) are available. This effort could be added to the ANL facility by extending the schedule for preliminary tests by one year and providing approximately $200K for additional equipment.

The compatibility of insulators in contact with liquid metals can be investigated using the existing lithium loop at ANL. The phenomenon of self healing of cracks should be included plus parallel mechanical testing. This first phase can be accomplished with about one man year of effort and $50K for equipment and services. Larger scale testing of a coated duct would require a different facility and preferably would be done in the presence of a magnetic field to assess the effect of any leakage currents on compatibility; however, a suitable facility does not exist. Scoping studies on fabrication and mechanical testing of insulated ducts with sandwich-type construction (structure/insulator/metal-skin) could be accomplished with a two year program at 1.5 m-y/y and $75K/y for materials and services.

4.10 Risk from Presence of Water in Near Blanket Components

4.10.1 Issue

Several fusion components, such as limiters, divertors, beam dumps, choke coils, and halo scrapers, experience severe heat loads which require a high level of heat removal. Water is considered to be the most efficient coolant for these components, but there are serious safety concerns about the use of high-pressure, high-temperature water in a reactor which also contains highly reactive materials like liquid metals and vanadium. Therefore, helium and liquid metals are being considered as alternate coolants for high heat load components. There are uncertainties, however, concerning the ability of the alternate coolants to adequately remove the heat.

4.10.2 Required Data

Information is needed in two areas. First, the safety issues related to the use of water in systems where highly reactive materials are present should be examined in detail. Second, the heat loading limits for alternate coolants need to be evaluated in order to determine if water can be eliminated completely from the reactor. The work should include a design effort to optimize the heat removal for alternate coolants, a modeling effort to address thermal hydraulics issues and the consequences of coolant failures, and an experimental effort to obtain data on the heat loading limits and failure rates.

4.10.3 Status of the Data Base

Preliminary designs have been developed for liquid lithium-cooled limiters and helium-cooled limiters, both of which appear viable, although this work to date is insufficient to establish firmly that water-cooled components can be eliminated from a reactor that uses liquid metals.

Failure rate information for fusion in-reactor components is nonexistent. Extensive data are available on failure rates of water-cooled systems in non-fusion applications.

4.10.4 Required Resources

Development of non-water-cooled component designs would require a period of perhaps three years at expenditures of approximately $500 K per year for each separate coolant/component combination. Studies to examine the safety issues will also be required at about 3 my-y for 3 years. These designs would then need verification testing in a simulation facility. This would perhaps require an additional five years at approximately $1 M per year, with a capital cost of the order of $1 M to provide the test facility.

Resources required for development of the data base for the probability and consequences of in-vessel water component failure could be quite large. In the absence of a fusion test reactor, simulation testing and modeling would be used to develop the data base. The modeling effort would make use of previous experiments and modeling of water cooled systems. The modeling work would cost ~ $500 K over 2-3 y. A simulation test facility to test the consequences of failures would need to be built.

4.11 Difficult to Meet Seismic Requirements

4.11.1 Issue

The self-cooled tandem mirror reactor (TMR) blanket with LiPb as the breeder and coolant may develop high seismic stresses in the front tubes and the beam zones because of the large mass of the coolant. The dynamic stress analysis of the thin-walled toroidal shell structures containing a flowing fluid of high density is complicated by the potential for fluid-structure interactions.

4.11.2 Required Data

The seismic analysis has to be carried out in two steps. First, the frequency-amplitude spectra at the supports for the blanket structure have to be determined from a global structural dynamics analysis of the blanket together with the building, foundation, etc. for the given input loading spectra at the bottom of the foundation. The stiffness, mass and damping of the blanket required to carry out this part of the analysis may be estimated from an approximate analysis of the blanket. The second step in the analysis will involve a detailed frequency response analysis of the blanket. During this phase of the analysis the importance of fluid-structure interaction has to be determined. If such an interaction is found to be unimportant, the analysis can be completed assuming that the fluid moves as a rigid body with the structure. If, however, the fluid-structure interaction is found to be important, some code development work will be needed because such an analysis is currently beyond the state of the art. Also, some experimental work will have to be conducted in order to determine the effect of this interaction on the damping of the blanket.

4.11.3 Status of Data Base

A very conservative equivalent static analysis of the LiPb blanket, carried out in the BCSS program, failed to meet the allowable primary stress criterion by a significant margin, indicating that a detailed dynamic analysis would be needed to ensure its ability to withstand a Zone II earthquake loading.

4.11.4 Required Resources

The first step of the analytical work can be completed within 6-12 months using any of a number of available finite element programs such as ANSYS, NASTRAN or ABAQUS. If fluid-structure interaction is unimportant, the second step of the analysis can be completed in a similar period of time. On the other hand, if fluid-structure interaction is found to be important, then the development of necessary codes to carry out the analysis will take 1-2 years. A parallel experimental program to determine the damping of the blanket will require full-scale dynamic testing of a toroidal tube of representative geometry filled with a heavy liquid of comparable mass. Testing should be completed within 1-2 years.

4.12 Adequacy of Tritium Breeding

4.12.1 Issue

Attaining fuel self-sufficiency in fusion devices operated on the DT cycle requires that the achievable tritium breeting ratio (T_a) in the blanket be equal to or greater than the required breeding ratio (T_r). At present the uncertainties associated with estimating both T_a and T_r are too large to assure satisfying the fuel self-sufficiency requirement.

The uncertainties associated with T_r are related to many of the plasma and reactor parameters (e.g., tritium fraction burnup in the plasma, tritium inventories and extraction efficiencies in the blanket, plasma exhaust processing system and other reactor components). Estimating T_a suffers from uncertainties associated with system definition and with accuracy of prediction. The system definition uncertainties relate to: a) technology choices (e.g., limiter or divertor, neutral beam vs. RF) that impact parasitic neutron absorption and the fraction volume coverage of the breeding blanket, and b) the degree and accuracy of details of present blanket and reactor designs. The prediction uncertainties are those associated with the neutronics geometrical modeling, basic nuclear data, data processing, and calculational methods.

4.12.2 Required Data

Items needed for the required breeding such as the plasma fractional burnup, tritium inventories in reactor components and characteristics of the tritium processing system are fairly independent of the blanket (except for tritium inventory and extraction for the blanket). Such data should be generated by the present R&D programs on plasma physics and plasma support systems (e.g., impurity control and exhaust) and on tritium extraction, processing and handling (e.g., TSTA). Items related to the achievable breeding ratio (tritium production) require the following efforts. a) It will be necessary to reduce uncertainties associated with predicting the tritium production in a specified system. This requires improvement in basic nuclear data, neutronics calculational methods, and most importantly direct verification through integral neutronics experiments. Integral experiments with a 14 MeV neutron source are presently considered to be the most effective approach. Neutronics experiments should include blanket assemblies with sufficient details of breeding materials structure, coolant, neutron multiplier, and geometrical characteristics. These experiments should proceed from the simple "clean" type to the more prototypical assemblies. b) It is also necessary to reduce uncertainties associated with system definition. This requires interactive R&D programs to determine those choices of reactor components that provide satisfactory functional performance consistent with attaining the goal of fuel self-sufficiency. For example, selection of neutral beams or RF for plasma heating and/or current drive is an issue that involves not only the plasma performance and reactor attractiveness but can substantially impact the magnitude of the achievable breeding ratio. The choice between limiter and divertor for impurity control can have significant impact on TBR, as will the choice of coolant for limiters. Further impurity control system design work is needed.

4.12.3 Status of the Data Base

At present, the uncertainties in estimating the required and achievable breeding ratio are ~ 25% and ~ 15%, respectively. The margin in the achievable breeding ratio is not sufficient to cover all sources of uncertainties. There are a number of activities presently underway in the U.S. and the world fusion programs that should reduce many of these uncertainties. However, the present effort on breeder neutronics experiments will need to be enhanced to provide important data on a timely basis.

4.12.4 Resources Required

The tritium breeding experiments can utilize existing facilities in the U.S. and Japan (FNS). Expenditure levels of two to three million dollars per year for several years are necessary to support the cost of materials for the experimental blanket assemblies, instrumentation development and the support for experimentalists and analysts. Efforts in other areas (e.g., plasma fractional burnup, tritium processing efficiency) can be covered under existing programs provided that adjustments in the priorities of such activities are possible.

4.13 Inability to Accommodate Breeder Swelling

4.13.1 Issue

Increases in solid breeder volume due to thermal expansion and irradiation-induced swelling are concerns in blanket design because of the resultant effects on breeder-structure thermal performance, on breeder-structure mechanical interaction, and on coolant and purge flow channels. The major consequences of excessive volumetric increases are increased temperatures if coolant flow channels are narrowed, decreased lifetime due to reduced structural integrity, and increased tritium inventory and leakage if purge channels are narrowed. Designs that attempt to compensate for excessive swelling add complexity and are penalized in the area of reduced tritium breeding ratio. The problem is of particular concern in the Li_2O plate design. Li_2O has a higher thermal expansion coefficient than the structural materials of interest, and it has been observed to swell significantly in the FUBR and TULIP experiments in the temperature range of 600 - 900°C due to helium bubbles.

4.13.2 Required Data

Information is needed on the swelling rate of breeder ceramics as a function of microstructure (e.g., grain size), temperature (300° - 900°C), burnup (0 - 10%), and fast neutron fluence (0 - 20 MW-yr/m^2). In addition, the creep and hot pressing rates, as well as the breeder elastic properties, should be known in order to assess the impact of the swelling. For materials with high creep and hot pressing rates, some of the swelling can be accommodated by the fabricated porosity and plenum spaces (for the plate design).

4.13.3 Status of the Data Base

The FUBR-1A data indicate that the retained helium and the volume increases in fine-grain-size $LiAlO_2$ pellets are very small. Li_2O swelling data inferred from the capsule plenum-volume decrease in TULIP, direct pellet diameter measurements in the 100 full power days (FPD) FUBR-1A, and neutron radiography in the 200 and 300 FPD FUBR-1A experiments indicate unconstrained volumetric expansion rates of ~ 0.8% $\Delta V/V_0$ per at.% of Li-6 burnup at 500°C and ~ 3% $\Delta V/V_0$ per at.% of Li-6 burnup at 700° - 900°C. These data need to be confirmed or refined based on PIE microstructural measurements. The diametral measurements include pellet cracking and pellet clad mechanical interaction which tend to distort the results. Also, data and/or model calculations for higher burnups and fast neutron fluences are required as swelling may not vary linearly with burnup.

Information from properties tests is forthcoming on Li_2O uniaxial creep, elastic properties, and hot pressing. Also, the planned FUBR-1B experiment will provide additional data on Li_2O swelling under more representative temperature-gradient conditions. Further modeling and analysis of the FUBR-1A and FUBR-1B data could be used to determine constrained swelling rates and to estimate the swelling behavior at higher burnups.

4.13.4 Resources Required

Assuming continuation of the solid breeder materials properties tests and the FUBR-1B experiment, no additional resources (with regard to experiments) are required to resolve this issue. Given the input from these studies, modeling efforts funded at a level of $200K for one year would be sufficient to both calculate the swelling rates at high burnup and fast fluence and to assess the degree of breeder-cladding structural interaction. Also, any in-reactor tests designed to generate high fluence tritium release data for Li_2O in a purge environment would also provide swelling data at a minimal extra cost.

4.14 Temperature Range for Tritium Release Much Less Than Predicted

4.14.1 Issue

The upper and lower temperature limits define the window within which a solid breeder blanket will be designed and operated to achieve its primary functions (i.e., tritium generation and recovery, heat generation and removal, etc.) while providing adequate blanket lifetime. The currently-recommended temperature windows are 350 - 1000°C for $LiAlO_2$ and 410 - 800°C for Li_2O. If the temperature windows are less than these values, then the potential impact is a reduction of breeder-structure fraction and breeding ratio. The primary concern with $LiAlO_2$ is the low temperature limit of 350°C. Uncertainties in the diffusion coefficient (~ ± 100%) and the burnup and fast neutron fluence effects on bulk diffusion could result in unacceptable tritium inventories and long times to reach tritium self-sufficiency. For Li_2O, the low temperature limit is based on avoiding the formation of an LiOT separate phase which could lead to high tritium inventories. The high temperature limit of 800°C is based on parameters that indirectly affect tritium release and blanket lifetime (e.g., grain growth, sintering, LiOT mass transfer and redeposition, helium swelling, etc.). For example, LiOT mass transport and redeposition could result in unacceptable lithium loss from the system and unacceptable tritium inventories due to plugging of the purge channels.

4.14.2 Required Data

Fast fluence data at low temperatures (300-500°C) are required for both $LiAlO_2$ and Li_2O in a purge flow environment to resolve the uncertainties in tritium inventory calculations. These experiments should be done in a TRIO-type, purge flow environment with careful control and monitoring of hydrogen and oxygen partial pressures in the system. It would also be preferable to perform these experiments in a fast reactor for a long enough period of time to assess the impact of fast fluence and burnup effects. For Li_2O, out-of-reactor tests should also be performed to study the steady-state and transient solubility of hydrogen isotopes in Li_2O for various ranges of hydrogen, oxygen, and water vapor partial pressures in the helium purge.

High temperature thermal, sintering, vaporization, and mass transfer studies are required for both sintered and sphere-pac solid breeders to provide the baseline information necessary for defining the upper temperature limit. Neutrons are required to assess radiation effects on grain growth and sintering.

4.14.3 Status of Data Base

A number of post-irradiation annealing experiments have been conducted on polycrystalline pellets and powders of $LiAlO_2$. Inadequate characterization of microstructure, poor control of moisture levels and impurities, and inconsistent methods for data correlation have resulted in considerable scatter in trying to determine a bulk diffusion coefficient from these experiments. The TRIO-1 test has provided considerable information on tritium release from irradiated $LiAlO_2$ as a function of temperature and purge stream chemistry. The limitations in the data base are that the run-times for the low temperature tests were too short to establish equilibrium inventories and release rates and that the fast fluence and burnup levels were too low (equivalent to ~ 2 months in a STARFIRE type reactor) to allow their effects to be assessed. In addition to unresolved issues for tritium inventory at low temperatures (e.g., 300-500°C), high burnup and fast fluence effects, the influence of moisture and oxygen partial pressure in the purge stream needs further study. This is particularly apparent when one compares TRIO runs with and without isotopic (protium) swamping to the closed-capsule FUBR-1A runs with $LiAlO_2$.

For Li_2O, the bulk diffusion coefficient in tritium is reasonably well known in the temperature range of 600-900°C. Data are needed in the temperature range of 300-600°C at high fluences and burnups in a purge flow environment with the purge stream chemistry carefully controlled and monitored. The solubility of lithium hydroxide in lithium oxide at various moisture overpressures has been measured. Preliminary vapor phase transport of LiOH at 700°C and 10 volume ppm H_2O in flowing helium appears to confirm projections based upon the thermodynamic data of Norman at GA and Tetenbaum at ANL. Additional information is needed at higher moisture contents and at other temperatures. The data of Ihle and Wu on the solubility of deuterium in Li_2O led to the unusual conclusion that tritium solubility is more sensitive to T_2 (or HT) partial pressure at some temperatures than to T_2O (or HTO) partial pressures. These tests need to be repeated under more controlled circumstances to resolve this issue.

Finally, tritium retention due to surface adsorption for Li_2O and $LiAlO_2$ needs to be distinguished better than was possible in VOM-15H and TRIO purge flow tests. It is not clear whether the higher inventories in TULIP and FUBR resulted from surface adsorption, solubility, or a lowering of the bulk diffusion coefficient.

4.14.4 Required Resources

Most of the issues involved in determining whether or not the temperature range for tritium release is much less than predicted can be resolved with carefully planned, controlled, and monitored TRIO-type tests with Li_2O and $LiAlO_2$ in a fast reactor. The coolant inlet temperature in EBR-II is low enough to provide information on these breeders for T > 350°C. The experiences with VOM-15H (Li_2O) and TRIO-1 ($LiAlO_2$) will allow more prudent selection of run times, temperatures and purge flow chemistry. The estimated cost for the planning, testing, and data analysis of second generation Li_2O and $LiAlO_2$ open-capsule tests is ~ $5M over a three year period. Some of the cost of the Li_2O experiment could perhaps be shared with the Japanese who have an interest in this breeder material. Similarly, cost-sharing with the French could be explored for the $LiAlO_2$ experiment.

Many of the issues concerning radiation and temperature gradient effects on grain growth and densification for sintered and sphere-pac solid breeders will be resolved in the FUBR-1B tests if they proceed as planned. Additional bench-type experiments (without neutrons) should be performed with Li_2O to resolve issues concerning solubility, vaporization, and redeposition at high temperatures and the temperature gradients. The estimated operational cost for these experiments is ~ $2M over a period of two years.

4.15 Unacceptable Temperature Predictability of Breeder (e.g., Breeder-to-Structure Gaps are Created)

4.15.1 Issue

Assuming that the temperature windows presented in the previous section for $LiAlO_2$ (350°-1000°C) and Li_2O (410°-800°C) are correct, then uncertainties in temperature predictions based on a number of phenomena would cause a reduction in the nominal temperature window used for design to allow for hot-spot factors. Higher-than-predicted breeder temperatures are more probable than lower-than-predicted temperatures. For helium-cooled, plate-type breeder designs, irradiation damage resulting in lower thermal conductivity than predicted (50% reduction), thermal stress cracking creating thermal resistance gaps within the ceramic, and narrowing of the coolant passages between plates due to breeder (Li_2O) swelling would all result in higher than predicted temperatures. For the water-cooled, sphere-pac designs ($LiAlO_2$), gaps could be created between the sphere-pac breeder and the coolant tube due to settling, compaction, and thermal or irradiation sintering. In addition, if the effective thermal conductivity for sphere-pac $LiAlO_2$ in an irradiation environment is lower than predicted, higher temperatures would result. The primary concerns with higher-than-predicted temperatures are vapor transport and redeposition, grain growth and sintering, all of which could affect the tritium inventory and release characteristics of the system.

4.15.2 Required Data

Baseline effective thermal conductivity data are needed for sintered breeder products (Li_2O and $LiAlO_2$) in contact with metal plates as a function of temperature (300° - 1000°C), porosity (75-100%), fast neutron fluence (0.5 - 2 x 10^{26} n/m^2 which is equivalent to 0.7 - 2.8 MW/y· m^2), and contact pressure (0 - 50 MPa). In addition, comparable data are needed for sphere-pac $LiAlO_2$.

Thermal stability experiments with prolonged test periods and thermal cycling are required to determine the degree to which densification and cracking will affect thermal peformance. To a large extent, if thermal instability is a problem, it will be observed in neutron irradiation tests at fluence levels up to or before the saturation fluence level for the ceramic. The issue of settling and subsequent change of thermal properties for a sphere-pac breeder with coolant tubes may require a full blanket module or at least a number of unit cells. Some useful information can be obtained from an out-of-reactor unit cell test with a central coolant tube and external heating. However, extrapolation of this data to a full blanket module operated in a neutron environment is quite difficult.

4.15.3 Status of Data Base

Out-of-reactor thermal conductivity measurements have been performed over adequate ranges of porosity and temperature for Li_2O but not for $LiAlO_2$. Current values for unirradiated, sintered Li_2O have a relatively small uncertainty (± 2-3%). The estimated thermal conductivity of unirradiated sintered $LiAlO_2$ is based on extrapolation of low-temperature data and the assumption of the same porosity dependence as Li_2O. The combined uncertainties for unirradiated $LiAlO_2$ are probably ~ 30%. Irradiation degradation of the thermal conductivities was based on a theoretical model with estimated uncertainties ~ 50%. No data are available for the effective conductivity of sphere-pac $LiAlO_2$. Model calculations are based on the results of the effective conductivity of sphere-pac UO_2.

The TRIO-1 experiment provided data which allowed an estimate of thermal conductivity as a function of temperature and time. However, the fluence was relatively low (equivalent to 2-3 months in a STARFIRE type design) and the low density of the $LiAlO_2$ used in TRIO means that the effective conductivity is less susceptible to irradiation degradation than a higher density material would be.

4.15.4 Resources Required

Facilities adequate for non-irradiation scoping tests already exist at some of the national laboratories and industries involved in the fusion engineering effort. No additional capital costs of significance are anticipated. Operating costs of $500K per year for two years are anticipated for resolving the questions concerning the temperature and porosity dependence of the thermal conductivity for sintered and sphere-pac $LiAlO_2$ and the gap conductance for sintered Li_2O plates. The same submodules used for the sphere-pac $LiAlO_2$ tests can be used for longer range stability tests under conditions of thermal cycling.

Studies at ANL have indicated that it is possible to use microwaves to heat a solid breeder unit cell or blanket module in a fashion that closely simulates the heating profile in a fusion reactor. Experiments on a modest size test piece (~ 10 x 10 x 20 cm) of $LiAlO_2$ are underway. A suitable test stand utilizing this technique could be built for < $1M.

In-reactor exploration of the temperature control issue on a full module basis would be very expensive and should probably await the demonstration phase of the fusion program. However, the issue of irradiation degradation for both the Li_2O (sintered) and the $LiAlO_2$ (sintered and sphere-pac) can be explored either by performing post-irradiation tests on samples from the FUBR irradiations or by direct thermocouple measurements in a TRIO-2 type test. Similarly, the in-reactor stability of sphere-pac and sintered products can be partly assessed from the results of currently planned in-reactor experiments.

4.16 Unacceptable Power Variation Capability

4.16.1 Issue

Power reactors, both demonstration and commercial, will be required to operate at power levels different from their nominal (100%) design points. Operation at power levels as low as ~5% may be necessary for extended periods for such purposes as startup and low-power operation during the licensing period, or checkout of newly-installed components such as replacement steam generators.

Solid tritium breeders will have allowable maximum and minimum temperature limits for normal operation. The extent to which these limits can be violated during power variations of certain durations is not known. In some cases such operation could permanently affect the breeder and thus the ability to adequately recover tritium from the blanket. The capabilities of the various solid breeder blanket concepts in terms of maximum and minimum allowable power levels cannot be determined until breeder temperature/time limitations are more firmly established.

4.16.2 Required Data

The primary focus of this effort should be to use the results of the experimental program needed to complete the solid breeder data base in determining through design studies of blankets (coupled with their power conversion systems) the power level/duration capabilities of the leading blanket concepts. The study results should indicate specifically:

- Allowable duration for operation at each power level considered.
- Allowable duration for operation at various breeder temperatures, specifically from the standpoint of effects on tritium release and structural-mechanical interactions with the breeder or beryllium.
- Operating adjustments necessary to affect changes in power levels, and related impact on design of subsystems (e.g., power conversion, tritium removal) in terms of economics and added complexity.

4.16.3 Status of Data Base

Of the key phenomena identified for the issues for solid breeder blankets (e.g. issues # 13, 14, and 15) those considered most important to determining power variation capabilities of the concepts are: temperature limits, grain growth/sintering/crack healing, tritium lattice and porous diffusion, and structural-mechanical interactions with solid breeders and beryllium. The status of the data base for each of these areas is summarized in the respective subsections of the other issues. In general, the experimental information available is very limited and not directly applicable to conditions for fusion power reactor blankets.

4.16.4 Required Resources

See the R&D assessment for other solid breeders key issues; Sec. 4.13, 4.14, 4.15, and 4.17.

4.17 Fabrication /Refabrication of Solid Breeder

4.17.1 Issue

Reasonable success has been achieved in fabricating Li_2O and γ-$LiAlO_2$ sintered and hot pressed pellets with grain sizes low enough to satisfy tritium inventory limits. Fine grain-size $LiAlO_2$ pellets also have been produced with tailored bimodal porosity distributions for the TRIO-1 experiment. It remains to be demonstrated that the sphere-pac form for $LiAlO_2$ can be fabricated and packed to the desired density in the complex blanket module with its large number of coolant tubes. In addition, if the breeder is to be refabricated, all of the above approaches need to be re-examined in the context of remote handling techniques. Improper control of grain size, porosity distribution, and impurity levels can have a dramatic effect on both thermal and tritium transport.

4.17.2 Required Data

While additional experience is needed to optimize fabrication methods for producing pellets and plates of the candidate breeder ceramics, a high priority in fabrication is to demonstrate that the sphere-pac form can be fabricated to specifications and packed to the desired density in a complex blanket module. As an additional part of the fabrication effort, out-of-reactor thermal stability experiments should be performed with the sphere-pac product.

4.17.3 Status of the Data

Cold pressing and sintering techniques have been developed for: (1) Li_2O, Li_4SiO_4, and $LiAlO_2$ samples irradiated in ORR; (2) Li_2O cylinders used in the TFTR Lithium Blanket Module (LBM) program; (3) γ-$LiAlO_2$ pellets with tailored microstructure used in the TRIO-1 experiment; (4) γ-$LiAlO_2$ plates and cubes to be used in the ANL microwave heating experiment; and (5) Li_2O pellets and pebbles used in the various Japanese experiments. Hot pressing techniques were developed for Li_2O, Li_4SiO_4, γ-$LiAlO_2$, Li_2ZrO_3, and Li_8ZrO_4 pellets for FUBR-1A and/or FUBR-1B experiments. Product dimensions vary from small pellets to relatively large γ-$LiAlO_2$ plates; production scale varies from laboratory to relatively large through-put operation for the LBM program. Efforts to fabricate sphere-pac γ-$LiAlO_2$ have been initiated at ANL and GA in support of the FUBR-1B experiment.

4.17.4 Required Resources

A five-year program is recommended with a facilities cost of \$5M and an operating budget of \$1M per year to optimize the fabrication of Li_2O and $LiAlO_2$ sintered products and to resolve the uncertainties associated with sphere-pac $LiAlO_2$.

4.18 Tritium Recovery/Containment and Control

4.18.1 Issue

Tritium recovery and control is a very serious issue for blankets and their power conversion systems. For example, in a steam generator the tritium leakage rate to the steam side may have to be limited to ~ 10 curies/d. To achieve this limit, the tritium partial pressure of the primary coolant (or secondary coolant if care is used) in the steam generator has to be ~ 10^{-7} Pa or less. The tritium partial pressure in the blanket, which varies according to the breeding material, is usually around 0.01 to 1 Pa, except for Li which is ~ 10^{-9} Pa. The large pressure differences and the high mobility of the hydrogen atoms at high temperature causes the difficult control problem. Therefore, it is necessary either to have a much more efficient recovery system to reduce the driving pressure or a very successful tritium diffusion control system to increase the resistance to tritium permeation.

In the temperature range of interest, tritium permeation through iron based alloys is rather rapid. Tritium control methods assumed in various blanket studies are usually either diffusion barriers and/or oxidation of the tritium to the oxide form. The key issues are:

- What are the mechanisms and how efficient are the diffusion barriers?

-- What are the kinetics of the oxidation and the effect of oxidation on permeation rates?

4.18.2 Required Data

Experimental results and theoretical understanding in the following areas are required:

1. The chemical form of the tritium being released from a breeding material.
2. The oxidation kinetics of the tritium in both bulk and surface applications.
3. The formation and effectiveness of a diffusion barrier and its performance in the fusion environment.
4. The effect of the chemical form of tritium on the permeation rate.
5. The deviation from the classical permeation relationship at low tritium partial pressure.
6. Tritium implantation and diffusion through the first wall, limiters, direct converters, etc.
7. The effect on tritium oxidation of He and O_2 in the purge stream.

Current theories are not adequate to reliably predict the results. Therefore, experiments have to be carried out under conditions as realistic as possible. In particular, the tritium partial pressure has to be kept as low as 10^{-7} Pa. Since isotopic effects may have a major impact on tritium permeation mechanisms, the hydrogen and moisture partial pressure should be kept somewhat lower than 10^{-7} Pa, which is difficult.

4.18.3 Status of Data Base

The data base is very sparse. The results from different experiments which attempted to measure the chemical form of tritium released from the breeding materials are not consistent with each other. Previous tritium permeation experiments were performed at much higher tritium partial pressures than are anticipated for fusion applications. In addition, the magnitude and effect of the background hydrogen partial pressure is uncertain. The term "barrier factor", commonly used for tritium permeation experiments, is not well characterized. The mechanism of the increased resistance is so unclear that it is not certain whether the "barrier factor" should be added to or multiplied by the base material resistance.

4.18.4 Resources Required

It is difficult to estimate the required resources in this area due to the poor understanding of the problem. Usually the best approach for such a problem is to combine theoretical modeling and experimental verification. The experimental results will validate the realism of the theoretical modeling, while the theoretical modeling will enable one to define the experiments and extrapolate the results. However, for this problem, that type of approach may not be feasible due to the complexity of the problem. An engineering mock-up type of experiment may be far cheaper. The information obtained will be relevant to the experimental conditions but not extrapolatable.

It is expected that it will take 3 m-y/y for 5 years to establish and calibrate the theory. It will take another 6 m-y/y for 5 years to obtain experimental results which should be carried out in parallel with the theory effort. If only engineering information is required, the effort can be cut by more than half, i.e., 4 m-y/y for 5 years to test a mock-up model for each type of blanket concept.

4.19 Loss of Be Integrity

4.19.1 Issue

Beryllium is found to swell under irradiation. Further, the swelling behavior is strongly temperature-dependent. If significant temperature gradients exist, the large differential swelling can result in the build-up of stresses. If, at the same time, the beryllium becomes embrittled by the irradiation, it can lose mechanical integrity. Although the beryllium in the fusion reactor blankets considered in the BCSS is not used in a structural mode, it must at least support its own weight under normal and off-normal conditions. Loss of mechanical integrity of the beryllium would result in potential flow blockage and blanket temperature distribution upset.

4.19.2 Required Data

The data required for the helium-cooled lithium aluminate ferritic steel beryllium blanket are the swelling characteristics and the chemical property characteristics of conventional beryllium rods under irradiation to fluences of up to 20 $MW\text{-}yr/m^2$, at temperatures between 300°-500°C.

4.19.3 Status of the Data Base

A significant data base exists for irradiation of beryllium in fission reactor assemblies under a wide variety of temperature and fluence conditions. What is needed for the fusion blanket situation is a verification irradiation in a fission reactor for the product form and the temperature conditions desired for the helium-cooled lithium aluminate beryllium blanket. Ultimate testing will require a fusion neutron spectrum and will have to be done in a fusion test reactor.

4.19.4 Resources Required

Although nuclear testing is required for the beryllium radiation effects, the extensive existing data base indicates that only a modest program is needed. A five-year program at $1M a year using existing facilities should be able to achieve most of the data required.

4.20 Inability to Reprocess Be at Low Loss Rates

4.20.1 Issue

Beryllium reprocessing (including refabrication) must be carried out with low loss rates (~ 1-2% per recycle) to avoid resource limitations. The reprocessing must also be done remotely because the beryllium will be radioactive due to activation of impurities in the beryllium. For the BCSS NS and H_2O blanket designs, this includes separating Be from $LiAlO_2$ microspheres and remanufacturing. Chemical purification may also be needed. Other Be forms such as balls, rods and blocks should also be investigated to help determine the best mechanization of the design concept.

4.20.2 Required Data

The required data includes development and validation of methods for remote manufacturing of Be microspheres, development and validation of methods for remote separation of Be from $LiAlO_2$ microspheres at high efficiency, and development and validation of methods for remote, low loss reprocessing and refabrication of other Be forms that may be used in fusion blankets.

4.20.3 Status of Data Base

There is considerable expertise in remote fabrication in the beryllium industry based on powder metallurgy technology. This is done remotely because of the toxicity of BeO. Processes would have to be upgraded for full remote operation because of induced radioactivity. Significant study of reprocessing and refabrication of Be balls was done for the Fusion Breeder Program. To our knowledge, no work has been done on microspheres.

4.20.4 Required Resources

Development of the chemical and mechanical process necessary to reprocess, refabricate and reuse the beryllium should be readily attained in a matter of approximately three years at less than $500K per year using existing facilities. Verification of the process in a radiation environment, however, will be more difficult. Simulation of the reprocessing using non-radioactive chemical analogs for the various activation products and impurity products should be sufficient to demonstrate the reprocessing capability. Risk will always exist, however, that impurities or reaction products not previously considered may result in higher radiation levels than planned for. This will ultimately require actual fusion reactor irradiation and is best done on an actual reactor application basis.

4.21 Tritium Release from Be to Primary Coolant

4.21.1 Issue

In addition to the copious quantities of helium produced in beryllium in a fusion neutron spectrum, tritium is produced in amounts that may pose safety concerns. The helium bubble density will establish quickly compared to the time required to produce significant quantities of tritium. Therefore, the diffusional aspects of tritium behavior in irradiated beryllium will largely be dominated by this bubble population and neutron-induced lattice defects.

In the first 10 centimenter region of a beryllium containing blanket, tritium will be produced in the beryllium at about 20 appm per MW$\cdot$ yr/m^2 of exposure. One gram of beryllium would produce 7×10^{-6} gms of tritium per

MW• yr/m^2. An estimate of tritium production in beryllium for the tandem mirror $LiAlO_2$/NS/HT-9/Be blanket is slightly over 1000 gms per full power year.

4.21.2 Required Data

Basic data is required on tritium diffusion and release rates from unirradiated beryllium samples. Testing variables include temperature (300 to 800°C) and grain size. It is desirable to investigate inventive methods of possibly injecting tritium into post-irradiated beryllium samples at various fluence levels for subsequent diffusion and release rate data. Fusion spectrum irradiation of beryllium would then be needed for real-time damage, He production and tritium production to various fluence levels (up to 10^{27} n/m^2) and temperatures. The (n,t) cross section has a threshold of 11.6 MeV and is 0.018 barns at 14 MeV. The importance of the fusion neutron spectrum is obvious.

4.21.3 Status of Data Base

A useful, but limited, amount of temperature-dependent swelling data exists from fission reactor irradiations. The neutron spectrum is too low to generate tritium or to provide the proper He/dpa ratio. However, this data can be used for bubble densities and sizes for numerical modeling.

4.21.4 Required Resources

The irradiation aspects of this issue will dominate the resource requirements. About two years and three manyears of effort would be required each for unirradiated Be data, fission reactor irradiations, and fusion spectrum irradiations. A 14 MeV irradiation source is required.

4.22 Excessive Chemical Reactivity of Be with Salt

4.22.1 Issue

Nitrate and nitrite salts (NS) are good oxidizing agents which can result in a high rate of energy release when put in contact with an active reductant. For the specific blanket design in the BCSS, leakage of draw salt into the sphere-pac mixture of Be and $LiAlO_2$ powder is the largest concern. Reactions with other potential blanket materials, such as 17Li-83Pb, are also a concern.

4.22.2 Required Data

Tests of NS with powdered Be/$LiAlO_2$ with the NS at 450, 500 and 550°C and the solid at 700°C are required. It will be necessary to monitor temperature and pressure response and chemical reaction products. Tests are also required of NS with 17Li-83Pb with the NS at 450, 500 and 550°C and the 17Li-Pb at 50 to 100°C higher temperature than the NS. Injection of NS into 17Li-83Pb and 17Li-83Pb into NS should both be done to simulate a blanket and an IHX. Screening tests of NS with other blanket materials and material forms should also be carried out. These include Li_2O, other ternary ceramics, Be, graphite, SiC, and lead at 500°C.

4.22.3 Status of Data Base

Much of the data comes from manufacturers and should be validated by well-characterized tests under fusion blanket conditions. Preliminary 17Li-83Pb tests have been run at HEDL.

4.22.4 Required Resources

The effort required on this issue is expected to be about 4 manyears and requires on the order of two years. No new major facilities are required.

4.23 Salt Stability/Decomposition Worse Than Predicted

4.23.1 Issue

Nitrate and nitrite salts decompose at high temperatures and may decompose under neutron and gamma irradiation. Some of the decomposition products are insoluble while others are gaseous. Typical reactions are:

$$2\ NaNO_3 \rightarrow 2\ NaNO_2 + O_2, \text{ and}$$

$$5\ NaNO_2 \rightarrow 3\ NaNO_3 + Na_2O + N_2.$$

Depending on the salt, an appropriate cover gas is used to drive the reactions to the left. If excessive decomposition occurs, a large clean-up system may be needed. During a temperature excursion, decomposition may affect safety.

4.23.2 Required Data

Data is required on decomposition rates and reactions at temperatures from 400°C to 600°C at 25°C intervals. Some temperature excursion data up to 800°C is also required. Decomposition rates and reactions are also required in a combined radiation and temperature field at temperatures from 400°C to 550°C and radiation dose rates from 5×10^5 to 5×10^6 rad/s at 25°C intervals and two intermediate radiation dose rates. Tests should also be carried out on decomposition in a temperature excursion from an initial temperature of 450°C at heat input rates from .01 to 1 W/cm^3. The effects, if any, of cover gas pressure on decomposition rates should be investigated and the pressure increase in the gas measured.

4.23.3 Status of Data Base

There is a reasonable data base on thermal decomposition from solar and other programs. However, it is difficult to assess, incomplete and not well documented. The data on radiation induced decomposition is effectively nil.

4.23.4 Required Resources

The thermal and radiation decomposition experiments should require one to two years with about an 8 manyear total effort. No new major facilities should be required.

4.24 Grooved First Wall

4.24.1 Issue

The concomitant requirements of moderate to high surface heat flux capability and long erosion life in the design of tokamak first wall/blanket systems will almost certainly require some form of stress relief in the first wall. Providing orthogonal grooves on the first wall has emerged from the BCSS study to be a potentially promising method for achieving this goal. Concerns for the viability of using grooved first wall have centered around the propensity for such a structure for premature crack initiation at the roots of the grooves due to stress concentration effects and the subsequent propagation of these cracks to early failure.

4.24.2 Required Data

Because of the complexity of the problems a combined analytical/experimental approach is recommended. The analytical program should precede the experimental program and help design optimum configurations of the grooves as a function of the surface heat flux. The stress analyses should provide the stress concentration factors and the nature of stress distribution in the grooved structure as it evolves in time because of time-dependent deformations. Specifically, the analyses should indicate the similarity and dissimilarity between the stresses and deformations produced in a grooved plate loaded by surface heat flux as opposed to loaded by mechanical means. These data will be particularly useful in determining if and how experimental data generated using the simpler and less expensive method of loading by mechanical means can be applied to structures loaded by surface heat flux.

Experimentally it is necessary to achieve a surface heat flux of the order of 1 MW/m^2 on structures of reasonable sizes. Facilities are available using either electron beams or ion beams. In addition, specimens of size 10 cm x 10 cm can be heated to a maximum of 0.5 MW/m^2 by using high intensity xenon lamps. The specimens will have to be cooled and constrained such that they develop stresses that are representative of the expected stresses in a blanket.

Because of the complexities and expenses inherent in such tests, it is recommended that the bulk of the tests be conducted on uniformly heated grooved plates or cylinders by subjecting them to steady and/or cyclic loads by mechanical means. Specimens should be fabricated from a representative material whose ductility has been reduced by appropriate metallurgical treatment such as cold work. The tests should be carried out to failure or crack initiation and a significant effort should be spent on correlating these data with the groove stress analysis, crack initiation and crack propagation data of the material.

A limited number of grooved plates should be tested with the more elaborate and prototypic setup using electron or ion beams as well as possibly xenon lamps. These tests should be used as validation tests for designs of grooves developed in the earlier phase of the program.

4.24.3 Status of Data Base

Approximate analysis carried out under the BCSS program has shown that grooves remove the region of highest stress in the structure from one at the highest temperature to one at a much lower temperature. However, the stresses at the root of the grooves are high and need to be determined as functions of the geometrical parameters of the grooves using more accurate analysis methods. No experimental data on the effects of a surface heat flux on a grooved plate currently exist.

4.24.4 Required Resources

The analytical part of the program can be carried out using any of a number of available finite element programs such as ANSYS or ABAQUS and should require an effort of one manyear. The bulk of the tests with the grooved specimens can be conducted in any laboratory having a conventional loading fixture and should be completed by 1-2 years. Possible surface heat flux test facilities exist or will soon be operational (e.g., ASURF and the PMTF). The testing could be completed within 2-3 years.

4.25 Excessive Leakage of Coolant to Plasma Chamber

4.25.1 Issue

Under the action of stress, irradiation and thermal cycle behavior, small flaws in a structural material may grow. Because of the thinness of the structural material between the coolant and the plasma chamber in the firstwall, a flaw can grow through the wall, permitting coolant to leak into the plasma chamber. Because of the high purity requirements of the plasma even small quantities of leakage of any of the coolants could extinguish the plasma.

4.25.2 Required Data

The data required to resolve the R&D issue is verification of the crack propagation and leak rate projections that are presently being made and that appear to give contradictory results.

4.25.3 Status of Data Base

Since the crack propagation is strongly influenced by radiation damage of the materials involved, relatively little data exists on the crack propagation concerns that would be relevant to a fusion first wall. The fracture mechanics methodology necessary to predict crack growth of this kind is in existence and is being used today as part of the ADIP program. The projections from these various models appear to be contradictory. Reference 1 projects that the crack growth will not prove to be a limiting concern and that surface cracks on the order of a few tenths of a millimeter deep which would be detectable by conventional techniques could be tolerated. Reference 2, by the same author, projects that the requirements for flaw detection may be much smaller. Reference 3 projects that, similar to Reference 1, the tolerable crack size to prevent leakage during the lifetime of a typical reactor first wall is on the order of a few tenths of a millimeter and should be readily detectable. Thus, the data base needs further experimental confirmation. The experimental information from water leakage in operating HTGRs which have similar restrictions on water in-leakage appear to confirm the projection of no unusual leakage concerns even under thermal cycling, at least in the absence of radiation. With radiation damage, however, additional questions must be resolved.

4.25.4 Required Resources

The resources required to resolve this issue may be quite large. Essentially what is needed is an irradiation under pressure and thermal cycling of a prototypical first wall component. In fact, this radiation test may be required to be done in an actual operating fusion reactor.

4.26 Coolant/Breeder Cleanup after Spills

4.26.1 Issue

During operation of a fusion power plant, spills of the various blanket fluids will occur. The spills may occur either inside the reactor building or into the vacuum chamber. For most fluids fairly standard cleanup methods are expected to be adequate. For certain fluids, such as LiPb and FLIBE, additional research is required. In all cases except helium, the actual cleanup of the blanket fluid will likely be difficult because of its radioactivity.

4.26.2 Required Data

Research is required to find an acceptable means to clean up a spill of either 17Li-83Pb or FLIBE. Chemical research to determine a suitable solvent is required.

4.26.3 Status of Data Base

Cleanup of water may be difficult, but existing methods should be acceptable. Sodium cleanup techniques will work for lithium. Specifically, large pieces will be removed mechanically and the rest can be steamed-cleaned since lithium reacts with water. This has been used at various research facilities using lithium. Because the nitrate salt is soluble in water, the methods for cleanup should be fairly straightforward but will be difficult in practice because of the very high radioactivity of the nitrate salt. For these reasons the technical risk and additional research for water, lithium, and nitrate salt spill cleanup appear to be minor. In the long run, maintenance-related research and development will need to verify that these fluids could be completely removed after a spill.

Cleanup of 17Li-83Pb or FLIBE does not appear to be straightforward. Neither material is soluble in water; their low chemical reactivity is a disadvantage in this regard. At present, the only identified cleanup method would be physical removal of the solidified spilled material. This would be particularly difficult in the case of a spill into the vacuum chamber. In fact, it is possible that a 17Li-83Pb or FLIBE spill could solidify in between two blanket sectors and perhaps "self-weld" the two sectors. No chemical solvent for 17Li-83Pb or FLIBE has been identified by the BCSS.

4.26.4 Required Resources

No new facilities are required. Although in practice much of the difficulty in cleanup would be associated with the highly radioactive nature of these fluids and the particular geometry of where they are spilled, the required research should initially focus on finding a solvent. A two man-year level of effort for two years is estimated. If, however, a modest amount o chemical research effort failed to find a suitable solvent, additiona resources would be required to attempt to find other cleanup methods suitabl for these two fluids.

4.27 Coolant Containment Reliability of Double Wall Tubes Less than Predicte

4.27.1 Issue

For blankets which have high pressure water coolant contained in smal diameter tubes, projections of single wall tube (SWT) reliability agains leaks lead to predictions of between 1 to 100 or more failures per year amon the 120,000 or so coolant tubes needed for a STARFIRE-sized power reactor. double wall tube (DWT) design approach has been adopted for the $LiAlO_2/H_2O/HT$ 9/Be blanket concept. The approach features two separate pressure boundaries each sized for the coolant design pressure, and separate welds of each tub end such that a through-crack would be required in each tube of any one DW assembly before coolant could leak into the breeder zone. The principa uncertainty in the feasibility of the apporoach seems to be the degree t which common-mode failures will affect the overall reliability of the system.

4.27.2 Required Data

To establish reliability data with sufficient confidence to choose th concept for DEMO, it will be necessary to test a fairly large matrix, wit extensive replication of samples, under (1) non-irradiated conditions, and (2 irradiation. Conditions to be simulated for the non-irradiated tests include

- coolant chemistry
- coolant temperature (280 to 320°C)
- coolant pressure (14.0 to 15.5 MPa)
- environment external to the DWT
  - He at 1 to 6 atm, with 0.1% H added
  - breeder thermal heating
- end weld configuration
- coolant thermal cycling (T-max to RT to T-max; up to 4×10^4 cycles)

4.27.3 Status of Data Base

For SWT, there is much data available from the fission reactor industry, but in general the data is for materials other than ferritic steel. The data spread appears to be very wide. Overall, the available SWT data is likely to be of very limited use as a guide to the performance of DWT in fusion power reactor blankets.

For DWT, some testing work has been done within the LMFBR program for steam generator tubing with sodium as the primary coolant. The data thus will have little applicability to the present blanket concept because of differences in: environment (surrounding materials), structural materials (different alloy for tubing), welds, tube thicknesses, coolant temperatures and pressures, and transient conditions.

4.27.4 Resources Required

For non-irradiated testing, adequate facilities should already exist at national laboratories and perhaps at certain industrial sites. No significant capital cost increment is foreseen for facilities. An operating level of effort of about 2 to 4 $M/yr would probably be needed for up to 10 years in order to generate adequate data to determine optimum design goals.

Subsequent irradiation testing under simulated power reactor conditions is seen as verification of the adequacy of the design details previously developed. The tests will determine whether irradiation effects such as reduced ductility will significantly affect failure rates previously predicted. Assuming that tests in fission reactors such as EBR-II are adequate, some modification of these facilities might be required. A capital cost increment of $5M is assumed for these changes. Test operations would require 3 to 5 $M/yr over a 5-year period. Tests in fusion reactors would be of limited use since a relatively large test volume and extensive test times would be required in order to obtain relatively high confidence in the results.

4.28 Excessive Activation Products from Nitrate Salt and 17Li-83Pb

4.28.1 Issue

The nitrate salt (NS) and lithium lead (LiPb) coolants will be highly activated in the fusion blanket. The key concern is determining the activation products, their forms and our ability to contain normal emissions. Of particular importance for the salt are the gaseous products, argon and CO_2, especially the former. Regarding the LiPb coolant, certain isotopes in lead, thallium, silver, sodium, potassium, and bismuth elements are of great concern. Particular emphasis should be placed on the potential biological hazard due to the alpha-particle emissions associated with the decay of ^{210}Pb, ^{210}Bi and ^{210}Po.

4.28.2 Required Data

Data is required to validate activation calculations by neutronics experiments on Na, K, N and O for the salt and on Pb (and possibly Bi) for LiPb. This can be done on NS itself, its elements or other compounds for the salt, and only on Pb (and Bi) for LiPb. The computational validation should be performed both for numerical methods and activation/decay data libraries.

It will also be necessary to determine the chemical form of products. This must be performed using the respective compounds, NS and LiPb, at temperatures anticipated for operating conditions; i.e., 350°C-500°C for NS and 380°C-530°C for LiPb. Tests should also be run to demonstrate the removal and control of gaseous products as well as solid/liquid products. This may be done in conjunction with tritium recovery experiments (see section 4.18).

4.28.3 Status of Data Base

Basic activation cross sections exist but the present data availability is insufficient to derive reliable evaluations. Information on many important reactions such as $^{39}K(n,p)^{39}Ar$ for the salt and $^{209}Pb(n,\gamma)^{210}Pb$ for LiPb is currently unavailable and needs to be developed. In addition, benchmark testing of existing numerical methods is required to validate their accuracy. For example, numerical evaluations on $^{16}O(n,^{3}He)^{14}C$ varies by several orders of magnitude among those methods currently used.

Regarding determinations of the chemical form of NS products, it is presently unknown whether carbon forms a gaseous product, CO or CO_2. CO_2 is predicted from thermodynamics, but the kinetics are unknown. No work has been done on argon or CO/CO_2 removal from NS.

Precise information is missing on the magnitude and the chemical forms of LiPb activation products. Work has not been done either on activation removal/control from LiPb.

4.28.4 Required Resources

A total effort of about 14 manyears is predicted for the task over a two-year time period with the largest effort (~ 10 manyears) on removal/control issues. New facilities will be required to study the removal and control aspects of this issue, with facility costs of up to $5M.

4.29 Electromagnetic Effects

4.29.1 Issue

Changing magnetic fields in a fusion reactor are closely coupled to the electrically conducting blanket and first wall. The primary electromagnetic concern is the arcing, forces, and torques induced in conducting components by a disruption or sudden loss of the plasma. A second major concern is facilitating the penetration of the field from the poloidal field coils of a tokamak through the shield, blanket and first wall to the plasma; this is more easily managed than the first concern. However, provision for field penetration together with requirements for maintainability places restrictions on the geometry of the first wall and blanket which can intensify the problems from plasma disruption.

4.29.2 Required Data

The required data are both experimental and computational. Most important are data on the current disruption of high-current plasmas, in particular, the time scale of the disruption and the spatial oscillations of the current during disruption. Also needed are data on the electrical properties of candidate first wall and blanket materials under the temperature and radiation conditions of expected operation. Data on arcing in intense magnetic fields in the expected conditions of vacuum and ionized gases are also needed.

Eddy-current computer codes are needed to predict the current paths, forces, torques, and voltages to be expected for different reactor geometries. These codes must be validated with data for geometries similar to those for which they will be employed. Both two-dimensional and three-dimensional codes are needed.

4.29.3 Status of Data Base

TFTR and JET should supply the needed data on plasma current disruptions. As those data are needed for many other programs as well as for electromagnetics, they will not be discussed further here.

There are some data on arcing, even some on arcing in the presence of a magnetic field, but none under the applicable vacuum and ionization conditions. Similarly, there are high temperature data on electrical properties of most materials under consideration but often not for irradiated material nor for composite, clad, or otherwise combined materials.

There are two-dimensional, finite-element eddy current computer codes which can be used to study the interactions of several axisymmetric, unsegmented systems. There are two-dimensional shell eddy current codes which can be used to study one or a few coupled, segmented systems. There are no codes which treat several coupled segmented systems. There are no three-dimensional codes which treat more than a single, simple body. The FELIX experiments are producing data to validate these computer codes.

4.29.4 Resources Required

Experiments with FELIX can provide data to validate codes on geometries similar to those which will occur in reactors (e.g., limiters and segmented blankets electrically connected at front or back). Many of these could profit from upgrading FELIX to higher fields and to better simulate disruption conditions. Experiments on arcing and other electromagnetic effects are also needed. Annual operating costs for FELIX are approximately $500K.

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BLANKET COMPARISON AND SELECTION STUDY

CHAPTER 5 - EVALUATION METHODOLOGY

CHAPTER 5

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5. EVALUATION METHODOLOGY

The Blanket Comparison and Selection Study (BCSS) devoted considerable effort to developing a comparison methodology and set of evaluation criteria which would facilitate the study's primary goal: the selection of a limited number of blanket concepts for further research and development. The evaluation and comparison process was developed as a three stage process as follows:

Stage 1: At the beginning of the study, blanket concepts were divided into mainline and alternate concepts. The determination of "mainline" designation was the judgement of the project based on previous fusion reactor and blanket design studies. Figure 5.1-1 shows the various categories of mainline concepts which are designated by liquid (Li and LiPb) or solid (Li_2O and ternary oxides) tritium breeding materials, various coolants (self-cooled liquid metals, He, and H_2O) and the possible need for a neutron multiplier (M). The reference structural material for the initial phase of the study was PCA. Design and special issue groups were then established to begin a careful evaluation of these mainline concepts. In addition, an alternate concept screening group was established to examine all blanket concepts not included in the mainline category (see Chapter X of Ref. 1).

Stage 2: In order to provide guidance for the alternate concept screening process and to provide a framework for the initial evaluation of the mainline concepts, a set of initial screening criteria was established (Table 5.1-1). These screening criteria were discussed in Sec. III.2.1 of Ref. 1. In addition, a set of design guidelines (described in Sec. 5.1) was issued near the beginning of the study for use by all study tasks and were periodically updated as required. The combined judgement of the BCSS team was used during the first year of the study to reduce the large number of possible blanket options to a limited number that were evaluated in detail during the second year of the study.

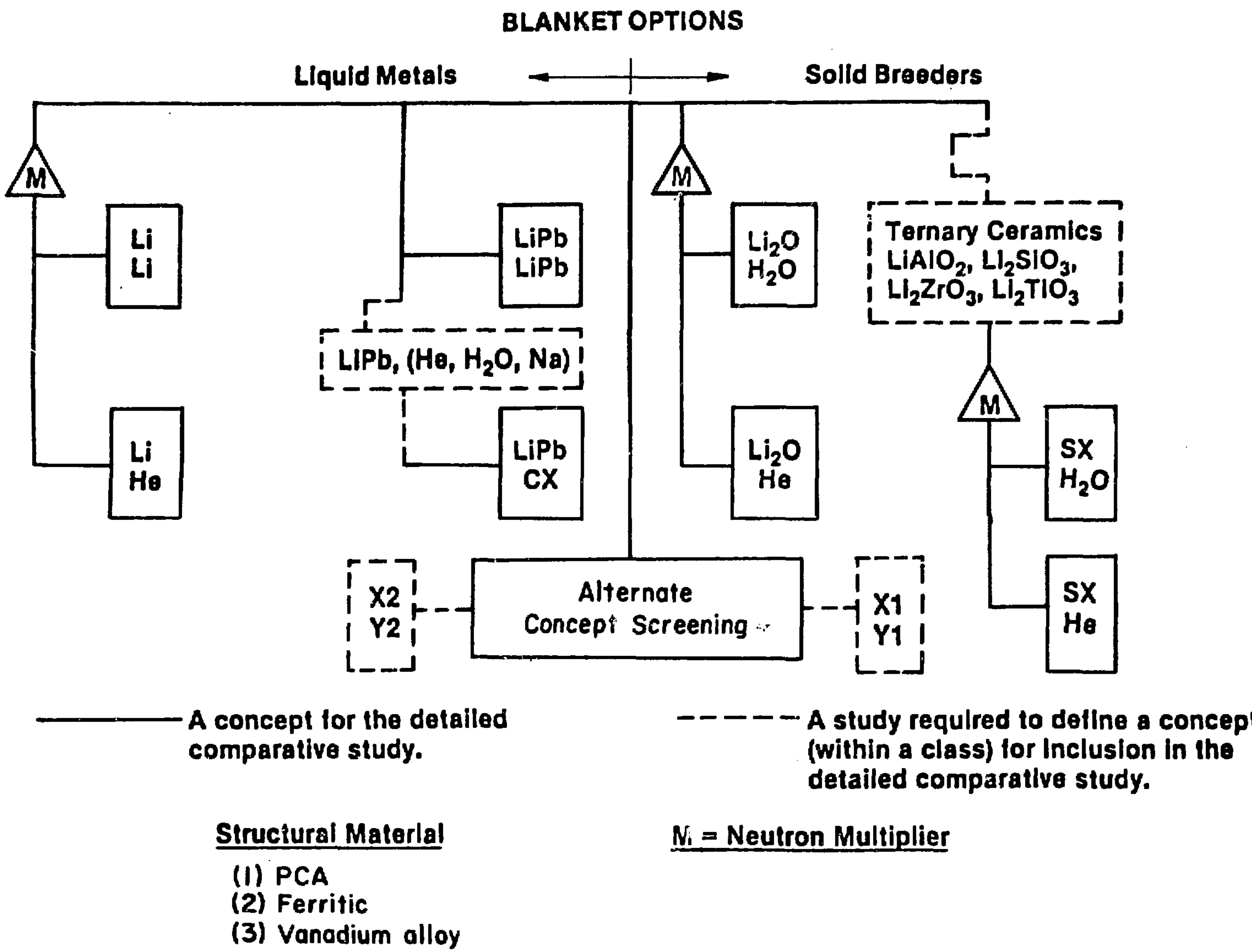


Fig. 5.1-1. Blanket options examined.

TABLE 5.1-1. INITIAL SCREENING CRITERIA

Criteria	Min./Max Value	Comments
Tritium Breeding Ratio	> 1.05 > 1.20	1-D Calculation 3-D Calculation
Thermal Efficiency	> 25%	Net Power (Gross Electric minus Pumping)[b]
Blanket Steady-State Tritium Inventory	< 10 g/MW_{th}[a]	Based Primarily on Startup and Tritium Decay Considerations
Lifetime	> 5 MW-yr/m^2	
Tritium Loss Rates (Routine Operation)	< 100 Ci/day[a]	Overall Blanket/Power Conversion System (for Plant)
Compatible Blanket Materials Combination	No Materials Combination Resulting in Large Rate of Energy Release	
Neutron Wall Loading	> 3 MW/m^2	
Engineering Feasibility	No Unduly Complex Configurations or Fabrication Procedures	

[a]For a 4000 MW_{th} power plant.

[b]Includes effect of blanket multiplication.

Stage 3: During the second year the study rated the top-ranked concepts in four areas (economics, engineering feasibility, safety and environment, and R&D requirements) which were the results of a quantified, systematic evaluation process described in Chapter 3. The methodology developed for each of these four evaluations is described in Sections 5.2 through 5.5.

5.1 Design Guidelines

Early in the study, a set of Design Guidelines was developed for use by all tasks. The purposes of these guidelines were:

(1) to establish the value (or range of values) of parameters and to state assumptions that require consistency in treating all blanket concepts, and

(2) to provide guidance on the approach to handling issues of broad interest in all concepts.

These guidelines were updated periodically to provide clarifications and additions as the need arose. Table 5.1-2 summarizes some of the major requirements from the design guidelines; these are briefly reviewed below. The final version of the design guidelines is provided in Fig. 5.1-2.

While the focus of BCSS is on the first wall and blanket, there is a need to define a reference reactor to facilitate the definition of boundary conditions and comparison of blanket concepts in terms of the impact on the overall reactor performance, safety and economics. For tandem mirrors, the MARS reactor design(2) is adopted. For tokamaks, STARFIRE(3) with limited modifications is utilized as the reference reactor design. These modifications include increasing the base case neutron wall load from 3.6 to 5.0 W/m^2, reducing the peak toroidal field from 11.1 T to 10.0 T, and reducing the number of TF coils from 12 to 10 and the number of blanket/shield sectors from 24 to 10. In addition, the tokamak was assumed to be a pulsed reactor with a burn cycle of 10^4 seconds, resulting in ~ 2500 cycles per year at an assumed 80% availability. This required that fatigue life be considered for first

TABLE 5.1-2. MAJOR REQUIREMENTS FROM BCSS DESIGN GUIDELINES

PARAMETER	TOKAMAK	TMR
Neutron Wall Load, MW/m^2	5.0	5.0
First Wall		
- Surface heat flux, MW/m^2	1.0	0.05
- Erosion rate, mm/y	1.0	0.1
Disruptions or rapid-loss-of-plasma events, no./y	3	1
Pressure on first wall due to sudden removal of plasma, MPa (psi)	0.77 (111)[c]	0.02(2.4)
Startup/shutdown cycles (warm-to-warm), no./y	12	12
Burn cycles		
- Length, s	10^4	∞
- Number per year	2500	1
Maximum allowable structure temp., ^0C		
- Austenitic stainless steel (PCA)	550	
- Ferritic steel (HT-9)	550	
- Vanadium alloy (V-15CR-5Ti)[a]	750	
Maximum allowable structure temp.[b] at liquid-to-metal interface, ^0C (for Li/17Li-83Pb at 1.5 m/s)	Li	17Li-83Pb
- Austenitic stainless steel (PCA)	470	410
- Ferritic steel (HT-9)	535	450
- Vanadium alloy (V-15Cr-5Ti)	750	650
Total allowable dose to TF coil insulator at 150 $MW\text{-}yr/m^2$ equivalent fluence at first wall, rads[d]	10^{10}	
Solid breeder allowable temperature limits, ^0C (minimum/maximum)[e]		
- Li_2O	410/800	
- $\gamma\text{-}LiAlO_2$	350/1000	

[a] Based on strength properties only.

[b] Based on uniform dissolution rate of 20 μm/y.

[c] Outboard blanket.

[d] Shield sizing criteria.

[e] See Fig. 5.1-2 for qualifications.

BLANKET COMPARISON AND SELECTION STUDY

DESIGN GUIDELINES

REV. 5 - 06 APRIL 1984

Purposes

- To establish the value (or range of values) of parameters and to state assumptions that require consistency in treating all (or most) blanket concepts.
- To provide guidance on the approach to handling issues of broad interest in all concepts.

[Note: Parameters and data specific to one blanket concept will be generated by the group working on the concept, not by Design Guidelines Group.]

Examples

- Maximum allowable temperature for structural materials
- Operating temperature range for solid breeders
- Surface heat load, erosion rate
- Divertor vs. limiter for tokamaks

GROUP ORGANIZATION

Leader: Smith/Morgan

Participants

Leader of Evaluation Criteria Group (Baker)

All leaders of individual tasks

e.g., Davis (structural material data); Liu (solid breeders data and limits)

Procedure

- Specifications for Design Guidelines will be communicated to all project members by Smith or Morgan.
- Guideline revisions will be provided in the future, for the duration of the project.

How parameters and assumptions are added to the list

- Based on requests initiated by the project manager or any member of the project.
- The technical specifications will be developed by the appropriate group (e.g., solid breeder tritium recovery group).
- The requests can be made to Smith, Morgan, or the appropriate group leader.
- The final specifications will be reviewed and issued by the Design Guideline Group Leader.
- Any member of the project can suggest changes. All requests will be reviewed.

Fig. 5.1-2. BCSS Design Guidelines (final version)

Parameter(s): Average Neutron Wall Load

Value: 5 MW/m^2 [Note: A range of values may be added later.]

Parameter: Surface Heat Load

Value: For tokamaks, consider 10, 50, and 100 W/cm^2. For cases where large changes occur in the results, also consider intermediate values (25 W/cm^2 and/or 75 W/cm^2). In cases where it is necessary to consider a single value for the surface heat load, assume 50 W/cm^2 for preliminary phases. Final evaluations will consider each concept's ability to achieve 100 W/cm^2 as a reference value.

For tandem mirror reactors (TMRs): consider 5 W/cm^2.

Parameter: Frequency of Disruptions or Rapid-loss-of-plasma events

Value: For computing fatigue life, the frequency per year of such events shall be assumed as follows:

DEVICE	SU/SD(1)	DISRUPTION OR RAPID PLASMA LOSS	BURN CYCLES(2)
TMR (SS)	12	1	0
TOKAMAK (PULSED)	12	3	2500

(1) STARTUP/SHUTDOWN, TYPICAL OF POWER PLANT OPERATING EXPERIENCE

(2) OTHER THAN SU/SD. BASED ON 10^4s BURN TIME, 80% AVAIL.

This assumes an improvement in disruption frequency in tokamaks to the degree necessary for viable power reactors. Rapid-loss-of-plasma events are not typically observed in TMR's but are included to guard against designs which cannot withstand such low-probability events.

Parameter: First Wall (FW) Thickness

Approach: Determine the maximum allowable thickness for each value of the surface heat load, accounting for constraints specified below and elsewhere in the guidelines. Consider all material to be continuous with first wall. The following guidelines shall apply:

- o Erosion rate (FW plasma-facing surface) = 1 mm/calendar year for tokamaks, 0.1 mm/calendar year for TMR's.
- o Orthogonal grooving may be assumed for tokamaks or TMRs to permit a thickness allowance of material for erosion of the plasma-facing surface. For a given wall thickness, the maximum structural temperature limit (e.g., 550°C for HT-9) shall be met at a distance of three times the thickness of the structural (non-grooved) portion of the wall, as measured from the coolant interface.

Fig. 5.1-2 (Continued)

STRUCTURE MAXIMUM TEMPERATURE LIMITS[a] AT LIQUID METAL INTERFACE (°C)

System	Flow Velocity, m/s	Austenitic Steel (PCA)[b]	Ferritic Steel (HT-9)	Vanadium Alloy[c,d] (V-15Cr-5Ti)
Lithium				
Circulating	1.5	430 (470)	535	750
	0.5	445 (480)	550	
	0.05	455 (495)	565	
Static	-	525 (575)	565	750
Pb-17Li				
Circulating	1.5	375 (410)	415 (450)	> 650
	0.5	385 (420)	425 (465)	
	0.05	395 (430)	435 (475)	
Static	-	395 (430)	435 (475)	> 650

[a] Limits based on a uniform dissolution rate of 5 μm/y (or ~5.5 mg/m^2-h). The values within brackets correspond to a rate of 20 μm/y.

[b] Temperature limits for Pb-17Li system are for 20% CW Type 316 SS.

[c] Nonmetallic element transfer are expected to dominate corrosion limits.

[d] Temperature limits based on materials considerations only. Safety considerations for use of vanadium alloy where contact with oxidants can occur in accidents may require temperature limit $\leq$ 650°C.

Parameter(s): Maximum Allowable Structure Temperature
Maximum Allowable Temperature at Structure Interface with Coolant or Breeder

Approach: The lifetime of any structural material varies with the operating temperature. This relationship is provided at discrete values of temperature for all candidate alloys under "Stress Limits."

STRUCTURE MAXIMUM TEMPERATURE LIMITS[a] (°C)

Austenitic Steel[b] (PCA)	Ferritic Steel (HT-9)	Vanadium Alloy[c] (V-15Cr-5Ti)
550	550	750

[a] Minimum temperature limits (if any) not included.

[b] Annealed or 20% CW.

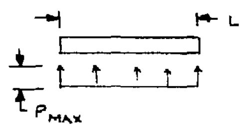
[c] Temperature limits based on materials considerations only. Safety considerations for use of vanadium alloy where contact with oxidants can occur in accidents may require temperature limit $\leq$ 650°C.

Parameter: Downtime for Blanket Failures (Unscheduled Maintenance)

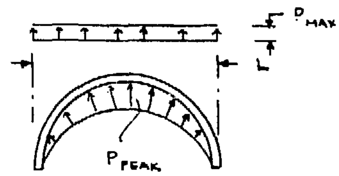
Value: Use goal of $\leq$10 days per calendar year.

Fig. 5.1-2 (Continued)

Parameter:	Forces in First Wall Due to Sudden Removal of Plasma
Value:	Forces caused by toroidal (TMR:axial) eddy currents shall be equivalent to those caused by a distributed pressure acting inward toward the plasma chamber.



FLAT FIRST WALL



LOBULAR FIRST WALL

o TOKAMAK:P_{MAX} = 111 PSI (OUTB'D)
98 PSI (INB'D)

o TMR:P_{MAX} = 2.4 PSI

(P_{PEAK} = maximum value for cosine pressure distribution which causes unit force equivalent to (P_{max}) X (L).

For a multi-layer tube bank first wall and blanket in a TMR, the added pressure will be 240 psi which will appear in the tubes if they carry a liquid metal. For water or helium coolant the forces will be distributed among all layers of the tube bank.

Forces and torques produced by radial or poloidal eddy currents (interacting with the toroidal or axial field) shall be evaluated for TMR's and for inboard (tokamak) and top/bottom/outboard regions in accordance with the memo of Turner and Moir dated March 27, 1984.

Parameter:	Shield Sizing Criteria
Approach:	The shield thickness for any specific blanket design shall be sized to produce a total dose to the TF coil insulator of no more than 10 X 10^9 rads, over a lifetime equivalent influence to 150 MW-yr/m^2 (40 yr plant life X .75 availability factor X 5 MW/m^2 neutron wall load).

Parameter:	Tokamak Burn Cycle
Approach:	The reference tokamak for all BCSS analyses will be a very-long-pulsed tokamak with these burn cycle parameters:

Fusion Period	$t_F = 1 \times 10^4$ s
EFC Ramp Down	$\Delta t_{EF} = 5$ s
OHC Reset	$\Delta t_{014} = 30$ s
EFC Ramp Up	$\Delta t_{EF} = 5$ s

Fig. 5.1-2 (Continued)

Parameter: Total First Wall Plus Blanket Thickness
(Does not include shield thickness)

Approach: Optimize but limit to $\leq$ 70 cm if possible. (If thicker blanket is necessary, then trade must be made with shield thickness. Dose rate to SC magnets will govern total blanket/shield thickness for a given concept.) Larger thickness may:

- Increase reactor capital cost
- Increase tritium breeding ratio
- Increase the fraction of thermal power recovered as useful energy

Energy deposited in the shield is by definition not recoverable. All (neutron) energy recovery regions (breeder, multiplier, reflector, etc.) should be included in the first wall and blanket.

Parameter: Tritium Breeding Ratio (TBR)

Approach:

- Each blanket concept will be judged by its Tritium Breeding Potential (among other factors).

- Example (not a specification): $(TBR)_N$ = maximum net TBR
  - $(TBR)_N < 1.05$ Rejected
  - $1.05 \leq (TBR)_N < 1.10$ High Risk
  - $1.10 \leq (TBR)_N < 1.15$ Medium Risk
  - $1.15 \leq (TBR)_N < 1.20$ Low Risk
  - $(TBR)_N \geq 1.20$ Attractive

- For tokamaks and TMR's, concept designers should ensure that the breeding ratio with full coverage is greater than 1.2. Examine methods to increase the breeding ratio, i.e., evaluate breeding potential.

- The reference limiter for 3-D neutronics analyses of blankets in tokamaks shall be the water-cooled FED-INTOR type of limiter shown in these Guidelines.

- Tritium Breeding Issues Group is examining:
  - Breeding requirement (depends on T inventory, doubling time, reactor type, blanket type, etc.)
  - Impact of overall reactor on breeding ratio

Parameter: Tokamak Blanket L/R Time Constant (τ)

Value: 100-200 ms (desired range in a segmented blanket)
Note: It is strongly preferred that the eddy current path be constrained to remain near the first wall.
(See L. R. Turner memo of 26 July 1983)

Fig. 5.1-2 (Continued)

Parameter: Stress Limits

Approach: The allowable stresses for all three candidate structural alloys in both unirradiated and irradiated conditions are shown in the following tables. Both primary and primary-plus-secondary stress limits are to be met for all FW and blanket structure, including any FW erosion allowance. (Background and rationale for these limits are discussed in Saurin Majumdar's memo of June 1, 1983.)

ALLOWABLE STRESSES FOR PCA (25% CW)

Temp.	Unirradiated ($\leq$ 10 dpa)					Irradiated (> 10 dpa)		
	S_m	S_{mt} [a] (MPa)				S_m	S_{mt} [b] (MPa)	
(°C)	(MPa)	10^4 h	2×10^4 h	3×10^4 h	5×10^4 h	(MPa)	75 dpa	100 dpa
100	225	225	225	225	225	225	190	155
200	208	208	208	208	208	208	190	155
300	205	205	205	205	205	205	190	155
400	205	205	205	205	205	205	190	155
500	205	205	205	205	205	190	185	150
550	195	195	195	190	175	175	150	130

[a] For $t \leq 2.8 \times 10^4$ h, $S_{mt} = S_m$ (unirradiated).

[b] For dpa $\leq$ 70, $S_{mt} = S_m$ (Irradiated) based on a maximum strain limit of 5%.

ALLOWABLE STRESSES FOR HT-9 (NORMALIZED AND TEMPERED)

Temp.	Unirradiated					Irradiated				
	S_m	S_{mt} (MPa)				S_m	S_{mt} (MPa)			
°C	(MPa)	10^4 h	3×10^4 h	10×10^4 h	30×10^4 h	(MPa)	50 dpa	75 dpa	100 dpa	150 dpa
20	250	250	250	250	250	250	235	198	163	125
200	220	220	220	220	220	220	220	198	163	125
300	210	210	210	210	210	210	210	198	163	125
400	200	200	200	200	200	200	200	198	163	125
500	175	160	150	140	125	175	175	175	163	125
550	160	110	90	80	75	160	160	160	150	115
600	140	60	50	40	30	140	105	80	70	50

ALLOWABLE STRESSES FOR V-15Cr-5Ti

Temp.	Unirradiated (< 10 dpa)					Irradiated (> 10 dpa)				
	S_m [a]	S_{mt} (MPa)				S_m [a]	S_{mt} (MPa)			
(°C)	(MPa)	10^4 h	2×10^4 h	3×10^4 h	5×10^4 h	(MPa)	100 dpa	150 dpa	200 dpa	250 dpa
20	275	275	275	275	275	275	165	125	105	90
200	250	250	250	250	280	250	165	125	105	90
100	240	240	240	240	240	240	165	125	105	90
400	230	230	230	230	230	230	165	125	105	90
500	220	220	220	220	220	220	165	125	105	90
600	235	235	235	235	235	235	165	125	105	90
650	235	235	235	235	235	235	165	125	105	90
700	235	230	180	175	155	235	165	125	105	90
750	230	160	125	115	95	230	155	115	95	80

[a] Since these S_m values are based on average rather than equal minimum values of S_u from a single heat of material they are equal to $1/3\ S_u$ and not $1.1/3\ S_u$.

Fig. 5.1-2 (Continued)

Parameter: Allowable Sector Deformation in Toroidal Direction Due to Time-Dependent Effects

Value: For the width of a single sector (36° circumferentially around the torus), the maximum allowable deformation as measured for the widest part of the sector (outboard midplane) shall be 2.0 cm. This value shall be the difference between the widths measured (1) at steady-state operating conditions during initial operation (zero fluence, BOL) and (2) at steady-state operating conditions at maximum fluence (EOL).

Parameters: Properties and Operating Temperature Ranges for Solid Breeders

Value(s):

VALUES FOR CANDIDATE SOLID BREEDER MATERIALS[a]

	PROPERTIES			RECOMMENDED TEMPERATURE LIMITS		INFORMATION ONLY		
Breeder	MP, °C	ρLi, g/cm³	K[b], W/m-K	T_{min}, °C	T_{max}, °C	ΔT, °C	K·ΔT	ρLi·K·ΔT
Li_2O	1433	0.93	3.4[i]	410[d]	800[c,h]	390	1325	1230
γ-$LiAlO_2$	1610	0.28	2.2[i]	350[d]	1000[f]	650	1430	400
Li_5AlO_4	1047	0.61	2.3	350[d]	780[g]	430	989	603
Li_2SiO_3	1200	0.36	1.5	410[d]	1000[f]	590	885	319
Li_4SiO_4	1250	0.54	1.5	320[d]	950[f]	630	945	510
Li_2ZrO_3	1616	0.33	1.3	400[e]	1400[f]	1000	1300	429
Li_8ZrO_6	1295	0.68	1.5	350[e]	980[g]	630	945	643
Li_2TiO_3	1550	0.33	2.0	400[e]	1185[g]	785	1570	518

[a] Separate effects data not available, therefore effects of radiation on temperature limits is unknown and not reflected in this table.

[b] Estimated for sintered product at 85% smeared density at 1000 K.

[c] Established from chemical considerations, i.e., reaction with moisture to form LiOT.

[d] Established from diffusion/inventory considerations, based on 0.2 μm grain size.

[e] Estimated assuming similar properties.

[f] Estimated for thermal sintering limit.

[g] Estimated assuming $T_{max} = 0.8\ T_m$, K.

[h] 1000°C for design approaches with helium purge gas flow directed only to the "cold" region of the breeder.

[i] For estimates of K for sintered Li_2O and sphere-pac Li_2O and γ-$LiAlO_2$ with irradiation effects, guidance will be provided by the Solid Breeder group.

Fig. 5.1-2 (Continued)

Tokamak Impurity Control

The reference limiter for the BCSS is based on the FED/INTOR Phase 2A Study, 1982. This is a bottom limiter with a curved configuration. The reference limiter is constructed of a copper heat sink with water coolant (T_{max} = 150°C) and a 10 mm beryllium coating. An alternate limiter for liquid metal blankets will use lithium coolant (T_{max} ~ 250°C), vanadium allow (V-15Cr-5Ti) or tantalum alloy (Ta-10W) heat sink, with a 10 mm beryllium coating. The figure below is a schematic of the FED/INTOR limiter with approximate dimensions for the BCSS limiter.

For the BCSS limiter it is assumed that the total fusion power is 5000 MW, that the major radius is 7.2 m and that the limiter radial width is 2.0 m. The total area of the limiter is:

$$A_{BCSS} = \pi\,(8.2^2 - 6.2^2) \simeq 90\ m^2.$$

For BC55, the case considered for the power levels to the limiter and first wall shall be as follows:

80% of the α-power uniformly radiated and 20% in particle power to limiter.
Total power to limiter = 280 MW (200 particles + 80 radiation).
Average power = 3.12 MW/m^2.
Peak power = 4.12 MW/m^2.

BCSS REFERENCE LIMITER FOR TOKAMAKS

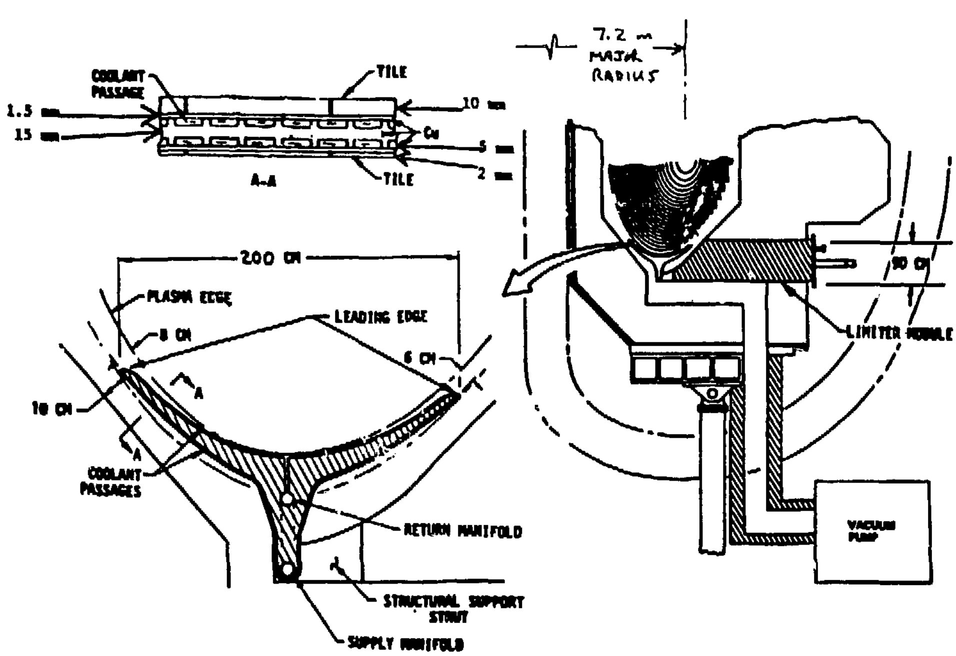


Fig. 5.1-2 (Continued)

<u>Reactor Configuration</u>

For designs that may be strongly influenced by the reactor configuration (e.g., MHD effects for flowing liquid metals), consider:

Tokamaks: STARFIRE (with 10 DEMO-type TF coils, 10 T max. field)

Mirrors: MARS (to be provided)

Parameter:	Loads Due to Seismic Events
Value:	Apply ±4.4 g's in 1, 2 or 3 orthogonal directions simultaneously, on a worst-case basis, to any detail part, subassembly, or assembly. The resulting stresses shall be combined with those resulting from normal operating conditions and compared to limits applicable for normal operation, except that S_m may be used instead of S_t or S_{mt}.

Parameter:	Steady-state Magnetic Forces on Ferritic Steel Structure in Tokamaks
Value:	For an <u>approximation</u> of the induced forces, use the magnetic force factor (weight multiple for <u>structure</u>) shown in Figure 1 below, except that the factor should be reduced by 10% to account for the reduced field for TF coils in the BCSS compared to Starfire. Directions for forces (which act on all structure) are shown in Fig. 2. (Note: Forces also occur in other directions but are not considered significant for our purposes.)

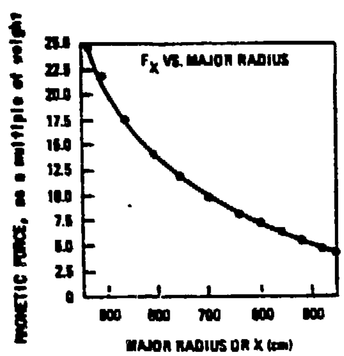
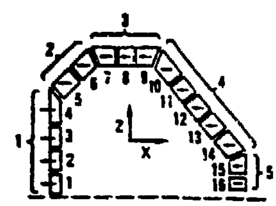


Fig. 1

The force distribution in the direction of major radius on blanket module.



(b)

Fig. 2

Schematic showing position and forces on blanket modules: (a) shows forces in Y-direction, and (b) shows forces in X-direction. Brackets indicate full blanket modules while small numbers refer to elements used in GFUN calculation.

(Source: T. L. Lechtenberg, C. F. Dahms, "Magnetically-induced Forces on a Ferromagnetic HT-9 First Wall/Blanket Module")

Fig. 5.1-2 (Continued)

walls and blankets and the inclusion of thermal storage capability in the power conversion system for each blanket concept.

The reference neutron wall load of 5 MW/m^2 is presently believed to be near the upper end of the optimum range for tandem mirrors and tokamaks. While all the mainline and alternative concepts have been evaluated at 5 MW/m^2, a limited effort was devoted during the study's first year to evaluating first wall/blanket concepts at higher wall loadings. This effort is summarized in Appendix D of Ref. 1. The results indicated that for the present candidate materials and reactor design concepts, no clear benefits can presently be identified in terms of the cost of energy for operating at wall loads significantly greater than 5 MW/m^2.

The design of the first wall is greatly influenced by the value of the surface heat load and the rate of the wall erosion. There are considerable differences in this area between tandem mirrors and tokamaks. In the MARS design, the surface heat flux is ~5 W/cm^2. The charged and neutral particle fluxes at the first wall are very low, and the resulting erosion rate was assumed to be 0.1 mm/y. In contrast, tokamak designs have shown high surface heat fluxes and in some cases high erosion rates at the first wall. There is a trade-off possible between the heat load on the first wall and that transported to the limiter or divertor plates. Since limiters and divertor plates have limited surface area, the results of the tradeoff generally favor a large fraction of the α-power being radiated to the larger surface area of the first wall. The study selected a limiter located at the bottom (INTOR type) and assumed the following case for a total fusion power of 5000 MW:

- Major radius = 7.2 m, limiter radial width = 2.0 m
- 80% of the α-power uniformly radiated to first wall, 20% in particle power to the limiter
- Surface heat flux at the first wall = 1 MW/m^2
- Total power to limiter = 280 MW (200 MW particles, 80 MW radiation)
- Limiter power loading = 3.12 MW/m^2 average, 4.12 MW/m^2 peak

The erosion rate is determined primarily by the magnitude of the flux and energy of the charge-exchange neutrals at the first wall. Previous reactor studies predicted erosion rates at the first wall as high as 10 mm/y. More comprehensive modeling performed for INTOR Phase II[4] indicate that the charge exchange flux is very low, hence the erosion rate on the first wall will be low (~1 mm/y) except for localized areas near the limiter tips and divertor throat. Thus, the final reference values adopted for tokamak first wall surface heat flux and erosion rate were 1.0 MW/m^2 and 1 mm/y, respectively.

The maximum structure bulk temperature limits were determined by the Structural Materials group as described in Sec. 6.1. The temperature limits adopted on the basis of strength properties were 550^0C, 550^0C, and 750^0C for PCA, HT-9 and V-15Cr-5Ti, respectively. The allowable stresses for these structural materials in both the irradiated and unirradiated conditions are given in Sec. 6.7 and in Fig. 5.1-2. In addition, maximum temperature for V-15Cr-5Ti was constrained to be 650^0C or less for situations where contact with oxidants could occur in accidents, because of safety concerns for rapid oxidation of vanadium alloys.

Evaluation of liquid metal corrosion (Sec. 6.2) provided limits on structure temperature at the liquid metal interface. Depending on the velocity of the liquid metals, these interface temperature limits for austenitic steels are in the range of 470^0C to 495^0C for lithium and 410^0C to 430^0C for 17Li-83Pb (LiPb). These limits are based on a maximum allowable uniform dissolution rate of 20 μm/yr. The interface temperature limits for lithium are higher by ~270^0C and ~80^0C for vanadium alloys and ferritic steels, respectively. However, in the case of the more corrosive LiPb, the austenitic steel temperature limits go up by ~230^0C for the vanadium alloys but only by ~40^0C for ferritic steels. These corrosion temperature limits proved to be among the most critical drivers for liquid metal designs with PCA and HT-9.

The lower and upper temperature limits for solid breeders were established by the Solid Breeder Materials group (Sec. 6.3). The specified temperature limits are given in Sec. 6.7. Recent experimental and analytical results led to specifying temperature limits for solid breeders that, in general, resulted in wider temperature windows (allowable temperature ranges) than those assumed in most previous studies.

The specifications for plasma disruptions in the case of tokamaks and rapid-loss-of-plasma in the end plugs in the case of TMR's were examined. The first wall is required to withstand only a few major disruptions or plasma loss occurrences during the lifetime of the blanket, ~5 for TMR's and ~15 for tokamaks. This assumes an improvement in disruption frequency in tokamaks to the degree necessary for viable power reactors. Thus, the first wall erosion resulting from the thermal energy deposition is not significant. However, the requirement to withstand the electromagnetic forces induced in the first wall is an important constraint that has to be satisfied by all design concepts. Other electromagnetic requirements for all blanket concepts are listed in Fig. 5.1-2.

REFERENCES - SECTION 5.1

1. M. A. Abdou et al., "Blanket Comparison and Selection Study - Interim Report," report ANL/FPP-83-1 (Vols. I and II), Argonne National Laboratory (October 1983).

2. "Mirror Advanced Reactor Study Final Design Report," report UCRL-53480, Lawrence Livermore National Laboratory (July 1984).

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

4. W. Stacey et al., "U.S. FED-INTOR Critical Issues," report U.S.A. FED-INTOR/82-1, Georgia Institute of Technology (1982).

5.2 Engineering Feasibility

The items included under "Engineering Feasibility" include important, even vital, blanket criteria that either deserve separate consideration or do not readily fit under the categories of safety and economics. These items are listed in Table 5.2-1.

The results for most of these criteria would be quantifiable in economic terms if analyses and trade-off studies were performed in much greater depth, and if results of blanket development tests were available. The related rankings among blanket concepts could then be made on the basis of economics. For the purposes of the BCSS, however, the blanket concepts must be examined for each criteria from the standpoint of engineering judgement on the basis of information provided by the concept advocate groups. In this manner, relative differences among the concepts in these areas can be ascertained and a relative ranking thus established.

The basic approach to be followed is similar to that used for safety and environmental criteria as described in Sec. 5.4. The overall engineering figure of merit (EFM) is defined as the weighted (W_i) sum of an index (I_i) for each item listed above:

$$EFM = \sum_i I_i W_i \qquad [5.2\text{-}1]$$

where I_i has a value of 0 to 1 for each item listed in Table 5.2-1. The maximum score is 100. The weighting values, W_i, are also given in Table 5.2-1.

Because the method uses weighted sums rather than weighted products, a zero in one of the indices of Table 5.2-1 would not result in an overall value of the EFM of zero. If the reason for assigning a zero value for a particular index was judged to be so serious that it is considered a fatal flaw for that particular blanket concept, then the concept would not be further considered in the evaluation process. This has not been a problem primarily because initial screening and selection process of the first year of the study eliminated nearly all blankets having fatal flaws.

TABLE 5.2-1. ENGINEERING FEASIBILITY EVALUATION INDICES

INDEX NAME		WEIGHTING VALUE (W_i)
1. Tritium Breeding and Inventory		25
2. Engineering Complexity and Fabrication		25
3. Maintenance and Repair		15
4. Use of Resources		5[a]
5. Accommodation of Power Variations		10
6. Increased Capability		10
6.1 Increased Neutron Wall Loading	5	
6.2 Higher Surface Heat Flux, Higher Erosion Rates	5	
7. Startup/Shutdown Requirements		10

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

5.2.1 Tritium Breeding and Inventory

Obtaining adequate tritium fuel self-sufficiency is clearly a major feasibility issue(1) and has been given a weighting value equal to one-fourth of the total possible points. This subsection briefly smmarizes the technical basis for the tritium breeding index developed in the study. Section 6.8 presents a detailed discussion of breeding requirements and uncertainties.

Tritium Breeding Requirement - The required tritium breeding ratio (T_r) in a self-sustained fusion power economy must exceed unity by a margin (G) to cover losses due to radioactive decay during the period between production and use, to supply inventory for startup of other fusion reactors, and to provide some reserves for periods of scheduled maintenance or failures of the fuel processing subsystem. G is often called "doubling time margin". The equation for T_r can be written as

$$T_r = 1 + G_o \qquad [5.2\text{-}2]$$

where G_o is the breeding margin for a reference conceptual design. G_o is a function of the reactor tritium inventory (I) and the doubling time (t_d), and can be written as

$$G_o = \frac{I}{\tau N^-} \times F(t_d) \qquad [5.2\text{-}3]$$

where τ is the mean decay time of tritium, N^- is the rate of tritium consumption and $F(t_d)$ is a function of t_d. G_o increases rapidly as I increases and t_d decreases. I includes the tritium inventory in the blanket, fueling and exhaust systems, in other reactor components, and the storage inventory for use in off-normal conditions and for startup of a new reactor.

Uncertainties in Breeding Requirements - The uncertainty in determining the required breeding ratio T_r is related directly to the uncertainty, Δ_G, in the estimated required doubling time G_o. In turn, Δ_G is primarily a function of the uncertainty in inventory I (Eq. 5.2-3), since τ is well known, N^- can be calculated precisely, and we can arbitrarily specify our required doubling time t_d.

At present, there are large uncertainties in estimating I. For example, the magnitude of the tritium inventory retained in a solid breeder blanket is probably uncertain by at least one order of magnitude. Similar uncertainties exist for the fueling and exhaust system because of lack of information on the achievable tritium fractional burnup in the plasma, and for the fuel processing subsystem because of uncertainties in defining economically feasible efficiencies for tritium recovery methods in power reactors. Therefore, the magnitude of the required tritium breeding ratio (T_r) of future power reactors is uncertain today. This uncertainty is designated by the parameter Δ_G.

Uncertainties in Calculated Breeding Ratio - Prior to the construction of an actual fusion reactor, the only means to evaluate the breeding potential of a given blanket concept in a preliminary fusion reactor design is to calculate a tritium breeding ratio (T_c) using present codes and data. The calculated breeding ratio and achievable breeding ratio (T_a) can be related as

$$T_a = T_c - \Delta_c \qquad [5.2\text{-}4]$$

where Δ_c is the uncertainty in the calculated breeding ratio. The sources of uncertainties that contribute to Δ_c are numerous but are here broadly classified into two areas: (1) reactor design definition (Δ_s) and (2) neutronics calculations, nuclear data, and modeling (Δ_p). The methodology used to determine Δ_G, Δ_s, and Δ_p is thoroughly discussed in Sec. 6.8.

Risk in Achieving Required Breeding Ratio - The uncertainties Δ_G and Δ_c previously defined represent a desired margin in the calculated breeding ratio T_c over and above T_r (the minimum breeding ratio requirement if all uncertainties are zero) for a given blanket concept and a specific reactor system. If

$$(T_c - T_r) \geq (\Delta_G + \Delta_c), \qquad [5.2\text{-}5]$$

then we would presently have high confidence that the blanket and reactor system for which T_c was calculated would ultimately meet or exceed T_r. At the other extreme, if T_c is exactly equal to T_r, then we would presently have low or zero confidence that T_r would ultimately be met. (If T_c is less than T_r, the concept by our definition is not feasible.) The relative magnitude of the ratio $(T_c - T_r)/(\Delta_G + \Delta_c)$ can thus be considered a measure of the reduction in risk in ultimately achieving the required breeding ratio in a reactor.

During the first year of the study[2], the risks for various concepts were determined and classified as follows. Rewriting Eq. [5.2-5] and using Eq. [5.2-2], we have:

$$T_c \geq 1 + G_o + \Delta_G + \Delta_c \qquad [5.2\text{-}6]$$

where G_o was estimated to be 0.05, Δ_c was estimated to be in the range of ~0.05 to 0.15, and Δ_G was estimated to be in the range of -0.04 to +0.25. Based on these reference conditions, concepts were classified based on the calculated tritium breeding ratio T_c (obtained from 3-D calculations for the detailed reference design) as follows:

$T_c \geq 1.2$	Low Risk
$1.1 < T_c < 1.2$	Medium Risk
$1.05 < T_c < 1.1$	High Risk
$T_c < 1.05$	Reject ($T_c < T_r$)

For evaluation of the smaller number of concepts considered during the second year of the study, an equation for a TBR figure of merit, F, was developed as

$$F = \frac{T_c - (1 + G_o)}{\{[(\Delta_G)(1+G_o)]^2 + [(\Delta_s)(T_c)]^2 + [(\Delta_p)(T_c)]^2\}^{1/2}} \qquad [5.2\text{-}7]$$

where all terms have previously been defined. The equation for F is similar to Eq. [5.2-5] except that the uncertainty terms have been summed and the root of the sum is used as the denominator. It is recognized that this does not represent a fully rigorous statistical treatment of the uncertainties; see Sec. 6.8.

The score for index I_1 is equal to F (see Table 5.2-2).

TABLE 5.2-2. FIGURE OF MERIT AND SCORING FOR TBR

$$I_1 = F = \frac{T_c - (1 + G_o)}{\{[(\Delta_G)(1+G_o)]^2 + [(\Delta_s)(T_c)]^2 + [(\Delta_p)(T_c)]^2\}^{1/2}}$$

T_c	= Net TBR calculated for the blanket under consideration in 3D geometry for reference reactor conditions (e.g., MARS(3) with a set of assumptions about design choices; or STARFIRE(4) with specified limiter, lower hybrid, etc.)
G_o	= Required doubling time gain under reference conditions and assumptions
Δ_G	= Uncertainty in predicting required doubling time margin
Δ_s	= Uncertainty associated with system definition
Δ_p	= Uncertainty in predicting TBR for a given system

[a]Based on Gaussian distribution.

Estimate of G_0 - The conditions assumed for the calculation of G_0 are discussed in detail in Sec. 6.8. The assumptions which have the strongest influence on G_0 are doubling time (5 yr), tritium fractional burnup in the plasma (5% for tokamaks and TMR's), tritium reserve inventory (2 days equivalent throughput), and 0.1% non-radioactive losses due to processing inefficiencies.

Estimates of Uncertainties - The calculated and assumed values used to determine the values of Δ_G, Δ_s, and Δ_p for each blanket concept evaluated are discussed in Sec. 6.8. By far the most influential uncertainty of the three at this time is Δ_G, for which a value of 0.224 was selected. The value of Δ_G is very sensitive to the uncertainties, and distribution of uncertainties, assumed for the various factors which are considered in determining G_0. The values of Δ_s and Δ_p are lower by comparison primarily because the most important factors involved are calculable today with relatively small uncertainties.

5.2.2 Engineering Complexity and Fabrication

The basic engineering design and required fabrication procedures for the blanket are clearly important considerations and have been accorded the same weighting value as tritium breeding. This index is much more difficult to quantify than breeding ratio at this stage of blanket development and thus requires a different approach. The approach that we have selected is to identify eight important features (see Table 5.2-3) which, if the best for each were realized in a blanket, would reduce its engineering and fabrication complexity. Each of the first seven features can score a maximum of three points and the eighth can score six, for a total maximum score of 27. While it is possible to consider some weighting factors among these eight features, this was not done in order to keep the procedure from becoming too complex. The index is obtained by dividing the score by 27 so that the maximum value of the index is 1.0.

It should be noted that aspects of engineering complexity also appear in other evaluation categories. For example, estimates of capital cost which includes consideration of fabrication difficulties is considered under economics. In addition, issues related to failure frequency and mode, which are

TABLE 5.2-3. FEATURES FOR ENGINEERING COMPLEXITY AND FABRICATION

			Score
(1)	First Wall[a] (FW)		
	- Requires stand-alone FW		0
	- Integral FW, separate coolant loop		1
	- Integral FW, separate or successive flow path with blanket		2
	- Completely integrated FW/blanket		3
(2)	Neutron Multiplier[a] (NM)		
	- Separate zone of NM, complex requirements		0
	- Separate zone of NM, simple requirements		1
	- NM integrated with breeder, but special requirements on NM fab. or assembly		1.5
	- NM integrated with breeder, simple requirements on NM		2
	- No NM required or NM is part of the breeder compound/eutectic		3
(3)	Breeder Fabrication[a]		
	- Many small pieces or containment requirements are complex		0
	- Few large pieces or containment requirements are simple		1
	- Complex module fill operations		1.5
	- Simple module fill operations		2
	- Combined breeder and coolant		3
(4)	Coolant Containment/Flow Path Requirements[b]		
	- Pressure confined by module walls:	o High pressure	0
		o Medium pressure	0.5
		o Low pressure	1
	- Containment fabrication:	o High complexity, many pieces	0
		o Moderate complexity	.5
		o Low complexity, few pieces	1
	- Coolant flow path requirements:	o Many paths, balancing required	0
		o Few or single path, little or no balancing	1
(5)	Tritium Removal[a]		
	- Many connections, He purge		0
	- Few connections, He purge		1
	- Breeder/coolant circulation		2
	- Passive system (e.g., permeation)		3

TABLE 5.2-3. FEATURES FOR ENGINEERING COMPLEXITY AND FABRICATION (CONT'D.)

	Score
(6) Manifolding[a]	
- Separate from blanket, many connections	0
- Separate from blanket, few connections	1
- Structurally integrated; connections to blanket	2
- Fully integrated, no connections	3
(7) Inboard/Outboard Blanket[a]	
- Different inboard blanket coolant with different breeder	0
- Different inboard blanket coolant with same breeder	1
- Inboard blanket coolant and breeder same as outboard but with different configuration	2
- Inboard blanket same as outboard blanket	3
(8) Manufacturing Operations[b] [3 parts]	
Coatings:	
- Separate material applied	0
- Natural oxides needed	0.5
- No requirements for coatings or oxides	1
Weld requirements	
- Number of welds, lengths, criticality of a leaking weld to continued system operation	0 to 1.5
- Special welding requirements such as inert atmosphere, pre- or post-weld heat treatment required	0 to 1.5
Machining requirements (any shaping of metal, or metal removal from raw stock)	
- Complexity and number of special operations required on details or subassemblies	0 to 2

[a] Select one value from 0 to 3 from those given.
[b] Select one value from the range for each of three parts.

intimately related to engineering complexity, will be considered in the safety evaluation.

5.2.3 Maintenance and Repair

This category evaluates various features of a blanket concept that impact the ability to maintain and repair the blanket. As with item 2 - Engineering Complexity and Fabrication - this is very difficult to quantify at this stage of reactor design and blanket maintenance development. Table 5.2-4 lists the maintainability features and the scores used to determine a maintenance and repair index.

TABLE 5.2-4. FEATURES FOR MAINTENANCE AND REPAIR

Desired Feature	Maximum Score
o Replacement operations involve simple push/pull movements and few disconnects.	0.3
o Simple inspection procedures including vacuum leak checking.	0.2
o Coolant spills result in simple clean-up operations.	0.2
o Blanket can continue to operate with a few coolant tube failures or does not require a large number of tubes.	0.3

5.2.4 Use of Resources

For certain blanket concepts, the use of potentially limited resources is an important consideration, e.g., the use of Be as a neutron multiplier. We also considered lithium resources but did not consider tritium resources required for startup, which was a factor in the tritium breeding and inventory index, Sec. 5.2.1. The approach is to consider the most limiting material(s) in the overall blanket design as a fraction of U.S. reserves and world resources, where a total fusion economy of 1000 Gwe (or about 833 STARFIRE-size reactors with a 40 year lifetime) is assumed. The values of the index are given in Table 5.2-5. The largest index for either U.S. or world resources is

TABLE 5.2-5. RESOURCES INDEX

- Recycling losses for Be and solid breeders : 3% spherepac, 1% all other forms
- Burnup assumed same for all breeders
- Blanket lifetime = 3 FPY (full power years)

	I_i	
Requirements, %	Of U.S. Resources	Of World Resources
< 1%	1.0	0.5
1 → 10%	0.5	0.3
10 → 50%	0.2	0
> 50%	0	0

selected for each concept. If a blanket uses more than one limited material (n > 1), then the overall index will be given by

$$I_i = \prod_{j=1}^{n} I_{ij} \qquad [5.2-8]$$

where I_{ij} is the index score for the jth material.

5.2.5 Accommodation of Power Variations

Fusion power plants, particularly those based on tokamaks and tandem mirrors, are generally viewed as being base-load plants. However, there is still the need to operate at reduced power for extended periods of time (ranging from hours to months) to accommodate standard utility practices for startup procedures, or perhaps to operate the plant with a failed component until a convenient or scheduled replacement time. Thus we are interested in the ability of a given reference blanket concept, designed for a nominal wall loading of 5 MW/m^2, to operate over a range of power loading from as low as 5% to as much as 120%. The higher percentage is to accommodate spatial variations (e.g., poloidal variation in tokamaks) at nominal conditions as well as possible modest power surges above the nominal operating point. The coolant temperatures were assumed to remain constant while the flow rate was changed to accommodate power changes. It was assumed that the surface heat flux remains proportional to the neutron wall load. The index values are given in Table 5.2-6.

Another important consideration is whether it will be necessary to recover tritium from the blanket during low power operation; this may be a particularly important question for solid breeder blankets. If the reactor were to operate for extended periods of time (days to weeks) and tritium was not released from the blanket (due, for example, to lower temperatures in the solid breeder material), then low power operation would entail serious consequences such as the need for substantial storage of reserve tritium. Therefore, blanket designs were checked to assure adequate release of tritium during extended operating periods at reduced power levels.

TABLE 5.2-6. POWER VARIATION INDEX

- Reference design assumed.
- No design or configuration changes permitted; operating changes only.

Variation Accommodated	I_i
5% → 120%	1.0
25% → 120%	0.8
50% → 120%	0.5
90% → 110%	0.2
None	0

5.2.6 Ability to Increase Neutron Power Loading

The ability to design a particular blanket concept to operate at wall loadings above the nominal value of 5 MW/m^2 is a desirable feature. This item is included to assess the suitability of the blanket to accommodate improved plasma confinement concepts which may operate at higher power densities. In assessing the possibility of increasing the wall loading for a particular blanket concept, mechanical design modifications from the reference design at

5 MW/m^2 would be permitted. Examples would include changing the coolant tube spacing, first wall thickness, and total blanket/shield thickness. It is assumed that the surface heat flux is proportional to the total wall loading. The value of the index for this item is given by

$$I_i = \frac{P_{NW} - 5}{5} \qquad \begin{cases} I_i = 0 \text{ @ } P_{NW} \leq 5 \text{ MW/m}^2 \\ I_i = 1 \text{ @ } P_{NW} \geq 10 \text{ MW/m}^2 \end{cases} \qquad [5.2\text{-}9]$$

Increased blanket lifetimes would of course be desirable for operation at higher wall loads. However, at present all blanket concepts appear to be life-limited by fluence, specifically radiation damage to the first wall, which would not change for higher wall load values. In addition, there may be economic advantages to operation at higher wall loads even if blanket life is shortened and changeouts are more frequent, e.g., capital cost savings due to more compact reactor sizes. Thus no penalties for shorter calendar lifetimes were assessed for this index unless the life as measured in $MW\text{-}yr/m^2$ decreased.

5.2.7 Higher Surface Heat Flux and Higher Erosion Rates

For the tokamaks, the surface heat flux and erosion of the first wall might be considerably higher than those adopted in the design guidelines. For the tandem mirrors, such a situation is considered possible but not likely. The lack of capability to accommodate increased values for those parameters can thus be considered a measure of the relative risk for each concept of not being able to successfully operate under such worsened conditions.

The equation to determine the index for measuring the capability for accommodating higher surface heat fluxes or higher erosion rates at the reference neutron wall load value of 5 MW/m^2 is

Tokamak

$$I_i = \frac{q - 1.0}{2.0} + \frac{t_e - 2}{16} \qquad \begin{cases} I_i = 0 \text{ at } q \leq 0.5 \text{ and } t_e \leq 2 \\ I_i = 1 \text{ at } q \geq 1.25 \text{ and } t_e \geq 10 \\ (q \sim MW/m^2,\ t_e \sim mm) \end{cases} \qquad [5.2\text{-}10]$$

$$I_i = \frac{q - 0.05}{.10} + \frac{t_e - 0.2}{1.6} \quad \begin{cases} I_i = 0 \text{ at } q \leq 0.05 \text{ and } t_e \leq 0.2 \\ I_i = 1 \text{ at } q \geq 0.10 \text{ and } t_e \geq 1.0 \end{cases}$$

In determining the achievable values for q and t_e, the reference designs and operating parameters were assumed to be unchanged, except for adding more erosion thickness allowance to the first wall surface. In general, the most important constraint on improved performance was the maximum temperature limit for first wall structural material.

5.2.8 Startup/Shutdown Operations

There are two basic types of startup/shutdown operations: "hot" and "cold" sequences. Hot startup/shutdown operations refer to the required sequences resulting from those conditions in which the blanket remains near nominal operating temperature and the plant is in a stand-by mode. There are apparently no significant differences among top-ranked blankets in this case, except for the subsystems that are required to keep the blanket in "hot" conditions. Therefore, hot startup/shutdown is not explictly considered.

On the other hand, "cold" startup/shutdown operations, which results from longer shutdowns when the plant is not producing power, do depend on blanket characteristics. It is desirable that the time to bring the blanket from a cold condition to operating conditions be minimized and that there be no or at least simple additional components of the blanket system or the power conversion system to accomplish such operations. The index chosen for this item is given by Table 5.2-7. Sixty percent of the total points available relate to "cold" startup/shutdown times. The remainder relate to the need for and complexity of subsystems needed to accomplish startup/shutdown, e.g., heater systems for liquid metal piping.

TABLE 5.2-7. STARTUP/SHUTDOWN INDEX

Time Required for Cold Startup	I_1
< 1 day	0.6
1 → 3 days	0.3
> 3 days	0
Additional Blanket and Auxiliary Subsystems Required	
None	0.4
Simple	0.2
Complex	0

REFERENCES - SECTION 5.2

1. M. A. Abdou, "Tritium Breeding in Fusion Reactors," report ANL/FPP/TM-165, Argonne National Laboratory (October 1982).

2. M. A. Abdou et al., "Blanket Comparison and Selection Study - Interim Report," report ANL/FPP-83-1 (Vols. I and II), Argonne National Laboratory (October 1983).

3. "Mirror Advanced Reactor Study Final Design Report," report UCRL-53480, Lawrence Livermore National Laboratory (July 1984).

4. C. C. Baker et al., "STARFIRE - A Commercial Tokamak Fusion Power Plant Study," report ANL/FPP/80-1, Argonne National Laboratory (September 1980).

5.3 Economics

Economics is one of the four important evaluation tools employed in this study. The economic evaluation includes not only the capital cost of the blanket components, but also includes all blanket-induced economic factors on the overall cost of electricity (COE). These influencing factors include power production, geometry constraints on other systems, need for special ancillary systems and recirculating power requirements. Thse influencing factors are utilized in the study to provide a uniform basis of comparison and selection. The single, final economic criterion is the Cost of Electricity.

The next few sections will explain the methodology employed in the economic evaluation, the economic groundrules, and the specific design data input into the systems code. The results of the analysis are discussed in Section 3.2, Economic Results.

5.3.1 Economic Evaluation Methodology

The cost of electricity was adopted as the sole criterion for the economic evaluation for the blanket concepts considered. When evaluating competing blankets concepts, comparison of only the direct capital costs of the blanket components is an insufficient evaluation technique and may lead to erroneous conclusions. A direct cost comparison will favor the lower cost options and slight the more expensive but higher performance blankets. Use of the cost of electricity criterion integrates the weighted effects of capital costs, operating costs, and overall system performance. Assessment of these factors in a consistent manner requires the blanket be evaluated in the context of a rather detailed and comprehensive conceptual reactor design. Thus, the STARFIRE[1] tokamak reactor and the MARS[2] tandem mirror reactor were adopted as the technical and economic baselines for this blanket comparison. Blanket concepts were evaluated in both reactor configurations, where possible, in order to assess the influence of the inherent magnetic and configurational differences on the blanket selection. Resultant COE values will be normalized within each reactor type to focus comparison on the blanket concepts rather than on the differences between the reactors. This was necessary because the two referenced conceptual designs have not been normalized and were developed with differing groundrules. The equation used for the Cost of Electricity is shown below:

COE =

$$\frac{(DC+SPR+CTGY+ID+INT+ESCL)FCR+(O\&M+SCR+FUEL)(1+ESC\ RATE)^{YRS}}{(Thermal\ Power \times Gross\ Effcny - Recirc.\ Power)(Availability)(hrs/y)}$$

where	COE	= Cost of Electricity
	DC	= Direct Capital Costs
	SPR	= Spare Parts Allowance (2 to 4%, depending on system)
	CTGY	= Contingency Allowance (15% of Direct Costs)
	ID	= Indirect Costs
	INT	= Interest During Construction (Based on 10% cost of money over construction period)
	ESCL	= Escalation During Construction (Based on 5% escalation over construction period)
	FCR	= Fixed Charge Rate (Nominally 15%)
	O&M	= Operations and Maintenance Costs
	SCR	= Scheduled Component Replacement Cost
	FUEL	= Annual Fuel Cost
	ESC RATE	= Annual Escalation Rate (5% per year)
	YRS	= Construction Period

The cost of electricity is the total bus bar energy for the first year of operation[3]. The total capital investment is equally divided and charged to the annual operating periods through the use of a fixed change rate. Annual operating costs are also included with appropriate escalation from the year of the estimate (start of construction in 1983) to the initial operational date (1989).

Table 5.3-1 lists the major factors considered in this economic evaluation study. In this study, the direct costs of the blanket (including the first wall, multiplier, breeder, reflector, and plenum regions), shield, limiter, and magnets were calculated based upon both the materials chosen and the imposed geometry constraints. The blanket design also affected the

TABLE 5.3-1 FACTORS WHICH INFLUENCE ECONOMICS

Direct Costs	Annual Costs
Blanket Shield Limiter Magnets Heat Transport Steam Generator Turbine and Electric Plant Building Over Pressure Fuel Handling and Storage	Scheduled Component Replacement: - Blanket Replacement Cost - Blanket Replacement Frequency (Life)
Power Output	**Availability**
Blanket Energy Multiplication Gross Efficiency Coolant Pumping Power	(Fixed)
Invariant Support Data[a]	
Other Reactor & Plant Direct Costs Spares Allowances Contingency Allowances Indirect Cost Allowances Construction Time Interest and Escalation Rates Fixed Charge Rate	Basic Reactor Geometry Direct Convertor Efficiency Other Recirculating Power Requirements Scheduled and Unscheduled Outages Unit Material Costs Operations and Maintenance Costs

[a] Data obtained from STARFIRE and MARS conceptual design studies.

ancillary systems of heat transport, intermediate coolant loops, buildings (special flooring or overpressure requirements), fuel handling and storage, steam generators and turbine plant equipment, for which the direct cost influences were calculated. The additional recirculating power requirements for some concepts caused the cost of the electric plant equipment to increase. Unit material costs were applied to estimated material and fabrication requirements plus unit costs for discrete components (see following Section 5.3.2 for details). All costs are estimated on the basis of, or scaled to, 1983 dollars. All other direct costs were adopted from the STARFIRE or MARS economic analyses. Allowances for spare parts, contingency and indirect costs are applied to the direct costs. Interest and escalation costs incurred during construction are levied to transform the estimated costs at the start of construction to the initial year of operation (1989). A fixed charge rate of 15% was applied to amortize the total capital debt and to provide for insurance, taxes and other annual capital-related costs.

The annual costs influenced by the blanket concepts are related only to the scheduled component replacement costs and frequency of replacement.

The power output of the plant has a significant influence on the economics of the plant. The blanket concept design approach, material choices and performance capabilities have a profound influence on the plant power output. The blanket energy multiplication times the neutron energy specified the total thermal power available. The temperature of the coolant media (both primary and secondary, if required) determined the gross power conversion efficiency. The choice of the coolant media and the operating pressure established the pumping power requirements. A portion of this pumping power was recovered in terms of useful heat, thus lessening this detriment.

Another blanket performance parameter which influenced the economics of the plant is the tritium breeding ratio (TBR). This ratio varied among blankets designs. The envisioned designs produce, over the long term, marginally adequate to excess amounts of tritium to overcome losses, decay, and replenishment of the original inventory. This effect has been assessed in the Engineering Feasibility Evaluation. However, assessment in the Economic Evaluation is more difficult and is subject to more assumptions and conjecture. The early power plants would desire a higher tritium breeding ratio to provide startup inventories for other D-T plants. Even the TBR for the early plants would be tailored to the pace of the introduction of the new plants. After the pace of introduction slows and the fusion economy matures, the TBR will likely be reduced as the support ratio is reduced. The current, pre-fusion cost of tritium is assumed to be in the neighborhood of 10 million dollars per kilogram. As fusion is introduced, it is expected that price of tritium will decrease. With a required tritium startup inventory in the few to tens of kilograms, this is still a significant capital cost. However a quantity of tritium equal or greater than the initial inventory will be produced by the blanket for resale to other plants. If supply and demand are rather equally matched within the fusion power community, the costs will approximately balance. This is the scenario chosen for the BCSS Economic Evaluation.

There are other scenarios involving the TBR which may occur. If many plants are constructed with high tritium breeding ratios, an excess of tritium will be generated, with the price of tritium falling to a lower value perhaps established by an indifference value equated to production of a fissile fuel. Even at a lower market price, the sale of tritium can generate significant income for the plants producing a sizeable excess of tritium. This argument may have some validity if a stable market is established. On the other hand, if no stable market, other than fusion, is forecast, the price will likely continue to fall such that the blankets will be redesigned to produce less excess tritium and more power. This effect could be evaluated with the present blanket designs. A maximum TBR would be established, accounting for losses, decay, uncertainties, and replenishment of the original inventory. The capability of any blanket to exceed this maximum value would be equated to its ability to produce more thermal power. This increased power production would then be included in the economic assessment. The problems arising from this evaluation technique in the BCSS is that this was not one of the original design guidelines and significant design changes to the blanket may be required to accomplishment this effect. Thus, this evaluation technique for TBR was not adopted and TBR is evaluated only in the context of Engineering Feasibility.

The final factor to consider is the plant availability. Availability is the percent of time the power plant is available for service, whether operated or not. The design life of the blanket determines the annual cost of replacement, but it also is a major factor in the scheduled downtime of the reactor. Most conceptual design and maintenance studies attempt to have blanket replacement occur simultaneously with other balance-of-plant scheduled maintenance, thus minimizing the blanket's maintenance-related influence. The diversity of the blanket design approaches would significantly influence the blanket maintenance which may result in altered availabilities. However it was deemed too difficult to quantify the availability influence associated with each blanket concept. The BCSS evaluation study adopted a fixed availability of 80% largely based upon the early availability assessments in the MARS study.

A computer code was developed for BCSS to analyze these factors discussed above. Design specifications and performance data were input along with the unit cost factors. These data were developed by the BCSS Concept Design Team or by the responsible Task Group (eg, Power Conversion). Where the reactor type, (tokamak or tandem mirror), necessitated a different system or a significant modification, the code was defined uniquely for each reactor type.

5.3.2 Economic Groundrules

Performance and economic groundrules were established to provide a consistent economic blanket comparison within the framework of the conceptual designs of STARFIRE and MARS. This provided realistic geometry constraints, auxiliary power requirements and economic burdens. Some modifications were required to achieve the $5MW/m^2$ neutron wall loading as a blanket design basis. The tandem mirror design adopted the plasma geometry of MARS (0.60m wall radius and an effective central cell length of 130.8m) with a minor variation in beta to achieve 5.0 MW/m^2 from the MARS $4.2MW/m^2$. The BCSS TMR will produce 3083MW of fusion power for all blanket concepts. All the neutron power is assumed to be deposited in the first wall, blanket or shield. The power recovered by the first wall and blanket is defined by the neutron power times the Blanket Energy Multiplication Factor for each blanket. The energy lost in the shield is the difference between the Total Energy Multiplication Factor and the Blanket Energy Multiplication Factor. Sixty-seven percent of the alpha power plus plasma heating power is recovered by the direct convertor at a 64.3% efficiency. The remaining alpha plus plasma heating power from the direct convertor is recovered by the halo scraper. This thermal energy plus the reject heat from the direct convertor is converted to electricity at a conversion efficiency of 34.3% by a separate moderate-temperature thermal conversion system. In MARS, the blanket design had a hot reflector which sent thermal energy to the moderate-temperature thermal convertor, which is not incorporated in any BCSS design. Instead all heat generated in the reflector zones, if incorporated, are sent to the primary loop. The heating power input to the plasma is assumed to counterbalance other miscellaneous losses. As the thicknesses of the blanket and shield vary, so do the radii and cost of the tandem mirror central cell magnets.

For the tokamak, the scaling of the reactor to the desired wall loading is a more difficult process. The approach adopted was to utilize the STARFIRE baseline geometry but increase the fusion power from 3510MW to 4875MW (i.e., wall load of 3.6MW/m^2 increased to 5.0MW/m^2). This was accomplished by allowing the average toroidal beta to increase from 0.067 to 0.097 and assuming a maximum field of 10 tesla as opposed to 11.1 for the STARFIRE baseline. Given a set of thicknesses for a particular blanket and shield combination, a new magnet geometry is calculated. This geometry coupled with a maximum field of 10T at the inner leg of the magnet yields a new field at the plasma center (B_o). The new tokamak fusion power is related to the plasma center by the equation.

$$P_{f(new)} = P_{f(ref)} \times \left(\frac{B_{o(new)}}{B_{o(ref)}}\right)^4 \qquad [5.3\text{-}1]$$

This approach holds the tokamak first wall geometry and maximum magnetic field constant while varying the wall loading and the innermost surface of the inner magnet leg. This approach does not require a recalculation of the plasma physics for each blanket concept. Since the approach chosen varies the fusion power (and wall loading), the economic model does not evaluate blankets at exactly the baseline design wall loading. In the economic model, the replacement lifetimes are adjusted to account for the reduced wall loading.

As in the case for the TMR, the power recovered in the first wall and blanket is the neutron power times the Blanket Energy Multiplication Factor. The alpha and heating power are split between the limiter and the wall. The energy recovered in the limiter is used for feedwater heating. The thermal conversion efficiency values quoted consider the utilization of the feedwater heating power and the thermal power quoted is a summation of the limiter, blanket and wall powers.

The specific geometry models were adopted from STARFIRE and MARS for the plasma, scrapeoff, blanket, shield, gaps dewars and magnets are shown in Figures 5.3-1 and 5.3-2. The physical length of the MARS central cell region was adopted at 135.3 m.

The amount of materials in the blanket was calculated by using the designer-specified zone thicknesses, the materials in the zone, and their respective volume fraction and percent of theoretical density. After the designer determined the necessary thickness of the blanket, the shielding composition and thickness for each blanket concept was calculated with a 1-D neutronics analysis. Separate shielding thicknesses were determined for the inboard and outboard tokamak regions and areas between TMR coils and the area under TMR coils for each blanket concept. The economic analysis code is capable of evaluating these shielding variations as well as the blanket options. Each zone (shell) volume and material mass is calculated and the material units cost applied. Table 5.3-2 describes the structural material applications and the material unit costs with applicable references. Table 5.3-3 is a summary of the lithium material unit costs and applicable references. Table 5.3-4 illustrates that beryllium costs have a high degree of uncertainty. There is only one major supplier which has their own mines, and are reluctant to be competitive. A cost sensitivity study in Section 2.2 illustrates the influence of this cost item. Table 5.3-5 is an overall summary of the blanket concept material applications and the respective unit costs. Most of the unit costs in the prior tables were escalated from a fusion handbook of standard unit costs and scaling laws [4] and from data used in recent fusion conceptual designs. [1,2,5-8]. The magnet costs were scaled from the STARFIRE[1] and MARS[2] economic analyses. A more detailed discussion of magnet cost scaling is given in Section 5.3.4.

The replacement cost for the blanket was also considered. Structural materials will require replacement due to radiation damage. The structural materials in the blanket will be replaced at the same cost as the original blanket costs, except for those designs where radioactive components must be installed in the new blanket or the material can be reconstituted or refabricated and returned to service. A cost penalty of $5/kg is added to the

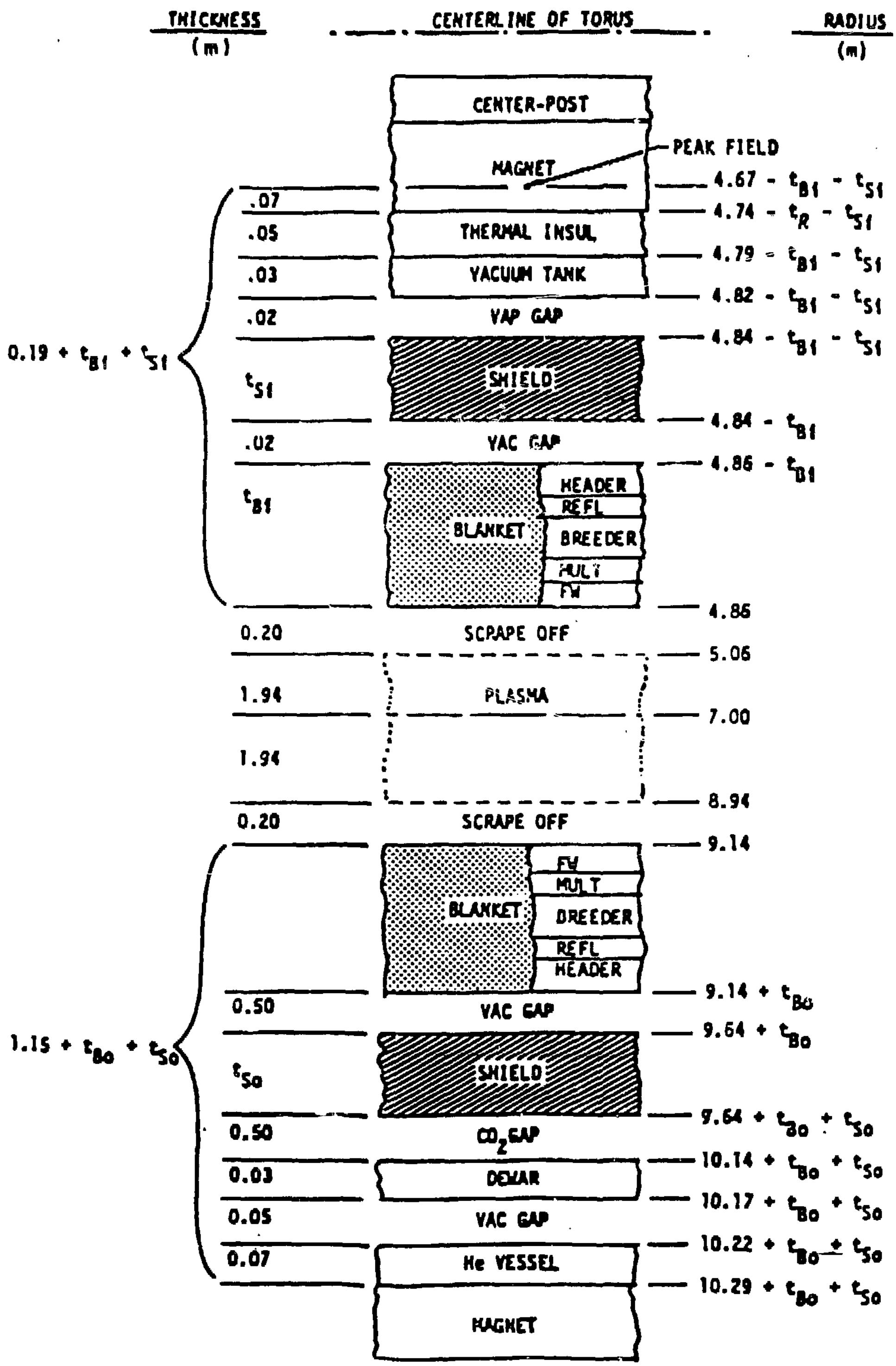


Figure 5.3-1 **TOKAMAK BLANKET AND SHIELD GEOMETRY ASSUMED IN ECONOMIC ANALYSES (STARFIRE IS REFERENCE)**

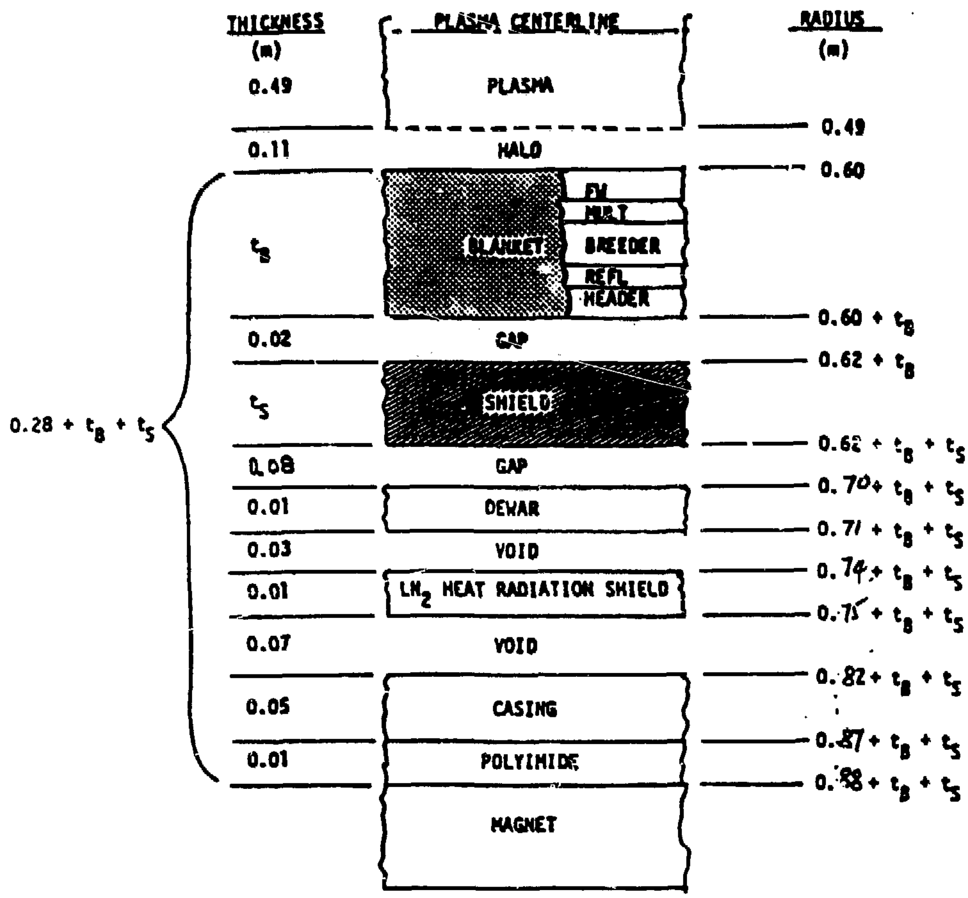


Figure 5.3-2 **TANDEM MIRROR BLANKET AND SHIELD GEOMETRY ASSUMED IN ANALYSES (MARS IS REFERENCE)**

Table 5.3-2. DESCRIPTION OF STRUCTURAL MATERIAL APPLICATIONS AND INSTALLED UNIT COSTS

($/kg)

Designation	A	B(TMR)	C(TMR)	D(TMR)	D(TOK)	E	F	G	H	I
Breeder/Clnt/ Structure	$LiAlO_2$/NS/ HT-9	Li/Li/ HT-9	LiPb/LiPb/ V-15Cr-5Ti	Li/Li/ V-15Cr-5Ti	Li/Li/ V-15Cr-5Ti	Li_2O/He/ HT-9	$LiAlO_2$/He/ HT-9	Li/He/ HT-9	Flibe/He/ HT-9	$LiAlO_2$/H_2O/ HT-9
First Wall	Grooved Surf. (TOK) Smooth Surf. (TMR) Rect. Clnt. Passages (TOK-45, TMR-39)				Grooved Surf. (TOK) (V-250)	Grooved Surface (TOK) Smooth Surface (TMR) Internal Fins and Thermal Barrier(TOK) No Fins (Outer Walls), Plus Thermal Barrier (TMR) (TOK-50, TMR-45)				Grooved Surf. (TOK) Smooth Surf. (TMR) Rect. Clnt. Passages (TOK-45, TMR-39)
Blanket Structure and Coolant System	Single-Wall Coolant Tubes with Truss Stiffeners, Rect. Psags in Side Walls (39)	10 cm. Tubes (HT-9 -29, V -250) Rectangular Coolant Passages Box-Beam Structure (HT-9 -29, V -250)			Angled Passages (V- 250)	Fin-Stiffened Outer Side Wall; Crushable Panels as Th. Barrier; Crushable Panels in Cladded Fuel Plates; Purge Connections: Purge-to-Plate Crush Washers (50)	Double-Wall Thermal Barrier All Along Blanket Sides; Cladded Fuel Plates	Li in Tubes Cnctd to End Plenums, HT-9 Plates in Rfltr Zone	Flibe in Tubes Surrnd by Be Balls, Flibe in Tubes within SiC Refl.	Small U-Bend Double-Wall Coolant Tubes, Rectangular Passages in Side Walls (39)
Coolant Plenum	Annular Plenum on Each End (29)					Strong Back Beam Plate Coupled with Large Strong Back End Plates. Complex System of Baffles for Flow Distribution Along Length of Module (35)				Rectangular Passages (29)

TABLE 5.3-3
LITHIUM MATERIALS COST SUMMARY

Material	Enrichment (%)	Unit Cost ($/kg)		Ref (83$)
Lithium	Natural	40*		40[a],22[b], ~40[c]
	90	1200		1300[a],1200[c]
		Clad	Spherepac	
Li_2O	Natural	40*		43[a],40[d]
	90	600		990[a]
$LiAlO_2$	Natural	-		53[a]
	50	86	76*	
	60	100*		98[e]
	90	200	190*	211[a]
Flibe	Natural	37*		37[f]
LiPb	Natural	4		4[g]
	30	6.25*		6.25[c]
	90	12		

* These materials are used in the BCSS Study

a. S. Schulte, et. al., "Fusion Reactor Design Studies-Standard Unit Costs and Cost Scaling Rules", PNL-2987, Pacific Northwest Laboratory, Richland WA, Sept 1980.

b. J. D. Lee, et.al., "Feasibility Study of a Fission-Suppressed Tandem Mirror Hybrid Reactor", LLNL UCID-19327, 1982

c. B. Badger, "UTWOR", UWFDM-550, 1982

d. R. Moir, et,al., "Tandem Mirror Hybrid Reactor Design Study Final Report", LLNL UCID-18808, Sept 1980

e. C. C. Baker, et.al., "STARFIRE-A Commercial Tokamak Fusion Power Plant Study", ANL/FPP-80-1, Sept 1980

f. Derived from data in Reference (b).

g. B. Badger, "WITAMIR-I, A Tandem Mirror Reactor Study", UWFDM-400, Sept 1980

TABLE 5.3-4

BERYLLIUM COST ESTIMATES CONTAIN UNCERTAINTY

($/kg)

	BRUSH WELLMAN PRICE LIST	NUCLEAR METALS QUOTE	SPEED RING QUOTE	TRW ESTIMATE	BCSS ESTIMATE
Raw Material				350	
Scrap Metal	469				
Powder, Std. Qual	407				
, Hi Purity	512				
Spherepac					440
Rods, ½"d., mach	5,420				
, 1"d., mach	2,389				
, ½"d., drawn		2,333			
, 2 cm.				440	440
Balls, 1 cm. d.			7,226	440-626 (Dependant upon losses & inspection)	440

TABLE 5.3-5

SUMMARY OF BLANKET MATERIAL INSTALLED UNIT COSTS

DESIGNATION	A	B	C	D	E	F	G	H	I
Breeder/Coolant Structure	$LiAlO_2$/NS/ HT-9	Li/Li/ HT-9	LiPb/LiPb/ V-15Cr-5Ti	Li/Li/ V-15Cr-5Ti	Li_2O/He/ HT-9	$LiAlO_2$/He/ HT-9	Li/He/ HT-9	Flibe/He HT-9	$LiAlO_2$/H_2O HT-9
First Wall, Tok , TM	HT-9VC $45 HT-9VC $39			V-15Cr-5Ti$250	HT-9VC $50 HT-9VC $45	HT-9VC $50 HT-9VC $45	HT-9VC $50 HT-9VC $45	HT-9VC $50 HT-9VC $45	HT-9VC $45 HT-9C $39
Bkt Structure	HT-9C $39	HT-9M $29 (Tubes)	V-15-Cr-5Ti$250	V-15Cr-5Ti$250	HT-9VC $50	HT-9VC $50	HT-9VC $50	HT-9VC $50	HT-9C $39
Reflector		HT-9M $29 Fe-1422 $12	Fe-1422 $12	Fe-1422 $12			HT-9M $29 HT-9B $12	SiC $18	
Plenum	HT-9M $29				HT-9M $35	HT-9M $35	HT-9M $35	HT-9M $35	HT-9M $29
Multiplier -Form	Be $440 Spherepac		LiPb $6.25			Be $440 2-cm Rods		Be $440 1-cm balls	Be $440 Spherepac
Breeder -Enrichment -Form	$LiAlO_2$ $76 50% Spherepac	Li $40 Nat	LiPb $6.25 30%	Li $40 Nat	Li_2O $40 Nat Clad Plates	$LiAlO_2$ $100 60% Clad Plates	Li $40 Nat	Flibe $37 Nat	$LiAlO_2$ $190 90% Spherepac
Coolant	Nitrate Salt $1.5	Li $40	LiPb $6.25	Li $40	He	He	He	He	H_2O

NOTE: Differences in fabrication complexity of HT-9 are denoted by the following letter designations. Within gross categories, a gradation of costs are possible.

VC = Very Complex
C = Complex
M = Moderate
B = Bulk

The cost of beryllium has risen dramatically in the past few years. The costs stated above assume some price reduction from the current prices based upon an increased demand, quantity reductions, and improved fabrication techniques.

installed in the new blanket or the material can be reconstituted or refabricated and returned to service. A cost penalty of $5/kg is added to the original structural material unit cost. The beryllium or solid breeder Spherepac balls designs pose a minimal cost penalty due to their ease of removal and installation. The liquid breeders and/or coolants can be continually reused with sufficient makeup of more highly enriched ^{6}Li. An assessment of $1M is assumed to be sufficient for replenishment. The SiC reflector in the Flibe design is assumed to not degrade and require replacement, but 40% of the original costs is allocated to account for breakage and remote handling in the remanufacturing process. The beryllium multipliers are conservatively assumed to only require remote handling in the remanufacturing process and no new material is required. The beryllium remanufacturing process is assumed to cost:

Spherepac	$120/kg	Rods	$100/kg
1-cm Balls	$100/kg		

The clad $LiAlO_2$ (60% enriched) plates will be reconstituted and remanufactured using remote handling equipment for a cost of $45/kg. The Li_2O will be cheaper to purchase new breader components at $40/kg than remanufacture the old units. The Spherepac $LiAlO_2$ will require enough new material to counteract the lithium burnup. A Spherepac cost of $30/kg will include the cost of remote handling refabrication and the new material required.

A major departure from the design baseline of the STARFIRE was not using the steady-state current drive system. It was felt that the design basis would be more credible to the fusion community if tokamak was depicted to be a long-pulse burn (10,000 sec burn with a 40 sec downtime). This is in concert with a recent published study (9) on the comparison of steady-state and pulsed tokamak reactors. This study will form the basis for the differences between the STARFIRE reactor and the tokamak reactor used for the BCSS study. Although the costs of the effected systems are considerably higher, this change does not play a major role in the comparison of the blanket design comparison or selection. The major reactor systems impacted are the magnets, rf heating, power supplies, and thermal storage. This long pulse can be achieved by driving a current with a transformer and then maintain the toriodal current with a non-inductive current driver while the transformer is being reset (at low plasma density and temperature). One performance effect is the reduction of the power consumed by the steady-state current drive system (~90 MW) as compared to a small nominal value of power required to charge the OH coils when averaged over the burn cycle. Counteracting this reduction in power is an additional pumping power of 12MW for the thermal storage system for the helium-cooled blanket design. The proposed thermal storage systems for the other primary coolants consume minimal amounts of pumping power. The cost of the magnet system is significantly increased. Per D. Ehst's work,[9] the cost of the TF coils for pulsed operation increase by 37% to keep the fatigue stress to acceptable values. The EF and CF coils are unchanged but the OH coils with hybrid-current drive is increased by approximately $90M. The power supplies are unchanged except for the OH and RF systems resulting in an $110 increase. The above delta costs are applied equally to all tokamak blanket concepts and do not influence the comparison or selection process. The tokamak thermal-energy storage costs were developed by the BCSS Power Conversion Systems and IHX Task Group. The requirement is to store the reactor thermal power for the reset period of 40 seconds. The thermal storage costs are:

Primary Coolant	Storage Technique	System Cost
H_2O	Storage of primary coolant in insulated tanks	$5M
Li	Storage in primary coolant in insulated tanks	$10M
He	Steel balls heated by He (requires 12MW pumping power)	$25M
NS	Storage of primary coolant in insulated tanks	$6M
LiPb	No tokamak design	N/A

Another system affected by the choice of the blanket is the Fuel Handling and Storage System. This system also includes the Plasma Exhaust and Atmospheric Tritium Cleanup subsystems. Table 5.3-6[10] presents the design parameters affecting the Fuel Handling and Storage System. The costing of this reactor system was handled separately from the remainder of the systems. A full explanation of the Fuel Handling and Storage System cost methodology, results and conclusions are found in Subsection 5.3.3. An overall summary of the Fuel Handling and Storage System costs is shown in Table 5.3-7.

As shown in Table 5.3-6, the limiter coolant is either water or lithium. The water-cooled limiter design uses a copper structural material, beryllium coating and tantalum nose tip. The cost of the bottom-mounted limiter, representative of the FED design is approximately $4.4M. Blanket and shielding, which may be housed in the limiter module, are included in the blanket and shielding cost accounts. The lithium-cooled limiter, comprised of V-15Cr-5Ti structure, beryllium coating and tantalum tip is costed at $14.7M.

The choice of the blanket coolant and operating temperature has a significant impact on the system design and costs. Table 5.3-8 is a summary of the coolants used in the BCSS study. Heat Transport unit costs for the major system elements are shown in Table 5.3-9. Since the lithium design is thought to require the use of a sodium intermediate loop, these extra costs are included. Since the placement of the intermediate heat exchanger at some midpoint between the blanket and the steam generator reduces the respective piping lengths, the primary and secondary piping costs for the lithium system are each reduced to 60% of the value associated with only a primary loop system. Ferritic steel piping is assumed for the primary loop. In the case of the blanket with vanadium structure, the piping is converted to steel near the exit from the blanket and shield. Cost relationships for other system elements, such as pumps, dump tanks, pressurizers, and coolant cleanup, were developed and used, but not shown here for brevity.

Another impact of the choice of coolant is the consideration of overpressure effects of the loss of coolant on the reactor building. The pressurized water coolant is estimated to be capable of creating an overpressure of 80 to 100kPa whereas a loss of coolant accident with the other coolants will result in an overpressure of less than 10kPa. This is estimated to result in a

TABLE 5.3-6

SPECIFICATIONS OF SEVEN BLANKET SYSTEMS CONSIDERED FOR COST COMPARISON

CONCEPT	A	B, D	C	E, F	G	H	I
Blanket Material	$LiAlO_2$	Li	LiPb	Li_2O/TC[a]	Li	Flibe	$LiAlO_2$
Limiter or Halo Scraper Fluid	H_2O	Li	H_2O	H_2O	Li	H_2O	H_2O
Blanket Fluid	Nitrate Salt	Li	LiPb	He	He	He	H_2O
Tritium Extraction Technique - Blanket	He Purge	Yttrium Beds	Counter Current He	He Purge	Yttrium Beds	Electrolysis	He Purge

[a] Ternary Oxide

TABLE 5.3-7. FUEL HANDLING AND STORAGE SYSTEM COSTS

(M$)

TYPE OF SUBSYSTEM	A	B	C	D	E	F	G	H	I
Bkt & Clnt Processing									
- Tandem Mirror	135	130	146	140	128	140	129	168	95
- Tokamak	135	-	-	91	135	151	85	186	97
Exhaust Processing									
- Tandem Mirror	31	31	31	31	31	31	31	31	31
- Tokamak	40	-	-	40	40	40	40	40	40
Atmos. Trit. Processing	36	36	36	36	36	36	36	36	36
Total									
- Tandem Mirror	202	197	213	207	195	207	196	235	162
- Tokamak	211	-	-	167	211	227	161	262	173

TABLE 5.3-8

PRIMARY COOLANT DATA

Coolant	Max T, ^0C	Ave. Density	Press., MPa
NS	405-450	1.85	0.4
He	500-540	.0033	5.2
LiPb	530	9.4	1.1
Li	500-550	0.5	0.6-3.1
Na	480-525	0.8	0.4
H_2O	320	0.7	15.2

TABLE 5.3-9

MAJOR HEAT TRANSPORT SYSTEM UNIT COSTS

Primary Coolant	Intermediate Coolant	IHX	SG	Piping Cost for 3800 MW [(a)] Intermediate	Primary
NS	-	-	\$11/kW	-	\$39.6M
Li	Na	\$17/kW	\$19/kW	\$70M*60%	\$87.6M*60%
LiPb	-	-	\$20/kW	-	\$109.3M
He	-	-	\$28/kW	-	\$72.6M
H_2O	-	-	\$11/kW	-	\$39.6M

(a) Piping costs for TMR are assumed to be 30% higher due to the longer pipe lengths. Piping costs are scaled up and down from the 3800 MW value according to (power ratio)$^{0.85}$. Also, the limiter power and pump power addition have been subtracted out for these costs.

$12.25M delta cost, attributable to less materials required in thinner building walls. The use of a liquid metal or salt (Li, LiPb, Flibe, or Nitrate Salt) coolant necessitates the use of a steel floor to minimize the safety hazard. This floor is estimated to cost an additional $5.75M.

The remainder of the STARFIRE and MARS Reactor and Balance of Plant were adopted for the BCSS to form a performance and economic basis. Specific items included were buildings, ancillary reactor equipment, turbine plant equipment, miscellaneous plant equipment, maintenance equipment, spare part and contingency allowances, indirect cost assumptions and annual operations and maintenance costs. If the blanket concepts altered cost-influencing parameters of these systems (eg., recirculating power), the appropriate costs were modified.

5.3.3 Costing Methodology for the Blanket Tritium Processing Systems

The methodology used to estimate tritium system costs for the proposed blanket concepts is discussed in this section. The major goal was that the costing be self-consistent for all designs. To achieve this, it was assumed that each blanket recovery process could be separated into component subsystems, each of which was costed. The total cost was the summation of the subsystem costs. To provide a total tritium processing cost, the blanket processing costs were then added to the fuel processing costs and to the atmosphere tritium recovery costs. Tokamak blanket designs and mirror blanket designs were separately assessed.

5.3.3.1 Basic Assumptions

Information on a given blanket system was provided by the responsible design team. Summaries of the data provided are found in Tables 5.3-10 and 5.3-11 for the tokamak and tandem mirror designs respectively. Although the blanket tritium processing rates for the tokamak and mirror blanket designs vary over a large range 600 to 1100 g/d, a base design handling 600-800 g/d can be used for both reactor types. When larger processing rates are required, the cost is incrementally increased.

The fuel processing costs for the ~4000 MW tokamak and the mirror designs were assessed as follows. There is both a fixed cost (~$20M) and a variable cost ($10-$20M) associated with these systems. The variable cost depends on the processing rate and is directly proportional to the fusion power and inversely proportional to the fractional burn. Since the mirror fusion power is ~80% of the tokamak's and the mirror's fractional burn is ~140% of the tokamak's, the respective capital costs for the fuel processing systems are $31 M for mirrors and $40 M for tokamaks.

The atmospheric tritium recovery system provided to handle tritium releases both during maintenance and accident conditions is sized so that it can clean up releases with a high probability ($>10^{-2}$) in 3 days and those with lower probability $<10^{-3}$ in 5 days. For a building with an internal volume of 2×10^5 m^3, the capital cost of systems capable of providing cleanup in 3 or 5 days is shown in Table 5.3-12. A system costing $36 M can handle all tritium releases associated with the different blanket designs' inventories (<1 g to 2.3 kg in Tables 5.3-10 and 5.3-11).

TABLE 5.3-10.
A SUMMARY OF THE TRITIUM PROCESSING PARAMETERS SUPPLIED[a] FOR THE NINE TOKAMAK BLANKET CONCEPTS

Concept	Concept[b] Description	Mass Flow Rate to the Primary Recovery System (g/d)	Added Protium (g/d)	Tritium Partial Pressure in Breeder (Pa)	Tritium Load into Coolant (g/d)	Tritium Partial Pressure in the Coolant (g/d)	Tritium Load to Steam Generator (Ci/d)	Tritium Inventory[c] Location	Amount (g)
A	$LiAlO_2$/Be NS/SG HT-9 Limiter/H_2O[d] Helium Purge	924	9×10^4	1.3×10^{-2}	0.49	Not Available	Not[e] Available	First Wall Purge Coolant Blanket	19 10^{-4} 90 2000
B	---	---	---	---	---	---	---	---	---
C	---	---	---	---	---	---	---	---	---
D	Li Li/Na/SG V Limiter/Li Molten Salt/Electrolysis	1121	---	8.6×10^{-9}	0.03	---	3	First Wall Purge Coolant Blanket	6 --- --- 490
E	Li_2O He/SG HT-9 Limiter/H_2O Helium Purge	828	9×10^4	6.9×10^{-1}	1.86	3.6×10^{-4}	32	First Wall Purge Coolant Blanket	19 10.5 --- 134
F	$LiAlO_2$/Be He/SG HT-9 Limiter/H_2O Helium Purge	823	9×10^4	7.5×10^{-1}	12.45	1.3×10^{-3}	48	First Wall Purge Coolant Blanket	19 10 10^{-5} 38
G	Li He/SG HT-9 Limiter/Li Yt beds	866	---	10^{-7}	1.22	9.8×10^{-7}	27	First Wall Purge Coolant Blanket	19 --- 10^{-7} 330
H	FLIBE/Be He/SG HT-9 Limiter/H_2O Electrolysis	846	---	2600	122.22	1.9×10^{-1}	260	First Wall Purge Coolant[f] Blanket Blanket Structure	19 --- 0.2 0.5 189[g]
I	$LiAlO_2$/Be H_2O/SG HT-9 Limiter/H_2O Helium Purge	869	5.5×10^3	2.2	1.23	6×10^{-6}	2	First Wall Purge Coolant Blanket	19 0.03 53 230[illegible]

[a] The information was supplied by the team responsible for a given design.

[b] The first line is the breeder; the second line is the coolant; the third line is the structure; the fourth line is the coolant for the limiter/divertor; the fifth line is the tritium recovery method used. The abbreviations used are these: Be - beryllium; NS - nitrate salt; SG - steam generator; $LiAlO_2$ - γ-lithium aluminate; HT-9 - a ferritic steel; Li - lithium; Na - sodium; V - a vanadium alloy; Li_2O - lithium oxide; He - helium, and FLIBE - a lithium beryllium fluoride salt.

[c] The tritium inventory was used to size the atmospheric tritium recovery system. The options chosen were a 3-day cleanup for high probability releases ($>10^{-2}$) and 5-day cleanup for low probability releases ($< 10^{-2}$).

[d] For all limiters, it was assumed that 1 g/d of tritium permeated into the coolant.

[e] It was assumed that the tritium load was <20 Ci/d. If this is not valid, an additional control system needs to be included and costed.

[f] An assumed value since none was supplied. It was derived by scaling from the other helium concepts.

[g] Only blanket structure with an inventory >10 g.

TABLE 5.3-11.
A SUMMARY OF THE TRITIUM PROCESSING PARAMETERS SUPPLIED[a] FOR THE NINE MIRROR BLANKET CONCEPTS

Concept	Concept[b] Description	Mass Flow Rate to the Primary Recovery System (g/d)	Added Protium (g/d)	Tritium Partial Pressure in Breeder (Pa)	Tritium Load into Coolant (g/d)	Tritium Partial Pressure in the Coolant (g/d)	Tritium Load to Steam Generator (Ci/d)	Tritium Inventory[c] Location	Location (g)
A	$LiAlO_2$/Be NS/SG HT-9 HSDC/H_2O[d] Helium Purge	619	9×10^4	1.3×10^{-2}	0.45	Not Available	Not[e] Available	First Wall Purge Coolant Blanket	3.7 10^{-4} 50 250
B	Li Li/Na/SG HT-9 HSDC/H_2O Molten Salt/Electrolysis	564	---	8.6×10^{-9}	0.02	---	2	First Wall Purge Coolant Blanket	4 --- --- 336
C	LiPb LiPb/SG V HSDC/H_2O Counter Current He	731	4.2×10^4	1.3×10^{-4}	---	---	10	First Wall Purge Coolant Blanket	1 <1 --- <1
D	Li Li/Na/SG V HSDC/H_2O Molten Salt/Electrolysis	735	---	8.6×10^{-9}	[illegible]	---	2	First Wall Purge Coolant Blanket	1 --- --- 336
E	Li_2O He/SG HT-9 HSDC/H_2O Helium Purge	550	5.6×10^4	7.8×10^{-1}	0.85	6.4×10^{-4}	20	First Wall Purge Coolant Blanket	4 1 10^{-5} 131
F	$LiAlO_2$/Be He/SG HT-9 HSDC/H_2O Helium Purge	554	5.7×10^4	8.4×10^{-1}	0.51	1.9×10^{-3}	43	First Wall Purge Coolant Blanket	4 1 10^{-4} 24
G	Li He/SG HT-9 HSDC/H_2O Yt Beds	558	---	1.1×10^{-7}	0.2	7.8×10^{-7}	17	First Wall Purge Coolant Blanket	4 --- 10^{-7} 233
H	FLIBE/Be He/SG HT-9 HSDC/H_2O Electrolysis	622	---	2600	83.2	1.4×10^{-1}	191	First Wall Purge Coolant[f] Blanket Blanket Structure	4 --- 0.14 0.4 139[g]
I	$LiAlO_2$/Be H_2O/SG HT-9 HSDC/H_2O He Purge	586	3.7×10^3	2.2	0.21	6×10^{-6}	2	First Wall Purge Coolant Blanket	4 0.03 53 1500

[a]The information was supplied by the team responsible for a given design.

[b]The first line is the breeder; the second line is the coolant; the third line is the structure; the fourth line is the coolant for the halo scraper/direct converter the fifth line is the tritium recovery method used. The abbreviations used are these: Be - beryllium; NS - nitrate salt; SG - steam generator; $LiAlO_2$ - γ-lithium aluminate; HT-9 - a ferritic steel; Li - lithium; Na - sodium; V - a vanadium alloy; Li_2O - lithium oxide; He - helium; FLIBE - a lithium beryllium fluoride salt; LiPb - 17Li-83Pb; and HSDC - halo scraper/direct converter.

[c]The tritium inventory was used to size the atmospheric tritium recovery system. The options chosen were a 3-day cleanup for high probability releases ($>10^{-2}$) and 5-day cleanup for low probability releases ($< 10^{-2}$).

[d]For all mirror concepts it was assumed that ~1 g/d of tritium entered the water system cooling the halo scrapers and the direct converter.

[e]It was assumed that the tritium load was <20 Ci/d. If this is not valid, an additional control system needs to be included and costed.

[f]An assumed value since none was supplied. It was derived by scaling from the other helium concepts.

[g]Only blanket structure with an inventory >10 g

TABLE 5.3-12.
CAPITAL COST OF ATMOSPHERE TRITIUM RECOVERY AS FUNCTION OF SIZE OF RELEASE AND CLEANUP TIME[a]

Size of Release (g)	Cost of Cleanup in 72 h ($M)	Cost of Cleanup of 120 h ($M)
10	36[b]	25
50	40	28
100	42	29
200	44	31
500	46	32
1000	48	33
1500	49	34
2000	49	34
2300	50	35[b]

[a]Cleanup is to 50 $\mu Ci/m^3$ in a building 2×10^5 m^3.

[b]The same size system provides cleanup of a 10 g release in 72 h or a 2.3 kg release in 120 h.

5.3.3.2 Subsystem Assumptions

The subsystems which were considered part of the blanket tritium processing system were the following: (1) the implantation tritium control system for the coolant for either the limiter/divertor in a tokamak or the halo scraper/direct converter in a mirror; (2) the tritium recovery and purification system used for a given breeder blanket; (3) the tritium recovery purification and control systems required on primary or secondary coolants associated with a given blanket design; and (4) the coolant purification systems required for all non-water coolants. It was assumed that water purification costs were part of the basic plant design for all blanket concepts.

Implantation Tritium Control

The size of the tritium control system for the limiter/divertor or the halo scraper/direct converter depends on the rate of tritium implantation and migration. It is assumed for this study that this is equivalent to 1 g/d for all designs. The limiter coolant is either lithium or water. The capital costs associated with removing tritium from a water coolant[11] are summarized in Table 5.3-13. The general equation is for $C = 15\ [Ci/d/Ci/\ell\ (1000)]^{0.55}$.

TABLE 5.3-13.
CAPITAL COST OF TRITIATED WATER RECOVERY SYSTEM[a]

Processing Rate (L/D)[b]	Cost ($M)
10^2	6
10^3	15
10^4	50
10^5	100

[a]Maintains levels of 1 Ci/L in the water coolant.

[b]Equivalent to Ci/D removed if steady state concentration is 1 Ci/L.

Blanket Tritium Recovery

There are two types of tritium recovery and purification systems considered: the first is that associated with solid breeders (Li_2O, $LiAlO_2$); the second is that associated with liquid breeders (Li, LiPb, FLIBE). For the solid breeder, the tritium recovery system consists of a helium purge stream, associated pumps, and a system for removing tritium from the purge. In addition, there is a tritium purification system, a system for handling protium for tritium control, and a system for removing tritium from discarded solid breeder modules. For the liquid breeder, there is a basic tritium recovery system consisting of sets of cycled yttrium getters, a molten salt/electrolysis unit, a counter-current helium flow system, or an electrolysis unit. In addition, there's a tritium purification system and, if necessary for tritium control, a system for handling protium. The capital cost associated with these units was determined as follows.

Since no pilot plant version of any of the tritium recovery units has been built, it was assumed that to a first approximation all of the units would cost approximately the same. Therefore, one unit was costed basing costs on those for centrifugal contactors.[12] The volume of material in the liquid breeder designs was assumed to be 10^6 ℓ. If this volume was processed 100 times a day, the processing rate would be 4×10^5 L/min. For a one stage contactor which is not heated, or designed for tritium use, the cost is \$2.5 M; for 4 stages, the cost is ~\$4.4 M. If the unit is used at elevated temperatures and designed for use with tritium use, the cost would double to ~\$10 M. If two different recovery units were needed for a given blanket design, the cost would double to \$20 M.

The capital cost of a tritium purification system which removes both gamma and other impurities is ~\$10 M for most systems. This cost covers not only the basic unit, <\$5 M, but also the support systems needed since the unit will be located either in the heat exchanger building or the hot cell area. The tritium purification system for 17Li-83Pb is expected to cost ~\$15 M due to the large amounts of gamma impurity expected.

The addition of protium in 5 to 90 kg/d quantities requires an additional unit both to remove the tritium from the protium and to remove other impurities. To provide perspective, the fuel processing system is expected to handle <10 kg/d of deuterium and tritium. The capital cost incurred for adding the protium system is shown in Table 5.3-14.

It is anticipated that the hot cell will contain an extraction system to remove tritium from discarded solid breeder modules. The capital cost is estimated to be ~\$10 M.

Coolant Tritium Recovery and Control

The coolant tritium recovery systems consist of a tritium removal system, a tritium purification system, special heat exchangers and piping, and a tritium removal system for the steam generator. The latter is needed only if the tritium input to the steam generator exceeds 20 Ci/d. The costing is that shown in Table 5.3-13.

The tritium load into the blanket coolant is the sum of contributions from the first wall (<0.2 g/d), from the limiter/divertor system (<1 g/d), from the helium purge (<0.01 to 113 g/d) and from the beryllium multiplier (<6 to 9 g/d). The beryllium multiplier contribution applies only to concepts F) and H). In concept H), the FLIBE design, the tritium entering the helium coolant from the helium purge stream is excessive being 113 g/d and 83 g/d respectively for the tokamak and mirror designs.

TABLE 5.3-14.
CAPITAL COST OF AN ISOTOPE SEPARATION UNIT[a]

Throughput (g/d)	Cost ($M)
5	5
42	13
50	14
56	14
57	14
90	17

[a]Cost equation used:
$c = 3 \times 10^6$ (throughput (g)/1160)$^{0.4}$.

The capital cost of the systems used to process the liquid coolants other than water is assumed to be $10 M for tritium recovery and $10 M for tritium purification. For pressurized helium, the basic tritium recovery system is also assumed to cost $10 M. However, when large amounts of tritium have to be removed the capital cost increases. An estimate of these cost increases is shown in Table 5.3-15.

TABLE 5.3-15
CAPITAL COST OF HELIUM COOLANT CLEANUP SYSTEM

Processing Rate (g/d)	Incremental Cost[a] ($M)
0.2	--
1.0	4
10.0	1[illegible]
80.0	40
120.0	60

[a]The base system is assumed to be $10 million including pressurized lines, etc.

For the $LiAlO_2$, water cooled concept I), the major capital cost associated with coolant tritium recovery was accounted for in implantation tritium control. The cost of the additional capacity needed is \$10 M. For the 17Li-83Pb system, concept C), the capital cost of the tritium removal and purification system are those under blanket tritium removal. However, there are additional costs for a special heat exchanger and piping, because of the high tritium partial pressure (10^{-4} Pa) in this system. The estimated capital cost is \$20 M.

Coolant Purification Systems

For each of the coolants, an impurity removal system is necessary. It's cost is estimated at \$10 M. For concepts with two coolants (B, D, G, H), the cost doubles because there are now two impurity removal systems needed. Because the 17Li-83Pb system is regarded as highly corrosive, its impurity removal system is estimated at \$20 M. Besides coolant purification systems, dump tanks are provided for all coolant and/or liquid metal systems. The estimated cost for the tank heating systems and tritium control measures is ~\$5 M each. Again, for some concepts (B and D) two dump tanks are needed, which doubles the cost.

5.3.3.3 Calculated System Costs

The blanket processing system capital costs calculated for the tokamak blanket concepts are found in Table 5.3-16. Those for the mirror blanket concepts are found in Table 5.3-17. The total processing system costs are shown in Table 5.3-18.

For a tokamak reactor, the processing capital costs range from \$90M to \$190M with concept H), the FLIBE design, having the highest capital cost. Concept H) has a high tritium partial pressure in the FLIBE. This procedures high permeation rates in the different subsystems which then require additional tritium control systems being used.

For a mirror reactor, the capital costs of the blanket concepts are equivalent although those for concept I), the water-cooled design, are lower than the rest and concept (H) is the highest because of a high partial pressure in the FLIBE. The capital cost for concept I) would be comparable to that of the other concepts if the tritium load to the halo scraper was <1

TABLE 5.3-16.
COSTS ($M) ASSOCIATED WITH THE TRITIUM PROCESSING AND TRITIUM CONTROL SYSTEMS REQUIRED FOR THE NINE TOKAMAK BLANKET CONCEPTS CITED IN TABLE 5.3-10

Process Costed / Concept	A	B	C	D	E	F	G	H	I
Tritium Control Systems for Water Cooled Limiter/Divertor	50	---	---	---	50	50	---	50	50
Blanket Tritium Removal and Purification Systems	40	---	---	41	37	37	32	21	27
Coolant Tritium Removal and Tritium Purification Systems	20	---	---	20	28	44	28	90	10
Coolant Purification Systems –									
NS, He, Na,[a]	10	---	---	10	10	10	10	10	---
Li, LiPb, FLIBE	---	---	---	10	---	---	10	10	---
Coolant Dump Tanks –									
DS, Na	5	---	---	5	---	---	---	---	---
Li, LiPb, FLIBE	---	---	---	5	---	---	5	5	---
Tritium Extraction Systems for the Solid (Hot Cell)	10	---	---	---	10	10	---	---	10
TOTAL	135	---	---	91	135	151	85	186	97

[a]Abbreviations are: NS – nitrate salt; He – helium; Na – sodium; Li – lithium; LiPb – 17Li-83Pb; FLIBE – a lithium beryllium fluoride salt.

TABLE 5.3-17.
COSTS ($) ASSOCIATED WITH THE TRITIUM PROCESSING AND TRITIUM CONTROL SYSTEMS REQUIRED FOR THE NINE MIRROR BLANKET CONCEPT CITED IN TABLE 5.3-11

Process Costed / Concept	A	B	C	D	E	F	G	H	I
Tritium Control Systems for Water Cooled Halo Scraper/Divertor Converter	50	50	50	50	50	50	50	50	50
Blanket Tritium Removal and Tritium Removal and Tritium Purification Systems	40	30	51	40	34	34	30	23	25
Coolant Tritium Removal and Purification Systems	20	20	20	20	24	36	24	70	10
Coolant Purification Systems -									
NS, He, Na[a]	10	10	---	10	10	10	10	10	---
Li, LiPb, FLIBE	---	10	20	10	---	---	10	10	---
Coolant Dump Tanks									
DS, Na	5	5	---	5	---	---	---	---	---
Li, LiPb, FLIBE	---	5	5	5	---	---	5	5	---
Tritium Extraction Systems for the Solid (Hot Cell)	10	---	---	---	10	10	---	---	10
TOTAL	135	130	146	140	128	140	129	168	95

[a]Abbreviations are: NS - nitrate salt; He - helium; Na - sodium; Li - lithium; LiPb - 17Li-83Pb; FLIBE - a lithium beryllium fluoride salt.

TABLE 5.3-18.
SUMMARY OF BLANKET CONCEPT COSTS INCLUDING ALL TRITIUM PROCESSING COSTS

Concept	A	B	C	D	E	F	G	H	I
Tokamak									
Blanket Processing	135	---	---	91	135	151	85	186	97
Plasma Processing	40	---	---	40	40	40	40	40	40
Atmospheric Tritium Recovery	36	---	---	36	36	36	36	36	36
GRAND TOTAL	211	---	---	167	211	227	161	262	173
Mirror									
Blanket Processing	135	130	146	140	128	140	129	168	95
Plasma Processing	31	31	31	31	31	31	31	31	31
Atmospheric Tritium Recovery	36	36	36	36	36	36	36	36	36
GRAND TOTAL	202	197	213	207	195	207	196	235	162

g/d. If it was 0.1 g/d, implantation costs for the other concepts would decrease by $35 M while implantation and coolant control costs for concept I) would decrease by $15 M. In concept (I) the tritium load from the blanket necessitates retention of a high capacity system, thus the smaller cost reduction.

5.3.3.4 Conclusions for the Fuel Handling and Storage System

The capital costs of all tritium processing systems needed for the nine different blanket concepts considered are equivalent; the costs are in the range of 160 to 260 million dollars. This represents 5% of the total capital cost of a four billion dollar reactor.

These costs are the best estimate of several complex systems which not only have not been built at pilot plant scale but also have frequently not been demonstrated even at lab-top scale. Nevertheless, these costs considered to be representative of the costs expected. Confirmation of these costs would be beneficial for evaluation of the overall economics.

5.3.4 Economic Analyses

The heart of the economic analyses performed for the BCSS was accomplished with a systems analysis code which consistantly combined both system performance and economic evaluations. The code was custom tailored for the BCSS application and primarily evaluated the blankets' performance and costs, but also blanket-related functions on both tokamak and tandem mirror power plants. The inputs for the code were derived from a) the blanket designers for the blanket-specific data, b) the BCSS Task Groups for the more general blanket-related data (eg., Heat Transfer and Neutronics), c) the reactor conceptual designers (ala STARFIRE and MARS) for reactor and BOP data and d) vendors and suppliers for specific detailed information. Specific data inputs used in the analyses are shown in this section along with their influences. The code was also used as a data source for other disciplines in the BCSS study, such as the Safety Analysis.

The code utilized the detailed configuration of each blanket to determine the volume, mass and cost of material in the blanket zones. This detailed information for blanket concepts are available in Sections 7 through 10. First, the code determined the first wall and blanket zone volume which may be different for the inboard and outboard in the tokamak or identical throughout a tandem mirror. The inboard section for the tokamak is assumed to cover 20% of the wall surface. Then using the designer's specification for the material volume fraction and percent of theoretical density, the mass of each material in each zone was determined. The cost of the materials in the blanket was determined by applying the unit costs as defined in Table 5.3-5.

Costs for the shielding were determined in a similar manner. The designers also gave the detailed blanket thicknesses and material compositions to the Neutronics Task Group for the determination of the Tritium Breeding Ratio and the Shielding. In turn, the required shielding materials and thicknesses were determined for the economic evaluation. The shielding requirements differed for the inboard and outboard tokamak designs and also for the tandem mirror between and under the central-cell coils. The code accommodated these design variations.

A summary of the blanket and shielding thicknesses for the evaluated options is shown in Table 5.3-19. The sum of the blanket and the shielding

TABLE 5.3-19
INPUT DESIGN DATA

CODE	BDR/CLT/STR/MLTR	RCTR TYPE	BLK THICKNESS TM (m)	BLK THICKNESS TOK, INNER (m)	BLK THICKNESS TOK, OUTER (m)	SHIELD THICKNESS TM, COIL (m)	SHIELD THICKNESS TM, B/C (m)	SHIELD THICKNESS TOK, INNER (m)	SHIELD THICKNESS TOK, OUTER (m)	BLNKT EMF	BLNKT LIFE* (y)
A	$LiAlO_2$/DC/HT-9/Be	TOK		0.517	0.517			0.60	1.12	1.323	4.5
		TM	0.51			0.59	1.08			1.316	4.5
B	Li/Li/HT-9/-	TM	0.803			0.487	0.803			1.313	4.5
C	LiPb/LiPb/V/-	TM	0.903			0.4	0.75			1.294	3.3
D	Li/Li/V/-	TOK		0.64	0.75			0.62	0.95	1.272	6.8
		TM	0.803			0.482	0.807			1.254	5.6
E	Li_2O/He/HT-9/-	TOK		0.41	0.85			0.73	1.02	1.223	4.7
		TM	0.68			0.55	1.04			1.228	4.5
F	$LiAlO_2$/He/HT-9/Be	TOK		0.41	0.7			0.74	1.16	1.280	5.5
		TM	0.58			0.64	1.14			1.291	4.4
G	Li/He/HT-9/-	TOK		0.61	1.2			0.64	1.04	1.279	4.8
		TM	1.08			0.52	1.01			1.270	4.7
H	Flibe/He/HT-9/Be	TOK		0.41	0.85			0.75	0.99	1.511	4.5
		TM	0.85			0.47	0.95			1.549	4.3
I	$LiAlO_2/H_2O$/HT-9/Be	TOK		0.35	0.7			0.70	0.99	1.372	4.8
		TM	0.7			0.45	0.55			1.386	4.9

* Blanket life is adjusted to account for lower tokamak neutron wall loading in economic analysis code.

thickness plus the design allowances for gaps, etc., were used to determine the magnet size and location. In the case of the tandem mirror this determined the cost of the magnets. However in the case of the tokamak, the thicknesses influence not only the magnet cost but also the magnetic field on axis. This is important because the reactor fusion power scales as the fourth power of the on-axis field. Also shown on Table 5.3-19 is the Blanket Energy Multiplication Factor (EMF). The EMF determined the efficiency of the conversion of the fusion power into useful thermal energy extracted by the primary coolant. Thus the thermal power from the tandem mirror blanket is determined solely by a fixed fusion power and EMF, whereas the tokamak thermal power is also determined by the blanket and shield thicknesses.

The final information on Table 5.3-19 is the expected blanket calendar life, adjusted for the neutron wall loading. As was previously stated, the fusion power as modeled in the tokamak is variable, hence the life is somewhat different than stated by the design advocates for a $5 MW/m^2$ wall loading. This lifetime data was used in determining blanket replacement frequency and allowances for remote maintenance equipment sets.

The cost of the magnets were determined by the combined thicknesses of the blanket, shield and the required gaps. The tandem mirror magnet configurations and cost basis were adopted from MARS. As the blanket/shield/gap thicknesses vary, the field in the central cell is maintained and within first order effects, the current density in the magnet pack remains constant. Thus the current and magnet cross-sectional area are proportional to the magnet bore. For small variations, the base cost of the MARS central cell magnets can be scaled on a cost per unit volume (and mass) basis. The tokamak TF magnets must be adjusted for positional changes in both the inner and outer legs of the magnet. Reference 13 developed the geometry of a constant tension "D"-shaped magnet in terms of the position of the inner and outer magnet legs. Using this relationship, the code calculated the TF volumes and cost, normalized to the STARFIRE magnet volume. The remainder of the tandem mirror and tokamak coils are assumed to be unaffected for this analysis.

The next major area to be computed in the code is the Heat Transfer and Transport System. In this study, the thermal output power is transferred by a variety of primary coolants over a range of inlet and outlet temperatures. Table 5.3-20 lists these data along with the steam conditions predicted

TABLE 5.3-20
BCSS THERMAL-HYDRAULICS DATA

Code	Bdr/Clnt/Str/Multr	Rctr Type	Coolant Inlet/Outlet (°C)	Max IHX (°C)	Max Steam Temp (°C)	Max Steam Press (MPa)	Gross Effcy	Pumping Power* / Thermal Power (MWe/MWT)	= (Ratio)
A	$LiAlO_2$/NS/HT-9/Be	TOK	330/405	--	390	10.4	0.375	7.9/4225	0.0019
		TM	375/450	--	435	15.4	0.400	7.9/4225	0.0019
B	Li/Li/HT-9/-	TM	350/500	480	454	15.0	0.405	13/3185	0.0041
C	LiPb/LiPb/V/-	TM	380/530	--	510	16.5	0.420	38/3185	0.0119
D	Li/Li/V/-	TOK	300/550	525	510	16.6	0.423	33/5675	0.0058
		TM	350/550	530	510	16.6	0.423	8/3185	0.0025
E	Li_2O/He/HT-9/-	TOK	275/510	--	460	8.3	0.392	176/4000	0.0440
		TM	275/540	--	490	8.3	0.400	84/4000	0.0210
F	$LiAlO_2$/He/HT-9/Be	TOK	275/510	--	460	8.3	0.392	179/4000	0.0448
		TM	275/540	--	490	8.3	0.400	108/4000	0.0270
G	Li/He/HT-9/-	TOK	275/510	--	460	8.3	0.392	196/4000	0.0490
		TM	275/540	--	490	8.3	0.400	108/4000	0.0270
H	Flibe/He/HT-9/Be	TOK	275/510	--	450	8.3	0.389	200/4000	0.0500
		TM	275/510	--	450	8.3	0.389	175/4000	0.0438
I	$LiAlO_2$/H_2O/HT-9/Be	TOK	280/320	--	299	6.3	0.357	32/4000	0.0080
		TM	280/320	--	299	6.3	0.357	32/4000	0.0080

* Primary Coolant Loop Only

achievable in the steam generator. The lithium primary coolant is the only primary coolant deemed to require an intermediate heat exchanger. The thermal gross efficiencies commensurate with these steam conditions are listed in the Table 5.3-20. Each blanket designer determined a pumping power for his particular coolant and thermal-hydraulic conditions to handle a nominal amount of thermal power. The code ratioed these data to determine the pumping power for the specific thermal power calculated in each individual case. A fraction of this pumping power was recoverable as useful thermal energy depending on the fluid (0.85 for helium and 0.90 for the remainder of the coolants.)

For the tokamak, all the recoverable thermal energy, except for approximately 200 MW recovered in the limiter, was transported by the blanket coolant. In the tandem mirror, a smaller fraction of the useful energy was transported by the primary coolant. A sizeable fraction of the energy was recovered by the direct convertor at an efficiency of 64.3% and a moderate temperature thermal conversion cycle at 34.3% efficiency. These data are representative of the MARS design data. The power flow of the tandem mirror reactor was modeled and scaled in a manner similar to MARS. The sizes and power-handling capabilities of the Heat Transfer and Transport System determine the costs used in the economic analyses.

Given the above data on thermal power, gross thermal efficiency, pumping power, and other recirculating power requirements, the net electric power is determined for each reactor associated with a blanket concept. The Turbine Plant Equipment is comprised of many different elements. The cost of these elements scaled as either the gross thermal power or gross electrical power. The cost elements of the Electric Plant Equipment scaled as the recirculating power requirement or were constant for this analysis. The Miscellaneous Plant equipment was assumed to be constant. The Special Materials are usually special, high-cost fluids which are loaded or installed just prior to initial operation. The liquid metals or salts contained in the heat transport systems are calculated for inclusion into this cost category. From these data, the Total Direct Cost, the Total Capital Cost including indirects were computed. Then using the Net Electric Power, the Cost of Capacity was determined. The Annual Cost was calculated by combining the blanket life with the refurbishment or replacement cost of the blanket on an annual basis. Added to this blanket cost are other high-cost, non-blanket items replaced on a regular basis. This was combined with the Annual Operations and Maintenance Cost and the Annual Fuel Cost to arrive at a total Annual Plant Cost. This data allowed determination of the Cost of Electricity, which is the desired economic evaluation parameter.

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5.4 Safety

Fusion's ultimate potential as an energy source and societal acceptance will depend, in part, on how well fusion development is successful in protecting the public and environment from potential harm. Just as blanket choices should decrease costs and enhance fusion's economic attractiveness, so should blanket choices decrease potential risk to the public and enhance fusion's safety and environmental attractiveness. An increase in safety and environmental attractiveness of designs at the current stage of fusion development will tend to increase public support and acceptance, translate into some economic advantages, and increase the future general potential of fusion energy.

These aspects have motivated evaluation of blanket concepts in the general area of safety, the subject of this section. The following subsections contain an introduction (5.4.1), the four major parts of the safety evaluation (5.4.2 through 5.4.5), and the results and conclusions (5.4.6). Also, the resulting safety results have been summarized in Section 3.3.

Readers may delve into the discussion at several levels. Those primarily interested only in the results may refer to Section 3.3. Those interested in more information on the general approach, limitations, results, and key assumptions should read Subsections 5.4.1 and 5.4.6. Readers interested in more specific details of the bases of comparison and corresponding technical justification should refer to those parts of Subsections 5.4.2 through 5.4.5 as deemed necessary.

5.4.1 Introduction

The final safety evaluation methodology is basically the same as during the first year of the study, as was discussed in the BCSS Interim Report.[1] This subsection contains a brief introduction into the safety evaluation: its purpose, differences from past studies, general approach, interactions with the other evaluation areas, and key assumptions and limitations.

5.4.1.1 Purpose

The safety evaluation fulfills two major purposes. First, during the study, safety analysis and evaluation have helped to assist in the evolution of concept design, focus study attention and resources, and assist in narrowing the number of concepts. Second, safety evaluation of the final designs has produced a rank ordering of their safety attractiveness as part of the final detailed concept evaluation. These two general purposes will now be discussed in more detail.

First, during the course of the study, safety analysis has had several purposes:

- Assist design teams in improving the safety attractiveness of their designs as they have evolved
- Define design safety philosophy (see Subsections 5.4.1.3 and 5.4.3) to provide targets for design teams
- Identify and eliminate, where possible, serious safety-related potential flaws in concepts
- Provide input into the narrowing of the number of concepts remaining under consideration
- Investigate specific safety-related issues
- Interpret new experimental data from other fusion programs as they have become available
- Help prioritize resources within the study.

Second, toward the end of the study, the primary purpose of safety analysis shifted to evaluation of concept designs. Purposes included:

- Systematically compare the safety attractiveness of concept designs

- Highlight areas where more research is required

- Highlight potential flaws

- Highlight potential areas of concept improvement

- Examine the impact of using "low-activation" versions of steels

- In general, examine the trade-offs involved among various safety concerns.

It is emphasized that this evaluation has not determined the safest blanket concepts. Rather, it has been intended to determine the most attractive concept designs as they were presented for evaluation, given present data and understanding. Many possible safety concerns were mitigated or eliminated by the design teams. In fact, snapshots of the safety rank-ordering of blankets during the course of the study would have shown some changes as designs evolved.

Where trade-offs between safety and other evaluation areas were involved, designers were free to attempt optimization, either sacrificing safety advantages for gains elsewhere or taking penalties elsewhere for improved safety characteristics. Such attempts at optimization were imperfect for several reasons: lack of important data, evolution of the data base, evolution of some aspects of the detailed evaluation methodologies, and insufficient BCSS resources for iterations. However, the BCSS still represents the most detailed attempt yet at such an optimization in the fusion program.

5.4.1.2 Past Studies

The BCSS safety analysis and evaluation have been strongly based on past studies. First, the concept designs in the BCSS were often similar to past design efforts so that prior safety analyses were generally relevant. Second, the reference fusion reactors for this study, STARFIRE[2] and MARS[3,4], served to define many of the nonblanket details needed for the comparison. Third, the evaluation methodology and some specific analyses were heavily based on three specific studies, (a) the study of material choice influence on potential accident consequences by Piet, et al.,[5,6] (b) the general environmental comparison done for the Generic Environmental Impact Statement (GEIS),[7-9] and (c) the DOE panel on low-activation materials.[10,11] Fourth, a wealth of safety-related information has now accumulated via the Fusion Safety Program[12-16], which was used in the specific safety analyses of this study.

Some specific mention of earlier comparative safety studies is relevant. The earlier comparisons of safety attractiveness of blanket materials heavily focused on lithium-compound chemical reactivity and/or structural material activation, e.g. References 17-20. These issues were also prevalent in early design studies, along with considerable concern for tritium control. Attention to the chemical issue motivated a search for an alternative to elemental lithium and its chemical reactivity.

Gradually, more issues and concerns have been considered. For example, Holdren[21,22] and Sawdye[23] included consideration of the relative ability of activated structures to become volatilized in the event of a thermal transient. Design studies have also considered more issues, such as cooling transients and waste management.[2-4] The safety examinations for INTOR[24-25] and TFTR[26] also included many issues.

The recent study by Piet[5,6] compared the influence of blanket material choice on potential accident consequences in seven different major areas. cooling transients, plasma disruptions, tritium inventory and control, public health effects from release of structural radioactivity, relative rates of structural oxidation and volatility in thermal transients, chemical reactivity, and corrosion product inventory. All of these issues were also considered in the present study, with the accident-related analyses often based on that past work. The DOE panel on low-activation[10-11] examined the potential impact of low-activation materials in three areas: accidents, maintenance, and waste management. These concerns have also been considered in the present study. The panel's recommendation that waste disposal be considered via the 10CFR61[27] regulation has been followed. The technical report for the Generic Environmental Impact Statement (GEIS)[7] considered several nonaccident issues, especially normal radioactive effluents, which are also considered in the present study.

In short, the present safety evaluation has built on past studies and considered all safety or environmental issues previously raised. Most were actually included in the comparative evaluation.

5.4.1.3 General Approach

When comparative studies have focused only on one issue, the resulting decision was fairly straightforward. For example, when chemical reactivity was the major issue, elemental lithium was clearly the least attractive option. As the number of issues increases, the complexity of the decision increases. The comparative accident study[5,6] defined twenty-one specific figures-of-merit, called Relative Consequence Indices, to judge among the blanket material combinations. These revealed conflicting material preferences as functions of which safety issue was considered. That study resolved such trade-offs by focusing final selection on those safety concerns that appeared more difficult to solve by design.

5.4.1.3.1 Formalism

Given the mission of the BCSS, some type of organized, systematic comparison technique was clearly needed. Probabilistic Risk Assessment (PRA), e.g. the (Fission) Reactor Safety Study,[28] was considered but rejected because of insufficient data and resources. Instead, an approach similar to the Relative Consequence Indices of References 5 and 6 was used. Eleven safety indices were established. Each is composed of one or more figures-of-merit evaluations and/or engineering judgment scores.

Whereas the previous study combined the Relative Consequence Indices via judgment on which problems were more solvable by design, the mechanism of the BCSS automatically allowed for problems to be solved by design teams within the study. A PRA would have allowed the importance of the various problems to be determined directly from their contributions to total risk, in much the same manner as the importance of economic factors is determined by their contribution to cost-of-electricity. In lieu of a PRA, engineering judgment was used to weigh the importance of the various concerns.

The resulting formalism is a single Safety Figure-of-Merit (SFM), defined as the weighted sum of eleven safety indices.

$$SFM = \sum_{i=1}^{11} I_i W_i$$

I_i = safety index score, 0.00 to 1.00

W_i = judged weight for each index.

The weights add to 100 so that the SFM ranges from 0 to 100. This formalism is heavily patterned after that of Kepner-Tregoe decision analysis.[29]

This methodology combines several features:

(a) Experimental data

(b) Calculations

(c) Figures-of-merit comparing blankets on a specific issue, based on items a and b

(d) Engineering judgment on the severity of other issues not covered by specific figures-of-merit, based where possible on items a and b

(e) Index scores directly based on items c and d

(f) Combined Safety Figure-of-Merit based on weighting the index scores via engineering judgments.

No scheme for decision analysis is perfect. A danger in this approach is that numbers are simply "cranked out." However, the comments of Crouch and Wilson[30] are relevant:

> "We are unashamed proponents of calculating numbers whenever possible. Without such calculations, we cannot be sure that the risk assessor has thought through the problem at all. Of course, a number can be stripped of the uncertainty estimates and misused, but a noun or verb in any piece of description prose can also be stripped of its qualifying adjectives or adverbs. Despite our emphasis on numeracy in risk assessment, it is necessary to warn against the tendency to overburden decision makers with intricate details of the arithmetic process used in arriving at the numerical results. What is required by the decision maker is a concise statement of numerical results, their uncertainties, and the simplifying assumptions made in deriving such results. But most important of all is a statement of the areas in which no numerical results are possible so that value judgments are required."

The numerics of the safety evaluation are not designed for random number-crunching, but rather as a tool to

- Quantify comparisons wherever possible
- Combine engineering judgment with quantified results
- Force the study to explicitly consider and explain trade-offs among safety issues
- Guide the allocation of study resources

- o Force the study to think through the various issues and what they imply for various blankets.

Subsection 5.4.6 contains both the design rankings as well as discussion of the underlying trade-offs and reasons for the results.

The various safety problems and associated indices were divided into three major areas. The first, source term characterization (Subsection 5.4.2), is concerned with the hazard associated with radioactivity inventory and chemical toxicity adjusted by the relative volatility of the elements involved. These first three indices, structure, breeder/multiplier, and coolant, were given 30% of the total weighting, 10% each. These index scores are based on quantified figures-of-merit.

The second set, fault tolerance (Subsection 5.4.3), measures the anticipated blanket response to various transients, considering both the consequence and the likelihood of occurrence. The possible transients were grouped into five indices, Indices 4 through 8, and were collectively given 30% of the total weighting, 6% each. These index scores are primarily based on engineering judgement, backed by analysis on past or current designs.

Together, the first eight indices are concerned with accidents and comprise 60% of the score. The remaining nonaccident concerns, radioactive effluents, maintenance, and waste management, were given 40%. The 60-40% split between accidents and nonaccidents was a compromise between the high importance given to accidents by the public and the low contributions that accidents have been found to have versus nonaccidents in terms of total societal risk for other technologies.[31] The nonaccident indices, Indices 9 through 11, are based on quantified figures-of-merit. The eleven indices are listed in Table 5.4.1-1.

The evaluation of concepts is performed twice, once for comparisons of blankets for tokamak application, once for mirror application. The indices and the weights remain constant. However, the designs and some of the parts of the indices do change. The differences between the mirror-design

TABLE 5.4.1-1. SAFETY EVALUATION INDICES

Index Number	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	Occupational Exposure	10
11	Waste Management	10

and the tokamak-design of a given concept are addressed in the evaluation where relevant. The differences between mirror and tokamak are addressed only in so far as they influence the bases of comparison. An example is that the relative thermal impact of plasma disruptions is considered in Index 6 for the tokamak evaluation but is not relevant for Index 6 for the mirror evaluation. Cross-comparisons among designs for mirror and designs for tokamaks were not performed.

5.4.1.3.2 Uncertainties

Some aspects of uncertainty sensitivity should be mentioned, according to the source of uncertainty. At the higher level of the uncertainty in the relative importance of safety problems, i.e. the index weights, the sensitivity of the rankings was explored by defining two alternate safety figures-of-merit. The reference figure-of-merit is given by

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

The first alternate figure-of-merit, SFM1, is given by

$$SFM1 = 9.09 \sum_{i=1}^{11} I_i$$

so that the indices are equally weighted. The constant 9.09 (= 100/11) results in SFM1 ranging from 0 to 100 as does SFM. The second alternate figure-of-merit, SFM2, is more directly patterned after a PRA. The total source term characterization score, Indices 1-3, is multiplied by the total fault tolerance score, Indices 4-8, giving an approximation of accident risk. This is then added to the nonaccident indices as follows:

$$SFM2 = \left(\sum_{i=1}^{3} I_i\right) \times \left(\sum_{i=4}^{8} I_i\right) \times 4.0 + \left(\sum_{i=9}^{11} I_i W_i\right) .$$

The constant of 4.0 preserves the 60-40% split between accident and nonaccident concerns as well as the 0-100 range for SFM2.

Calculation of the alternate figures-of-merit provided some insight into the sensitivity of the safety rankings to the uncertainty in the judged importance of different safety issues.

The other type of uncertainty is the relative attractiveness of a given blanket for a given issue, i.e. the individual index scores. No systematic analysis of the sensitivity of the results to these uncertainties has been conducted. However, throughout the evaluation, the key uncertainties are mentioned. This forms the basis for the R&D assessment as well as give some insight to the uncertainty in the index scores. Relevant considerations include uncertainty in the data, interpretations of the data, and design specifications. Specific areas examined include the sensitivity of the results to use of "low-activation" steels and to the uncertainty in tritium control calculations.

5.4.1.3.3 <u>Design Philosophy</u>

Criteria for judgment are inherent in a comparison. The design teams had to have some knowledge of how the safety attractiveness would be judged. These criteria can be either "musts" and "wants." The current NRC approach to fission reactor safety involves a large number of "musts," absolute requirements, some of which may not be directly related to controlling risk. It is hoped that fusion can avoid such proscriptive regulation, perhaps replaced by a probabilistic risk approach. For example, the NRC and others are considering a two-stage approach for fission whereby first a reactor must meet some societal risk goal, e.g. limiting the risk of early fatality below 5×10^{-7}/year.[32] Second, further risk reduction would be required based on some cost-risk reduction criterion, e.g. required expenditure of $1000 to reduce exposure by one man-rem. This cost-benefit approach has been applied to tritium system sizing for fusion.[33] However, the need for PRA's in this two-stage approach prohibited its use in the BCSS.

Since the proscriptive NRC-type criteria approach is not desired and the probabilistic approach was not feasible for the BCSS, some other approach was needed for he BCSS. In terms of design philosophy, the approach has been to define very few proscriptive safety requirements. These were indicated and explained in the Interim Report[1] as screening criteria for the alternate concept selection. The safety-related requirements are:

- Normal tritium loss rate via the steam generator must be below 100 Ci/day

- No materials combination resulting in large rate of energy released is allowed. The spirit behind this is that a single failure must not result in a large energy release. At the beginning of the study, this was taken to rule out a water-cooled lithium blanket. As discussed in Subsection 5.4.3.5, it was later decided that it also appears to rule out the combination of water-cooled limiters with lithium blankets.

- The steady-state blanket tritium inventory must be below 10 g/MWth, i.e. 40 kg for a 4000 MW plant.

Beyond these few requirements, all other safety concerns were only expressed as desires, "wants." To provide guidance to design teams, the bases for comparing designs were established early in the study. The interim report listed 13 safety targets associated with the 11 safety indices. These corresponding design philosophies are repeated in Subsections 5.4.2 through 5.4.5 with the associated safety evaluation of concept design. As designs and analyses advanced, only some modification of these targets occurred.

The net result is that designers had very few safety requirements, but several targets. These bases for comparative judgment were established for designers to take account of.

The underlying safety philosophy is an inherently safe reactor, i.e. one not needing any active protection systems. Thus, the approach for the fault tolerance part of the evaluation was to define what would demonstrate passive protection against various transients. Failure to meet this safety philosophy for each specific transient was not automatically considered a potential serious flaw for the concept because the fall-back position of using an active protection system was generally available. A perfect fault tolerance score would indicate an inherently safe reactor. Deviation from a perfect fault-tolerance score indicates increased need for active protection systems and/or redesign.

5.4.1.4 Interactions with other Evaluation Areas

Final designs were evaluated in all four evaluation areas of safety, economics, engineering feasibility, and R&D requirements. Thus, in principle, any design feature that improved safety that also adversely impacted other evaluation areas would both raise the safety score and decrease some other score. For example, use of a nonwater-cooled limiter was very important to the safety of blankets with liquid lithium or vanadium alloys and, in fact, nonwater-cooled limiters were chosen. This improved safety feature translated into a penalty in the R&D evaluation in the form of another potential flaw for those concepts--"what happens if the nonwater cooled limiter is not feasible?"

Table 5.4.1-2 lists some of the key safety design choices/features that impact nonsafety evaluations. These are discussed in the following subsections as relevant. On the other hand, most safety features were inherent to the concept and did not entail nonsafety penalties. For example, the liquid-metal self-cooled designs have several inherent passive safety features: there are no coolant tube break problems within the blanket, and they are generally very resistant to any cooling transient because of the good heat-sink characteristics of liquid metals.

TABLE 5.4.1-2. SOME KEY SAFETY DESIGN CHOICES/FEATURES IMPACTING OTHER EVALUATION AREAS

Blanket	Choice/Feature	Impact
Water-cooled	Selection of double-wall tubing within the blanket to reduce tube break frequency Selection of pod geometry to accommodate accidental water pressurization of blanket from tube break	More structure (E), Lower breeding ratio (F), More complexity (F,R)
Li/He, Li/Li	Use of building cover gas other than air and carbon dioxide i.e. nitrogen, to reduce lithium-fire concern	Additional potential flaw if not practical (R)
Li/He, Li/Li	Use of nonwater limiter coolant	Additional potential flaw if not practical (R)

E--economic evaluation
F--engineering feasibility evaluation
R--R&D evaluation

The major impact of safety-related design choices has been a significant improvement in the safety attractiveness of blankets containing lithium and/or vanadium. Both materials have severe chemical reaction concerns: energy release for lithium in contact with air, water, or carbon dioxide and rapid oxidation of vanadium above 650°C with oxidizing potential above 10^{-4} atmospheres. Designers largely eliminated those concerns via design choices, e.g. no air or carbon dioxide in the building for lithium designs, no air in the building for vanadium designs, and no water-cooled limiter for vanadium or lithium designs. The final safety results were strongly influenced by these design choices.

The interaction among evaluation areas had some limits. In fact, these limitations were a main cause for having to do separate evaluations. If all concerns could have been reduced to economics, there would have been less need for a separate comparative safety evaluation. Some key issues occur between safety and economics in the basic form of the economics evaluation not showing the economic advantages of safer blankets.

First, a blanket with fewer passive safety features, hence a lower safety score, would tend to require more active safety systems. Few of these could be costed, hence most were not included in the economic evaluation. Second, safer blankets could result in higher availability from either fewer failures or less downtime for repair. However, there are insufficient data to judge this impact; the economics evaluation assumed a fixed availability of 80% for all blankets. Third, safer blankets could result in reducing construction/licensing time, saving interest charges. A safer blanket should allow for faster licensing. However, there are no data to judge that impact; the economics evaluation had to assume a fixed construction time of years. Fourth, a safer blanket implies less health risk to the public. In many cases there is also likely to be less financial risk from accidents to the reactor owner. These examples serve to demonstrate the need for separate evaluation areas because all issues could not be put in quantified economic terms.

Some key issues also occur between safety and engineering feasibility. The actual safety and environmental attractiveness of a fusion reactor will depend on how well it is actually constructed and operated. However, this is not directly accounted for in the safety evaluation. To the extent that a blanket is simpler, it appears reasonable to expect that construction and operation would be done better. Since one part of the engineering evaluation is to measure blanket complexity, there is a strong connection between the safety and engineering evaluations: blankets evaluated as simpler in the complexity part of the engineering evaluation should have safety advantages in the area of better construction and operation.

5.4.1.5 Key Assumptions and Limitations

Many of the key design assumptions and evaluation limitations have been mentioned in Subsections 5.4.1.3 through 5.4.1.4 and not repeated here. Some underlying additional assumptions and limitations need to be mentioned here.

First, in the area of structures, issues such as a potential rise in the ductile-brittle transition temperature (DBTT) in HT-9, excessive ferromagnetic effects in HT-9, and differences in crack propagation among alloys and designs have not been included in the safety evaluation. A DBTT above operating temperature would likely eliminate an alloy from consideration. A DBTT above room temperature would impose severe constraints at shutdown and might not be feasible. It is not now known if the ferromagnetic properties of HT-9 would definitely allow its use in magnetic fusion devices. These matters were considered only in the R&D evaluation. The possibility of substantially different crack propagation behavior among alloys in reactor service could have important safety implications but could not be considered. The designs were set according to uniform ASME-type criteria, and crack propagation has not been directly considered.

Second, the blanket response to various transients has been largely based on judged extrapolation from past calculations to the current designs. In most blanket/transient cases, adequate tools do exist to perform the transient analysis, which would have allowed judgment to be replaced with "hard" analysis. Neither resources nor time were sufficient to do so. However, in most cases where the tools do exist, there is a good foundation of past calculations and understanding for which a high-confidence engineering judgment can be made for the current design. There are blanket/transient combinations where adequate calculational tools do not yet exist. These represent R&D needs, and details are mentioned in later subsections.

Third, all numerical values are based on a fixed 5 MW/m^2 design to fit STARFIRE or MARS, consistent with the engineering feasibility evaluation. Although other evaluation areas were focused on 7 (tokamak) and 9 (mirror) concepts, the safety evaluation was conducted on 16 concepts for both physics options. This allowed easier identification of relevant trends in the comparison. However, it should be noted that the evaluation of those concepts not in the narrower 7/9 list is more uncertain than those 7/9 concepts given a full evaluation in other evaluation areas because these additional concepts were not given as much design and evaluation attention. In a sense, the safety evaluation of blankets not given a final evaluation in other areas should be viewed as sensitivity cases, e.g. what is the effect in a design in replacing HT-9 with PCA?

Those readers not interested in the details of the evaluation are referred to Subsection 5.4.6 for discussion of results. Subsection 5.4.2 and 5.4.3 contain accident related evaluations, source term characterization and fault tolerance respectively. Subsections 5.4.4 and 5.4.5 contain nonaccident evaluations, effluents, maintenance, and waste management.

5.4.2 Source Term Characterization

The first part of the safety evaluation, the source term characterization, is the radioactivity and chemical toxicity source term available for possible accidental release. The second part, fault tolerance, compares the blanket responses to specific transients. Nonaccident issues, effluents, occupational exposure, and waste management, comprise the rest of the evaluation. Tritium, neutron activation products, and beryllium chemical toxicity are considered in the source term characterization. This first part of the evaluation is weighted 30% of the total Safety Figure-of-Merit and is divided equally among the structure, breeder/multiplier, and coolant.

The relevant design philosophy concerning the source term is that concept design should minimize radioactivity and chemical toxicity, both inventory and vulnerability. However, each blanket concept automatically defines the choice of structure, coolant, breeder, and multiplier. Thus, designers of each concept had relatively little leeway to reduce the source term compared with the flexibility designers had in improving the fault tolerance of their designs to specific transients.

Both the consequence (inventory) and the probability (vulnerability) elements of risk are included. The response to specific transients is not; rather, accident fault tolerance is covered by the second part of the evaluation. The following subsections include a description of the general approach; the evaluations for structure, breeder/multiplier, and coolant; and a brief summary.

5.4.2.1 General Approach

The source term characterization evaluation is divided equally into structure (Index 1), breeder/multiplier (Index 2), and coolant (Index 3). The corrosion product inventory in the coolant is considered with the coolant evaluation. These three source terms were separated into three indices because (a) the release pathways and mechanisms often differ, and there is insufficient data to know how to directly combine them, and (b) a separate presentation of results for each blanket component would be more transparent to the reader.

For each index, the evaluation is divided into two equal parts, relative activation product hazard and relative tritium hazard. Chemical toxicity is included with activation product hazards, where relevant. Tritium is considered separately because of its special nature and high mobility. Also, because the possible release pathways and conditions differ so much between tritium and activation products, there is no unambiguous method to combine their hazards, other than a PRA-type approach.

Index scores are based on relative values of various figures-of-merit, as defined in each index. Where the figure-of-merit varies among blankets by less than 1-2 orders of magnitude, the relative ranking is the appropriate linear utility function of the figure-of-merit. Otherwise, it is logarithmic, which has the virtue of minimizing the effect of uncertainties relative to a linear utility function. These various utility functions are normalized so that the best blanket receives the maximum credit for that particular figure-of-merit and the worst gets a 0.0. Individual index scores range from 1.00 to 0.00. Because the mirror concepts and tokamak concepts are scaled to different total reactor sizes, the absolute magnitudes of the inventories, hence the corresponding utility functions, differ. The evaluation is completely relative. The only potential for a fatal flaw due to the source term characterization is if the tritium inventory of a concept were to exceed 40 kg, as was established in the blanket concept screening criteria.(1) None of the final concepts came within an order of magnitude of this maximum tritium inventory.

5.4.2.1.1 Activation Product Hazard

The activation half of each evaluation is based on the relative hazard to the public of the material activation products adjusted by the degree of vulnerability. For the coolant and breeder indices, relative hazard is measured by the Biological Hazard Potential in air (BHP). BHP is defined as the activity divided by the Maximum Permissible Concentration (MPC) for each isotope for public areas. MPC values in 10CFR20(34) were used. For isotopes not listed in 10CFR20, MPC values by Fetter(35) were used.

For the structural material index, sufficient information exists to go beyond the BHP measure. Relative hazard is defined by the relative public health effects (PHE) per unit material released, in the manner of references 5, 6, and 36. Actual doses and corresponding health effects are a truer basis for judging hazard than is BHP.(5,6,22,36) The total expected latent health effects (cancer) per volume of material released atmospherically is used. The public health effects (PHE) considered in this study are the total latent health effects to the population surrounding a generic site.[5,36] This total sums over both the latent cancers from immediate exposure from an atmospheric release (groundshine, cloudshine, and inhalation pathways) and from the chronic exposure due to the presence of the activity deposited in the assessment area (ingestion, inhalation, and groundshine pathways). These are calculated with the FUSECRAC[44] code. The details of the generic site are not particularly important since only the relative, not absolute, effects from one material to the next are relevant. The interested reader can refer to Reference 5 for more details· ısically, the generic site is a composite of the sites considered in WASH-1400.[28] No evaluation of interdiction of food stuffs or evacuation was considered.

Generally, the BHP or PHE measure of hazard i^ adjusted by the relative volatility of the elements involved. This accounts for the relative vulnerability of the materials. Thus, the isotopes of more volatile elements, like molybdenum, are more heavily weighted. Since experiments indicate that different elements are released at different rates from materials, it is appropriate that the contribution of each isotope to total PHE or BHP be adjusted accordingly. The specific data are given under each index.

5.4.2.1.2 <u>Chemical Toxicity Hazard</u>

Many of the elements in the blanket are chemically toxic in addition to being radiologically hazardous after operation. The toxicity of the blanket elements have been examined to see if they are significant compared with radiological concerns. Toxicity is often expressed in terms of concentration limits, generally called Threshold Limit Values (TLV). These TLVs are industrial standards designed to protect workers from various toxins, somewhat analogous to MPCs for radiation standards. Just as public health effects from radiological dose calculations are a superior measure than MPCs, so would health effect determinations from chemical exposure be better than TLV measures. The NRC[37] has studied the data base for chemical toxin effect-exposure relationships. For fusion blanket-relevant elements, no definitive effect-exposure relationships can be derived from the data.[37] Thus TLVs are the best available measure of chemical toxicity hazards.

Table 5.4.2-1 lists concentration limits for some relevant elements, limits to avoid acute health impact, standards for worker protection, and standards for public protection. These limits were compared with radiological limits for those elements that are more chemical toxic and have lower activation levels, beryllium, lead, lithium, and vanadium. To perform the comparison, the MPC limits (in Ci/m^3-air) were divided by the specific activity (in Ci/kg) produced in various elements to obtain a radiological hazard limit (in μg element/m^3-air, the same units as chemical toxicity limits). The specific activity values are for average blanket conditions at 5 MW/m^2 wall loading for 5 years from References 1 and 5.

Easterly[9] has examined the special case of lithium. Assuming 700 mg/kg as the mean lethal dose of lithium carbonate and 1 Ci/m^3 of ^{60}Co in lithium, he determines that the time required to breathe a mean lethal dose chemically is 10^3 shorter than the time to breathe a mean lethal dose radiologically. Actually, the ^{60}Co concentration would likely be much higher than 1 Ci/m^3, and other isotopes would be present. Thus the radiological and chemical lethal doses are closer than calculated by Easterly. However, it is highly unlikely that a mean lethal dose would be obtained in either case as 17 hours of breathing the fumes from a lithium fire would be required to give an acute lethal dose of lithium carbonate.[9] Therefore, the comparison should be made on latent effects, not acute fatality. In this case, the radiological hazard appears worse since chronic lithium carbonate exposure has not been shown to have latent health effects and the chemical form of lithium carbonate could well change once out in the environment.

The results of the comparison are given in Table 5.4.2-2. The radiological hazard from lead and vanadium are seen to be worse than their chemical hazards because the radiological limit is more strict, i.e. lower. On the other hand, since beryllium activates so little and is very

TABLE 5.4.2-1. CONCENTRATION LIMITS IN AIR CAUSED BY CHEMICAL TOXICITY

	Concentration Limit		
Element	Acute Effect[a]	Workers[b]	Public[c]
Aluminum			
as Al_2O_3 dust		15 mg/m^3 (40)	
as Al		Harmless (40)	
Beryllium			
elemental or compounds	25 μg/m^3 (40, 41, 43)	2 μg/m^3 (39-43)	10 ng/m^3 (38, 40, 41)
Chromium			
insoluble chromates		50 μg/m^3 (43)	
Cobalt			
dust, fume		500 μg/m^3 (42), 50 μg/m^3 (43)	
Copper			
fume,		100 μg/m^3 (40), 200 μg/m^3 (43)	
dust, mists		1 mg/m^3 (40,43)	
Fluorine			
as F		2.5 mg/m^3 (42,43)	
as F_2	4 mg/m3 (43)	200 μg/m^3 (42), 2 mg/m^3 (43)	
as HF	~1 g /m^3 (41)	2-2.5 mg/m^3 (39-43)	
Lead			
inorganic compound	450 mg/m^3 (43)	150 μg/m^3(40,43)	50 μg/m^3 (18)
		200 μg/m3 (41,42)	
Lithium			
as LiH		25 μg/m^3 (39,42,43)	
Manganese		5 mg/m^3 (43)	
Molybdenum			
soluble	10 mg/m^3 (43)	5 mg/m^3 (41,43)	
insoluble	20 mg/m^3 (43)	15 mg/m^3 (41)	
		10 mg/m^3 (43)	

TABLE 5.4.2-1. (continued)

Element	Concentration Limit: Acute Effect[a]	Workers[b]	Public[c]
Nickel			
metal		1 mg/m^3 (43)	
soluble	300 $\mu g/m^3$ (43)	100 $\mu g/m^3$ (43)	
Vanadium			
V_2O_5 dust	1.5 mg/m^3 (43)	500 $\mu g/m^3$ (41-43)	
V_2O_5 fume		100 $\mu g/m^3$ (42), 50 $\mu g/m^3$ (43)	

a. Acute effect--concentration limit in air to avoid acute health impact. Reference numbers are given in parenthesis.

b. Workers--Concentration limit to avoid chronic health impact for workers, assuming 40 hour week.

c. Public--concentration limit to avoid chronic health impact for the public.

TABLE 5.4.2-2. COMPARISON OF CONCENTRATION LIMITS FROM CHEMICAL HAZARDS TO THOSE FROM RADIOLOGICAL HAZARDS

Element	Chemical Limit[a]	Radiological Limit[b]
Beryllium		
--worker	2 $\mu g/m^3$	High, impurity controlled
--public	10 ng/m^3	High, impurity controlled
Lead		
--worker	150 $\mu g/m^3$	<34 $\mu g/m^3$
--public	50 $\mu g/m^3$	<1 $\mu g/m^3$
Lithium (LiH)	25 $\mu g/m^3$	High for pure lithium
		$<10^3$ $\mu g/m^3$ for V/Li corrosion products
		<10 $\mu g/m^3$ for PCA/Li corrosion products
Vanadium	50 $\mu g/m^3$	<15 ng/m^3 for V-alloy

a. Chemical limit from Table 5.4.2-1.

b. Radiological limit based on MPC for isotopes produced at 5-year blanket exposure to 5 MW/m^2. Isotope inventory from References 1, 5.

toxic, the chemical toxicity of beryllium is significant. The case of lithium is less clear. The chemical toxicity of pure lithium would be worse than its radiological hazard if tritium and corrosion products in the lithium are neglected. In this study, the hazards of all elements except beryllium are taken to be dominated by radiological concerns.

5.4.2.1.3 Tritium Hazard

For the tritium half of each evaluation index, the vulnerable amount of tritium is the basis for comparison. Vulnerability is determined by determination of how much of the tritium present could reasonably be expected to be mobilized during transients. The primary case where the vulnerable inventory is substantially lower than the total amount present is $LiAlO_2$. Other materials generally have sufficiently high tritium diffusivity to allow tritium to escape within a short period (~hour) of time.

5.4.2.1.4 Transportability

For the public to be harmed by any of these toxins, the mobilized material must be transported into the reactor building from the reactor and then released from the building; i.e. a containment failure. There are insufficient data to distinguish among the transportability of activation products or among tritium released from one material to the next. Thus, transportability is not relevant to the relative comparison and is not included. The chance of containment failure is, in some cases, dependent on blanket material choices and design features. This is considered in the last fault tolerance evaluation, Index 8. It is emphasized that there is neither need nor intent to calculate absolute amounts of material that may be released to the environment. This study is comparative.

5.4.2.2 Structure Source Term Characterization, Index 1.

The structural hazard divides into activation products and tritium. The activation products can conceivably be mobilized by two general accident pathways, chemical or physical. The scoring scheme for the Structure Source Term Characterization is given in Table 5.4.2-3. The following sections contain discussion of these various pathways. It should be noted that the corrosion product release mechanism is accounted for under the coolant evaluation (Index 3) and not here.

5.4.2.2.1 Activation Hazard--Chemical Pathways

The amount or concentration of activations products is not, in itself, relevant to accident analysis. The real issue is what are the adverse effects to the public due to some accident mobilizing those activation products. Thus, a better way than activity (curies) to compare structural

TABLE 5.4.2-3. SCORING SCHEME FOR STRUCTURE SOURCE TERM CHARACTERIZATION, INDEX 1[a]

Relative Vulnerable Activation Hazard--Chemical Pathways (25% of Index)

Basis: F_{11} = Number of latent public health effects per element per unit volume of first wall material released atmospherically (PHE_e) times the relative volatility of that element (V_e)

$$F_{11} = \sum_e (PHE_e \times V_e)$$

Relative Vulnerable Activation Hazard--Physical Pathways (25% of Index)

Basis: F_{12} = Number of latent public health effects per unit volume of first wall material released atmospherically

$$F_{12} = PHE = \sum_e PHE_e$$

Relative Vulnerable Tritium Inventory (50% of Index)

Basis: F_{13} = Total amount of tritium in the blanket structural material

Index Score

I_1 (TMR) = $1.062-14.861(F_{11}) - 2.787(F_{12}) - 0.1046 \ln(F_{13})$
I_1 (TOK) = $1.299-14.861(F_{11}) - 2.787(F_{12}) - 0.1417 \ln(F_{13})$

a. Contributions of figures-of-merit to index score determined as follows: index arbitrarily divided equally into activation products and tritium because the release conditions and pathways differ too much to know how to combine them; the activation product half was divided equally into chemical and physical pathways for the same reason.

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materials is to compare the relative public health effects (PHE) for those activation products that are mobilizable. This is the approach used in this study.

If all the isotopes in a structural wall were released at approximately the stoichiometric rate, then only the PHE per amount of material would be needed for a comparison. However, in chemical-type transients, it is now known that the wall release rate is not stoichiometric.[45-51] Therefore, for chemical-type transients where chemical processes, generally oxidation, are the mechanisms for activation product mobilization, the PHE must be adjusted to give increased weighting for those isotopes that would be preferentially mobilized. Thus, the relative comparison is based on the <u>public health effects</u> (PHE_e) per element per unit volume of material times the <u>relative volatility</u> (V_e) of that element. Since the first wall is the most susceptible to accidental chemical attack of steam, air, carbon dioxide, etc. in the plasma chamber, the PHE per unit volume is based on the first wall.

Unfortunately, quantified release rates for all chemical environments do not exist. In general, the possible gases are air, steam, carbon dioxide, and nitrogen. Steam would be present if a water-containing component broke. Air, carbon dioxide, and nitrogen are possible building atmospheres. In the BCSS, only carbon dioxide and nitrogen were chosen as building atmospheres, carbon dioxide for non-elemental-lithium blankets and nitrogen for lithium-containing blankets. An additional dimension is the possible presence of other agents like lithium, 17Li83Pb, Nitrate Salt, FLIBE, etc. that may influence the reaction. One example is a lithium-air fire. In the whole matrix of possible chemical environments, significant data exist only for a few.

The relative volatility was judged via experiments in <u>air</u> because (a) all three structural materials have been tested in air in similar conditions in the same apparatus, (b) very few tests on these materials have been completed in other oxidizing environments at relevant temperatures, and (c) results in steam are not likely to be greatly different then air. The volatility was based on the fractional release of individual elements at various times and temperatures.[45-48] The temperature range for these experiments and corresponding assigned volatilities was 700-1200°C. The test durations varied from 1 to 20 hours.

The experimental results[45-48] were put into the form of fractional release, the fraction of each element present initially that was volatilized from the sample. Elements could then be compared on that basis. First, it was seen that the fraction of the initial amount of molybdenum in PCA, molybdenum in HT-9, and vanadium in V15Cr5Ti that was released for the same temperature and time appears similar. Thus molybdenum and vanadium were assigned relative volatilities of 1.00.

For each alloy, the relative volatility of others elements were based on their fractional release compared with the fractional release of either molybdenum or vanadium, averaged over the scoping tests. Thus, for example it was found that, on average, the fractional rate of chromium release from PCA is 10% of that of molybdenum from PCA. These ratios of fractional release rates form the basis for assigned volatilities of elements from the candidate alloys, Table 5.4.2-4.

The relative volatilities for some elements had to be estimated from others on the basis of the positions in the periodic table because existing tests were not sensitive for those elements or the elements are not present in unirradiated material samples. For example, scandium isotopes are very important in V15Cr5Ti, but scandium is not initially present in the alloy; therefore, the previous scoping tests in air did not determine scandium behavior.

An attempt was made to associate the scoping test findings with oxide or elemental vapor preserves. However, no correlation was found; the vapor pressures vary by over ten orders of magnitude, and experimental results of fractional release varied only by one or two.

Some tests have been conducted on the two steels in impure helium conditions.[49-50] The early findings do not show marked relative difference among elements from the comparison in air. Because these tests were performed on irradiated samples, some trace elements could be detected. Specifically, silver in HT-9 and arsenic in PCA appear very volatile. Because the amounts present are so small, such trace elements are not expected to be significant contributors to accidental exposure. However, they may be relevant to waste management.

Given the fractional release data, the volatility of an element in a specific transient could vary appreciably. However, the intent of this comparison is to examine behavior over all transients. Since the relative volatilities were assigned averaging over all test conditions, it is felt they are accurate within an order of magnitude. The major uncertainties are the specifics of a given transient and the behavior of these materials in nonair environments.

TABLE 5.4.2-4. ASSIGNED RELATIVE VOLATILITIES FOR RELEVANT ELEMENTS FOR THE STRUCTURAL MATERIALS (Mo ≡ 1.0)

Element	PCA	HT-9	V15Cr5Ti
Ca	--	--	(0.50)
Sc	--	--	(0.50)
Ti	1.00	(1.00)	0.50
V	(1.00)	(1.00)	1.00
Cr	0.10	0.08	0.20
Mn	0.50	0.80	--
Fe	0.08	0.05	--
Co	(0.07)	(0.33)	--
Ni	0.05	0.60	--
Nb	(1.00)	(1.00)	--
Mo	1.00	1.00	--
Tc	(1.00)	(1.00)	--
Ta	(1.00)	(1.00)	(1.00)
W	(1.00)	(1.00)	--

( ) = Estimate, assume Ca, Sc = Ti; Fe < Co < Ni; Nb, Tc, Ta, W = Mo.

5.4.2.2.2 Activation Hazard--Physical Pathways

The chemical pathway comparison of the accidental structural activation hazard is not, by itself, a sufficient comparison of activation hazard because (a) there are large uncertainties in the relative transportability of elements, and (b) there appear to be nonchemical means of mobilizing material. These physical means include sputtering of the first wall, disruption vaporization of first wall, and neutral beam vaporization of first wall. The vacuum chamber containment would have to be breached for release of these activation products to the reactor building, but that is a credible scenario. For these reasons, it appears necessary to also compare structural materials on the basis of stoichiometric activation product release.

Thus, this comparison is based on the relative public health effects per unit volume of first wall, assuming all isotopes are mobilized at the stoichiometric rate: $F_{12} = PHE = \sum_e PHE_e$

5.4.2.2.3 Relative Vulnerable Tritium Inventory

The amount of tritium normally contained in the structure varies considerably among materials and can reach significant values. Unfortunately, reasonably possible thermal transients could "bake out" this tritium in a short period of time, less than an hour, because of fairly high diffusion rates of tritium in metals. As discussed in Subsection 5.4.2.3, the tritium diffusion rate in Li_2O is sufficiently high to drive out most of its tritium in a thermal transient. Tritium in metals would respond faster.

Since most of the tritium in metals is vulnerable, the appropriate comparison is simply the relative amount of tritium in the blanket structural material. This refers to the equilibrium tritium values, without any contribution from radiation trapping. If the radiation environment causes trapping of tritium, it is not clear that such trapped-tritium would be vulnerable in the time scales and temperatures of prime interest. Thus, any radiation-trapped tritium is not to be counted.

5.4.2.2.4 Index 1 Results

The tritium inventory calculations have been performed on a consistent basis, as outlined in Subsection 6.6. The calculated partial pressure of tritium in the blanket determines the tritium inventory in the structure. In addition, the contribution of tritium from the plasma in the first wall has been calculated and added to the above contribution. For the reference conditions, it was found that an HT-9 first wall would have tritium inventories of 19 g (tokamak) or 4 g (mirror). The V15Cr5Ti first wall inventory was calculated to be 5.6 (tokamak) or 0.7 (mirror). The tritium parameters for PCA have not been calculated but are assumed for present purposes to be similar to HT-9.

The activation calculations are described in Subsection 6.12. The PHE values are based on the radiological concentration for 5 MW/m^2 wall loading and two years operation.

The results for the structural source term characterization are given in Tables 5.4.2-5 and 5.4.2-6. The scoring scheme was explained in Table 5.4.2-3.

Taking the volatility of elements into account, PCA was found to have 7.7 times more health effects than V15Cr5Ti; HT-9 has 9.0 times more than V15Cr5Ti. The comparison without including volatility shows that PCA is 23.4 times worse than V15Cr5Ti while HT-9 is 7.7 times worse. In either case V15Cr5Ti is clearly the most attractive candidate. The separation from PCA to V15Cr5Ti (factor of 23.4) is less than that calculated in reference 5 (factor of 100) because the present activation calculations indicated about an order of magnitude lower amount of ^{60}Co produced in the steel. In no case did impurities play a significant role, being under 1% of the calculated health effects.

The impact of "low-activation" steels was examined. Use of Modified HT-9 versus HT-9 decreases health effects by 9% when volatility is neglected but increases them by 11% when volatility is included. A BHP comparison was also performed and showed that the Modified HT-9 increases health effects by 15% (without volatility) and by 26% (with volatility). The primary problem is the addition of tungsten, which produces copious amounts of isotopes of tantalum, tungsten, and rhenium, which were assumed to be as volatile as molybdenum.

Use of Tenelon versus PCA as a "low-activation" steel was even worse. The health effects without volatility dropped 15% from PCA to Tenelon but increased 141% when the volatility of elements was included. Again the BHP comparison makes Tenelon look even worse. Tenelon increases BHP

TABLE 5.4.2-5. INDEX SCORES FOR STRUCTURE SOURCE TERM CHARACTERIZATION, INDEX 1, FOR MIRROR BLANKETS

Blanket	Activation-Chemical $(F_{11})^a$	Activation-Physical $(F_{12})^b$	Tritium Inventory (F_{13}) (g)	Index Score
Li/Li/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	1.2	1.00
Li/Li/HT-9	1.89×10^{-2}	3.08×10^{-2}	3.7	0.56
17Li83Pb/17Li83Pb/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	5.7	0.84
17Li83Pb/17Li83Pb/HT-9	1.89×10^{-2}	3.08×10^{-2}	3.9[c]	0.56[c]
Li/He/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	2.7[c]	0.92[c]
Li/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	3.8	0.56
17Li83Pb/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	6.7[c]	0.50[c]
FLIBE/He/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	142.7	0.18
Li_2O/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	4.7	0.54
$LiAlO_2$/He/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	4.5	0.54
$LiAlO_2$/He/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	4.5[c]	0.40[c]
17Li83Pb/NS/HT-9	1.89×10^{-2}	3.08×10^{-2}	6.7[c]	0.50[c]
$LiAlO_2$/NS/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	32.7	0.33
$LiAlO_2$/NS/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	32.7[c]	0.19[c]
$LiAlO_2$/H_2O/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	12.7	0.43
$LiAlO_2$/H_2O/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	12.7[c]	0.29[c]

a. Public health effects per cc of first wall times volatility.

b. Public health effects per cc of first wall.

c. Modification cases for which the tritium inventory was not calculated, values shown are based on simple extrapolations of other blankets.

TABLE 5.4.2-6. INDEX SCORES FOR STRUCTURE SOURCE TERM CHARACTERIZATION, INDEX 1, FOR TOKAMAK BLANKETS

Blanket	Activation-Chemical $(F_{11})^{a}$	Activation-Physical $(F_{12})^{b}$	Tritium Inventory (F_{13}) (g)	Index Score
Li/Li/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	6.1	1.00
Li/Li/HT-9	1.89×10^{-2}	3.08×10^{-2}	18.7[c]	0.52[c]
17Li83Pb/17Li83Pb/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	10.6[c]	0.92[c]
17Li83Pb/17Li83Pb/HT-9	1.89×10^{-2}	3.08×10^{-2}	18.9[c]	0.52[c]
Li/He/V15Cr5Ti	2.11×10^{-3}	4.00×10^{-3}	7.6[c]	0.97[c]
Li/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	18.8	0.52
17Li83Pb/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	22.7[c]	0.49[c]
FLIBE/He/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	207.7	0.18
Li_2O/He/HT-9	1.89×10^{-2}	3.08×10^{-2}	19.8	0.51
$LiAlO_2$/He/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	19.9	0.51
$LiAlO_2$/He/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	19.9[c]	0.37[c]
17Li83Pb/NS/HT-9	1.89×10^{-2}	3.08×10^{-2}	22.7[c]	0.49[c]
$LiAlO_2$/NS/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	60.7	0.35
$LiAlO_2$/NS/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	60.7[c]	0.21[c]
$LiAlO_2$/H_2O/HT-9/Be	1.89×10^{-2}	3.08×10^{-2}	27.7	0.47
$LiAlO_2$/H_2O/PCA/Be	1.62×10^{-2}	9.37×10^{-2}	27.7[c]	0.33[c]

a. Public health effects per cc of first wall times volatility.

b. Public health effects per cc of first wall.

c. Modification cases for which the tritium inventory was not calculated, values shown are based on simple extrapolations of other blankets.

by 59% without volatility or by 174% with volatility. For the manganese-steel, Tenelon, the problem is that the higher amounts of manganese lead to greater hazard from ^{54}Mn and ^{56}Mn than the nickel-related isotopes that are reduced by elimination of nickel.

Overall, use of the "low-activation" steels has a small impact on the accident-related source term, which could be either positive or negative. For relative comparison purposes, substitution of Modified HT-9 and Tenelon for HT-9 and PCA could increase the Index 1 score by 0.00 to 0.12 for HT-9 concepts and decrease the score by 0.00 to 0.04 for PCA concepts. In both cases the range depends on how one would renormalize the figures-of-merit. Because Index 1 is 10% of the total safety score, "low-activation" steels has only about a 1 point (out of 100) impact.

The relative ranking of structures for activation appears fairly certain. Impurity levels are not significant drivers and health effect calculations are fairly well known. In fact, a BHP comparison of V15Cr5Ti, HT-9, and PCA produced an almost identical partial index scoring, 0.50, 0.18, 0.04 for health effects versus 0.50, 0.14, 0.06 for the BHP comparison. The health effect calculations should be made more accurate by obtaining more and better dose factor information, more to determine the absolute level of impact than to change the relative comparison.

The volatility of various elements requires much more research. While improved understanding is not likely to change the relative vanadium to steel comparison, it could significantly influence (a) the absolute level of release, (b) the relative comparison of "low-activation" steels to reference steels, and (c) provide input to alloy development on which elements are most significant to accident-relevant radioactivity.

The tritium inventory in the structure for many concepts was dominated by the first wall inventory, which is caused by tritium implantation. It is generally acknowledged that implantation and the resulting tritium inventory (and tritium flux through the wall, Index 9) are very uncertain. For the reference case, the main impact of the tritium comparison was to hurt the FLIBE concept, which has a very high tritium partial pressure, hence structural inventory. If the implantation inventory in the steels were found to increase, then V-alloy concepts would look better and FLIBE would benefit relative to the other HT-9 blankets. If the implantation inventory in V15Cr5Ti were to increase, then HT-9 concepts would look more attractive. In either case the uncertainty is of order 0.3 for Index 1, which equals 3 points for the total Safety Figure-of-Merit.

5.4.2.3 Breeder/Multiplier Source Term Characterization, Index 2.

The hazard from the breeder and multiplier divides into activation products (and beryllium toxicity) and tritium. Unlike the structural material case where the release pathways and overall behavior are qualitatively similar, the comparison among breeder/multipliers is more conceptually difficult because of the diverse nature of the options. Lithium is a reactive liquid metal; 17Li83Pb is a less reactive liquid metal that highly activates. FLIBE is a fairly inert molten salt that highly activates. Neither of the solid breeders considered, Li_2O and $LiAlO_2$, highly activate. Li_2O is less chemically stable, while $LiAlO_2$ requires toxic beryllium as a neutron multiplier. These wide differences in behavior makes the comparison difficult and, given the major uncertainties involved, makes a clear preference less likely. The following sections indicate the comparison methodology among activation products and tritium, respectively, summarized in Table 5.4.2-7.

5.4.2.3.1 Activation and Chemical Toxicity Hazard

Because there are insufficient data for public health effect or dose calculations for most of the relevant isotopes in the breeders in this study, the Biological Hazard Potential (BHP) was selected as the next best means to compare the hazard inherent in the activation product inventory and beryllium toxicity. The BHP is defined by the inventory of isotopes (INV_i) and the Maximum Permissible Concentration (MPC_i) of each isotope in the air in public areas:

$$BHP = \sum_i INV_i / MPC_i$$

TABLE 5.4.2-7. SCORING SCHEME FOR BREEDER/MULTIPLIER SOURCE TERM CHARACTERIZATION, INDEX 2[a]

Relative Vulnerable Activation and Chemical Toxicity Hazard (50% of Index)

Basis: F_{21} = Biological hazard potential of the isotopes of each element times the relative volatility of that element. The chemical toxicity of beryllium is included.

$$F_{21} = \sum_e (BHP_e \times V_e)$$

Relative Vulnerable Tritium Inventory (50% of Index)

Basis: F_{22} = Total tritium inventory in the breeder times the vulnerable fraction.

Index Score

$$I_2 \text{ (TMR)} = 2.713 - 0.0620 \ln(F_{21}) - 0.0616 \ln(F_{22})$$
$$I_2 \text{ (TOK)} = 2.555 - 0.0574 \ln(F_{21}) - 0.0588 \ln(F_{22})$$

a. Index arbitrarily divided between activation/chemical toxicity and tritium because the possible release pathways and conditions differ too much to know how to combine them.

Because there is a TLV for public exposure for beryllium, 10 ng/m^3-air, it can be directly included in the BHP calculation. It is emphasized that such an approach is not fully satisfactory. First, the basis for the beryllium exposure standard is not explicit and not equal to the MPC isotope basis of 500 mrem/yr from continuous inhalation exposure. Until there is as good a health-effect-to-exposure relationship for beryllium or other chemical toxins as there is for radioisotopes, any comparison between chemical toxicity and radiation exposure will not be fully accurate. For the present time, the inclusion of beryllium with a TLV of 10 ng/m^3 in the BHP is the best available comparison basis. At the end of the study, a new report(68) was published that calculates a risk value for beryllium of 6×10^{-4} risk of cancer death for breathing 1 μg/m^3 for 70 years. If correct, the TLV may be 600 times too high. The potential impact of this is discussed with Index 2 results.

The other aspect of the activation/chemical toxicity comparison is the relative vulnerability. One could take the approach that all the activity present is vulnerable or that any activity in a liquid is vulnerable. However, for public exposure to occur, the material would seem to have to become airborne, either gas, aerosol, or particulate. The activation/chemical toxicity comparison is judged via the BHP for each material, adjusted for the relative ability of each element to become volatilized:

$$F_{21} = \sum_e (\sum_i INV_{ie}/MPC_i)\ V_e = \sum_e BHP_e \cdot V_e$$

where

INV_{ie}	=	the inventory of the i^{th} isotope, of element e
MPC_i	=	the MPC of the i^{th} isotope
V_e	=	relative volatility of the e^{th} element
$\sum_i (INV_{ie}/MPC_i)$	=	BHP of the e^{th} element = BHP_e

As in the case of the structural material comparison, the relative volatilities for breeder and coolant were generally based on their behavior in air. Relative volatility was based on what percent of material exposed to air could be volatilized. The assigned volatilities are listed in Table 5.4.2-8.

For lithium, the prime pathway for mobilization is chemical combustion. The primary activity in the lithium comes from the presence of corrosion products and alkali impurities. For comparison purposes the inventory of corrosion products considered was all of those entrained in the fluid and one-quarter of the deposited material.[5] Fortunately, there are experimental data on the ability of different elements to become airborne during a lithium-air fire.[20,52,53] Generally, about 10% of the lithium spilled becomes volatile, hence alkalis were assigned a relative volatility of 10. The simulated corrosion products were found to be preferentially retained in the lithium pool. The volatilities in Table 5.4.2-8 reflect the experimental findings.

TABLE 5.4.2-8. ASSIGNED RELATIVE VOLATILITIES FOR BREEDERS, MULTIPLIERS, AND COOLANTS

Material	Relative Volatilities[a] (%)
Lithium	Alkalis--10 Mn--1 V, Fe, Co--10^{-1} Ni, Cr--10^{-2} Mo, others--10^{-3}
17Li83Pb	Alkalis--10^{-1} Pb, others--10^{-3}
FLIBE, Li_2O, $LiAlO_2$	Volatile elements--10^{-2} Others--10^{-3}
Beryllium	Be--1.0 Volatile elements--10^{-2} Others--10^{-3}
Helium	Entrained Material--10^{2} Oxidized sputter products--1 Unoxidized sputter products--1 Wall oxide--1
Nitrate Salt	Ar--10^{2} Volatile elements--10^{-1} Others--10^{-2}
Water	Entrained material--10^{2} Wall oxide--50

a. Volatile Elements = Alkalis, Halogens.

For 17Li83Pb, a recent test in air provides volatility information.[54] The maximum lithium aerosol concentration was 10^4 lower than for a lithium-air fire. Since 17Li83Pb is only 0.7 weight percent Li, the fractional release was 10^2 lower than for a lithium-air fire. Thus, the assigned relative volatility for alkali release from 17Li83Pb is 10^{-1}, one-hundredth of the value for alkali release from lithium case. The Pb aerosol concentration was about the same as the Li aerosol, leading to a Pb volatility of 10^{-3}.

FLIBE and the solid breeders will not chemically react with air. FLIBE will not react with water. For these cases, generally volatile elements are assigned a volatility one-tenth that of lithium from 17Li83Pb, i.e. 10^{-2}. Generally nonvolatile elements are assigned a volatility equal to that of lead from 17Li83Pb, noting that lead itself does not react with air.

For the case of helium coolant, all the corrosion/sputter products entrained are already in mobile form, hence they are assigned a volatility of 100. Sputtering is important for the helium case because the corrosion pathway is so small. Recent tests show that the oxide layers on steels in helium are quite adherent even during mild thermal transients.[49,50] This agrees with helium experience in fission reactors where oxide loss is not noted for operating temperatures below 650°C.[55] Thus, assuming the weight gain/loss measurement[49,50] is accurate to about one percent, the assigned volatility for the wall oxide is 1. Because there will be an oxide film, sputter products cannot go directly from the base alloy into the helium, but will lodge in the oxide layer.[5] For those elements that may not oxidize in the low oxygen partial pressure environment, e.g. nickel, the sputtered ion will remain as an unoxidized atom in the oxide layer, releasable to helium only if the oxide layer is lost or the atom is sputtered again, which is unlikely. Hence, unoxidized sputter products are assigned a volatility equal to that of the oxide layer. For elements that may oxidize, e.g. chromium, the sputtering process would basically add to the corrosion rate by adding another transport mechanism away from the base metal. Sputtered, oxidized atoms in the scale are no different from corroded atoms in the scale and are assigned a volatility of 1.0.

A major issue for the Nitrate Salt is the presence of noble gas products, ^{39}Ar and ^{41}Ar. Argon solubility data for this salt has not been found. However it should be quite low. Noble gas solubilities for the LiF-NaF-KF salt have been measured.[56] All are quite low and follow Henry's Law. If the Nitrate Salt were found to have the same behavior as LiF-NaF-KF, then the amount of argon generated in the Nitrate Salt over five years operation would be three orders of magnitude above saturation. Thus, one expects virtually all argon to be removed from the salt at the tritium processing step, down to at least the saturation level with a saturation pressure of slightly over 1 atm of argon. In the event of a spill, it is anticipated that all of the remaining argon would be released from the salt. The decomposition of KNO_3 and $NaNO_3$ has been measured[57] and found to be quite high. At 519°C, the highest temperature reported, the mass loss rates in vacuum of sodium and potassium are 10-20 mg/m^2-s. It is interesting to note that the lithium reaction rate in air is 10 g Li/m^2-s with a corresponding volatilization rate of lithium in a fire of 1000 mg/m^2-s. Hence the sodium and potassium volatilization rate from the salt of 519°C is as high as 1% of the rate for lithium volatilization in a lithium fire. The maximum salt operating temperature is 450°C for the mirror version and 405°C for the tokamak version with 480°C structural/coolant interface temperature. Although no thermal transient calculations have been done for this salt, it is not unreasonable to believe the salt temperature could rise to 519°C; therefore, a volatility of 0.1, one-hundredth of the lithium from Li-air fire case, is assigned.

Other key elements in the salt are carbon and chlorine. After 5 years these activation products could rise to about 10^3 wppm in the salt. Carbon is soluble in the salt at these levels and would be in the form of carbonates.(57,58) Thus, the carbon will not tend to be as mobile as the nitrates themselves; a volatility of 10^{-2} is defined. The chlorine behavior is not known but is arbitrarily assigned a volatility of 0.1, the same as the alkalis.

Beryllium oxidation(59,60) and volatility(59,61) have been studied, but assigning a relative volatility is not straight forward. Assuming vaporization into a vacuum and vapor pressure data from reference 61, the beryllium vaporization rate ranges from 0.6 $\mu g/m^2$-s at 700°C to 15 g/m^2-s at 1284°C, the melting point. Oxidation of the surface will lower these values.(59) In one test, beryllium pieces in air ignited in the 1200-1300°C range, producing "large amounts of airborne beryllium oxide smoke" and temperatures up to 2750°C.(60) For comparison purposes, it is assumed that beryllium could reach 1000-1200°C in a thermal transient, which corresponds to loss rates of 6-400 mg/m^2-s. These values compare to 1.0 g Li/m^2-s for lithium volatility and thus give assigned volatilities of 0.06-4. A single value of $V_e = 1.0$ ($\sim$ 100 g/m^2-s) is used in the comparison, but is uncertain by at least an order of magnitude. Should beryllium ignition occur, volatility would be several orders of magnitude higher.

Corrosion products in water systems are a well-known problem for LWRs. As discussed in Reference 5, a good assumption in terms of reliability is that the outer portion, one-half of total oxide thickness, of the oxide layer could be dislodged in the event of a water blowdown. Hence, the assigned volatility is 50 for the wall oxide. All of the entrained corrosion products would be simultaneously released to the building. The violent flash to steam would cause mobilized corrosion product to be initially airborne.

5.4.2.3.2 Relative Vulnerable Tritium Inventory

The second part of the breeder comparison is the relative amounts of vulnerable tritium that is present. "Vulnerable" is defined as that fraction of the tritium that could be released in airborne form in reasonably-likely chemical or thermal transients. As discussed below, for most of these breeders, it appears that most of the tritium should be considered vulnerable.

For lithium, the possibility of chemical combustion to the point of lithium depletion means that all the tritium in lithium is vulnerable. Such tritium would likely to be initially in the LiOT form. Later conversion to the T_2O form is conceivable as the LiOT deposits and cools.

In the event of a 17Li83Pb spill, the small amount of tritium normally in the alloy would seem to be vulnerable, especially in the event of water or air presence. The T_2O form is more likely in the presence of oxidants.

The small tritium inventory in FLIBE would also likely be vulnerable, as in the 17Li83Pb case, because of the extremely low solubility.

Obviously, the solid breeders are not subject to spills; different scenarios are relevant. For cases where hydrogen gas is not normally added to the purge stream, there appears to be a significant surface inventory of tritium caused by kinetics limitations on release from the surface. Accidental intrusion of various gases could, in principle, strip such tritium from the surface, for example, a moisture leak into the system.

Accidental temperature rise could also do so. Calculations of the behavior of diffusively-held tritium with the TMAP code[62] indicate that the diffusivity of tritium in Li_2O is high enough for the tritium in Li_2O to bake out in the event of a thermal transient, e.g., loss of cooling.[62] The lower diffusivity of tritium in $LiAlO_2$ causes much of the tritium diffusively held in $LiAlO_2$ to not appear vulnerable. Specifically, for the simple transient where cooling stops but purge flow continues, the entire inventory in Li_2O was found to be released in about an hour. The tritium in $LiAlO_2$ is less mobile. In a four hour period, those calculations indicate that about 10% of the inventory would be released.[62] The value of four hours is consistent with the time scale taken to be of interest in the thermal transient comparisons, Index 5.

The above discussions are summarized in Table 5.4.2-9. Given the state-of-knowledge, the only breeder tritium inventory that appears invulnerable is most of the diffusively-held tritium inventory in $LiAlO_2$.

An unresolved issue, which surfaced late in the study, is the tritium inventory in the beryllium multiplier in $LiAlO_2$ and FLIBE concepts, see Section 6.6. Meaningful estimates of the tritium inventory could not be made because of highly uncertain diffusion parameters as well as the possibility of beryllium cracking. Any tritium in beryllium was therefore not included in the evaluation, but remains a future research need.

5.4.2.3.3 Index 2 Results

As for Index 1, the ground rules for the tritium and activation calculations are found in Subsections 6.6 and 6.12. It should be noted that although the activity concentration will fall off as the distance from the first wall, the activation calculations, hence BHP's, are based on material exposed at the first wall. For present purposes, this is not believed to have a significant impact. The figures-of-merit are based on the activity concentration times the volume of material in the blanket. The resulting scores are listed in Tables 5.4.2-10 and 5.4.2-11.

TABLE 5.4.2-9. FRACTIONS OF TRITIUM CONSIDERED VULNERABLE IN VARIOUS BREEDERS

Breeders	Vulnerable Fractions	Comments
Lithium	100%	Assumes oxidants present
17Li83Pb	100%	Low inventory leads to top score anyway
FLIBE	100%	Low inventory leads to top score anyway
Li_2O	100% (solid inventory)	Bakes out in thermal transient; oxidants could strip surface inventory
	100% (purge stream inventory)	
$LiAlO_2$	100% (surface inventory)	Oxidants could strip surface inventory
	10% (diffusion inventory)	Only slowly released in thermal transients
	100% (purge stream inventory)	

TABLE 5.4.2-10. INDEX SCORES FOR BREEDER/MULTIPLIER SOURCE TERM CHARACTERIZATION, INDEX 2, FOR MIRROR BLANKETS

Blanket	Activation-Chemical Toxicity[a]	Vulnerable Tritium (g)	Index Score
Li/Li/V15Cr5Ti	5.3×10^{15}	336	0.11
Li/Li/HT-9	5.3×10^{15}	336	0.11
17Li83Pb/17Li83Pb/V15Cr5Ti	1.4×10^{15}	0.1	0.69
17Li83Pb/17Li83Pb/HT-9	1.4×10^{15}	0.1[b]	0.69[b]
Li/He/V15Cr5Ti	8.2×10^{15}	233[b]	0.10[b]
Li/He/HT-9	8.2×10^{15}	233	0.10
17Li83Pb/He/HT-9	1.8×10^{15}	0.1[b]	0.68[b]
FLIBE/He/HT-9/Be	1.3×10^{16}	0.4	0.46
Li_2O/He/HT-9	9.8×10^{12}	132	0.56
$LiAlO_2$/He/HT-9/Be	2.3×10^{16}	3.4	0.30
$LiAlO_2$/He/PCA/Be	2.3×10^{16}	3.4[b]	0.30[b]
17Li83Pb/NS/HT-9	8.0×10^{14}	0.1[b]	0.73[b]
$LiAlO_2$/NS/HT-9/Be	3.1×10^{16}	30	0.15
$LiAlO_2$/NS/PCA/Be	3.1×10^{16}	30[b]	0.15[b]
$LiAlO_2$/H_2O/HT-9/Be	2.7×10^{16}	150	0.06
$LiAlO_2$/H_2O/PCA/Be	2.7×10^{16}	150[b]	0.06[b]

a. Biological hazard potential of blanket material adjusted by relative volatility of elements involved.

b. Modification cases, breeder tritium inventory estimates were scaled from blankets with the same breeder.

TABLE 5.4.2-11. INDEX SCORES FOR BREEDER/MULTIPLIER SOURCE TERM CHARACTERIZATION, INDEX 2, FOR TOKAMAK BLANKETS

Blanket	Activation-Chemical Toxicity[a]	Vulnerable Tritium (g)	Index Score
Li/Li/V15Cr5Ti	6.0×10^{15}	490	0.10
Li/Li/HT-9	6.0×10^{15}	490[b]	0.10[b]
17Li83Pb/17Li83Pb/V15Cr5Ti	1.6×10^{15}	0.1[b]	0.68[b]
17Li83Pb/17Li83Pb/HT-9	1.6×10^{15}	0.1[b]	0.68[b]
Li/He/V15Cr5Ti	6.3×10^{15}	330[b]	0.12[b]
Li/He/HT-9	6.3×10^{15}	330	0.12
17Li83Pb/He/HT-9	1.4×10^{15}	0.1[b]	0.69[b]
FLIBE/He/HT-9/Be	1.6×10^{16}	0.5	0.46
Li_2O/He/HT-9	6.1×10^{12}	135	0.58
$LiAlO_2$/He/HT-9/Be	3.5×10^{16}	4.8	0.27
$LiAlO_2$/He/PCA/Be	3.5×10^{16}	4.8[b]	0.27[b]
17Li83Pb/NS/HT-9	7.9×10^{14}	0.1[b]	0.72[b]
$LiAlO_2$/NS/HT-9/Be	3.7×10^{16}	209	0.05
$LiAlO_2$/NS/PCA/Be	3.7×10^{16}	209[b]	0.05[b]
$LiAlO_2$/H_2O/HT-9/Be	2.8×10^{16}	230	0.06
$LiAlO_2$/H_2O/PCA/Be	2.8×10^{16}	230[b]	0.06[b]

a. Biological hazard potential of blanket material adjusted by relative volatility of each elements.

b. Modification cases, breeder tritium estimates were scaled from blankets with the same breeder.

The BHP comparison included some surprises resulting from the level of impurities assumed and the relative volatilities. For lithium, the sodium and potassium impurities contribute ^{22}Na, ^{24}Na, and ^{42}K, 96% of the calculated BHP. These were sufficient to overwhelm any contributions from corrosion products. Corrosion rates of 200 kg/yr (LiPb/HT-9), 20 kg/yr (Li/HT-9), 200 g/yr (LiPb/V), and 10 g/yr (Li/V) were used to estimate the amount of structural metal in the liquid metal and any contribution to BHP, or any of the other radiological figures-of-merit used in Indices 10 and 11. At most, corrosion products contribute 1% to the total. This situation could change if the level of impurities were substantially reduced or if corrosion were higher. Lithium is also less attractive since the lithium and presumably sodium and potassium impurities are easily volatilized in the event of combustion. As shown in Table 5.4.2-12, lithium has a lower BHP per m^3 than all other breeders and coolants, other than water and helium. However, it is also seen that the BHP per m^3 for lithium is one of the highest, behind the Nitrate Salt and $LiAlO_2$, once volatility is included.

The table indicates the influence of including volatility in the comparison. One unexpected item was revealed in the activation numbers. Most breeders had sufficient potassium impurity so that ^{39}Ar and ^{41}Ar were present in significant amounts. For the purposes of evaluating accident source term, these argon isotopes were not included because the argon would not likely buildup in each material but would steadily leave the material via normal processing. Thus argon is assumed to be released operationally rather than be retained and potentially released all at once in a transient. This seems a good assumption for fluids, lithium, 17Li83Pb, and Nitrate Salt, but could be quite poor for solid breeders. If so, then the present comparison does not sufficiently penalize solid breeders. Table 5.4.2-12 shows how the BHP is reduced when the argon isotopes are assumed gone. One reason these isotopes are so significant is because they should be 100% releasable, unlike all other elements of interest. Keeping potassium and sodium to low levels in breeder materials would be a significant source term improvement.

TABLE 5.4.2-12. BIOLOGICAL HAZARD POTENTIAL (BHP) PER m^3 OF BREEDER, COOLANT, AND BERYLLIUM[a]

Substance	BHP With Ar Without Volatility	BHP With Ar With Volatility	BHP Without Ar With Volatility	Isotope Contributors[b] Over 10%
Lithium	3.2×10^{12}	6.7×10^{13}	1.8×10^{13}	Na^{22}, Na^{24}
17Li83Pb	2.4×10^{15}	1.3×10^{15}	4.0×10^{12}	Na^{22}, Na^{24} As^{76}, Po^{210}
FLIBE				
No Be toxicity	2.9×10^{13}	2.7×10^{11}	2.7×10^{11}	F^{18}
Low Be toxicity	2.9×10^{13}	2.9×10^{11}	2.9×10^{11}	Be, F^{18},
High Be toxicity	4.6×10^{13}	1.7×10^{13}	1.7×10^{13}	Be
Li_2O	2.1×10^{13}	3.4×10^{14}	3.2×10^{10}	Na^{22}, Na^{24} S^{35}, Mn^{54}
$LiAlO_2$	9.1×10^{15}	6.7×10^{14}	8.9×10^{13}	Na^{24}
Beryllium				
No Be toxicity	8.5×10^{13}	7.4×10^{11}	3.5×10^{11}	Na^{24}
Low Be toxicity	8.5×10^{13}	1.0×10^{12}	6.5×10^{11}	Be, Na^{24}
High Be toxicity	2.7×10^{14}	1.8×10^{14}	1.8×10^{14}	Be
Nitrate Salt	4.1×10^{15}	1.7×10^{17}	2.4×10^{14}	Na^{22}, Na^{24}
Helium				
With PCA	6.3×10^{9}	6.3×10^{9}	6.3×10^{9}	Mn^{54}, Co^{58} Co^{60}
With HT-9	1.7×10^{9}	1.7×10^{9}	1.7×10^{9}	Mn^{54}, Mn^{56}
Water				
With PCA	5.8×10^{10}	2.9×10^{12}	2.9×10^{12}	Mn^{54}, Fe^{55}, Co^{58}, Co^{60}
With HT-9	2.9×10^{10}	1.5×10^{12}	1.5×10^{12}	Mn^{54}, Fe^{55}

a. BHP with and without including Ar^{39} and Ar^{41}; with and without including relative volatility. The evaluation uses BHP without Ar isotopes with volatility effects.

b. Isotopes contributing at least 10% to the BHP without Ar isotopes with volatility effects.

The Nitrate Salt is a special case because sodium and potassium are prime constituents rather than impurities. If one includes the argon isotopes then the BHP (with volatility) measure is two orders of magnitude higher for Nitrate Salt than the next worse material, 17Li83Pb. Although the Nitrate Salt is not penalized in Indices 2 and 3 for the argon isotopes, since they are assumed lost operationally, this salt is strongly penalized in the area of effluents, Index 9.

The tritium part of the breeder/multiplier evaluation is directly based on the calculated breeder inventories. Tritium assumptions are explained in Section 6.6 while calculation details are described in the appropriate design sections. One interesting result is that the $LiAlO_2$/He design operates at a higher temperature than the other $LiAlO_2$ designs and hence has a substantially lower tritium inventory. As mentioned in Sections 6.3 and 6.6, it was believed necessary to add hydrogen to the solid breeder purge stream to prevent a breeder surface inventory.

The use of beryllium entails two special uncertainties. First is the issue of the level of chemical toxicity and how to include it with radiological concerns. Three BHP cases were calculated: (a) without any penalty for beryllium chemical toxicity, (b) with the low value of toxicity that is 600 times lower than established TLV, and (c) with the higher value of toxicity, the established TLV (reference case). With the exception of the FLIBE/Be blanket, the range of cases had only a 0.02 point impact on Index 2, which equals 0.2 points of the total Safety Figure-of-Merit. The low sensitivity is caused by $LiAlO_2$ and beryllium (with high value of chemical toxicity) having similar BHP's. Thus, reducing the beryllium contribution to BHP by $\sim$50%, which varies by four orders of magnitude among breeders, has a small impact. $LiAlO_2$/Be could be significantly improved if both beryllium toxicity were found to be lower <u>and</u> $LiAlO_2$ activation from impurities were lowered by orders of magnitude. FLIBE/Be is an exception (as would be Li_2O/Be) because FLIBE contributes only $\sim$1% to the reference BHP. Either of the alternate beryllium cases mentioned above would improve the FLIBE/Be Index 2 score by 0.32.

The second area of uncertainty is the potential tritium inventory in beryllium, not included in the evaluation. If the tritium inventory in beryllium were found to be significant, beryllium-containing concepts could be penalized in the future.

5.4.2.4 Coolant Source Term Characterization, Index 3.

The coolant comparison is quite similar to the breeder comparison. Again, the comparison is divided into activation products and tritium. The activation products can originate from corrosion (water, lithium, 17Li83Pb, Nitrate Salt), sputtering (helium), or constituents of the coolant itself (17Li83Pb, Nitrate Salt). Because two of the coolants are also breeders, they have a fairly large tritium inventory, which is therefore a relevant basis for comparison. The general scheme for the coolant comparison is indicated in Table 5.4.2-13.

5.4.2.4.1 Activation Hazard

As in the case of the breeder comparison, the coolant activation comparison is judged via the biological hazard potential (BHP) of the

TABLE 5.4.2-13. SCORING SCHEME FOR COOLANT SOURCE TERM CHARACTERIZATION, INDEX 3

Relative Vulnerable Activation Hazard (50% of Index)

Basis: F_{31} = Biological hazard potential of the isotopes of each element times the relative volatility of that element.

$$F_{31} = \sum_e (BHP_e \times V_e)$$

Relative Vulnerable Tritium Inventory (50% of Index)

Basis: F_{32} = Total tritium entrained in the coolant

Index Score

$$I_3(TMR) = 2.909 - 0.0687 \ln(F_{31}) - 0.0587 \ln(F_{32})$$
$$I_3(TOK) = 2.871 - 0.0671 \ln(F_{31}) - 0.0549 \ln(F_{32})$$

activation products, adjusted for their relative volatilities. The relative volatilities for the coolants was listed with the breeders in Table 5.4.2-8.

5.4.2.4.2 Relative Vulnerable Tritium Inventory

The coolant case is similar to the breeder case. For the two liquid metals, all of the tritium inventory is taken to be vulnerable.

The relative ease of tritium removal from the Nitrate Salt is a disadvantage in terms of vulnerability. The normal removal scheme is that the tritium will oxidize to T_2O quickly in the salt and then be drawn off. In the event of a salt spill, the modest pressure (few hundred kPa) of the salt will drop to atmospheric. This can be expected to cause release of the tritium normally entrained in the salt.

The violent blowdown of helium or water would result in all entrained tritium being vulnerable.

Thus, for the coolants in the study, all tritium entrained in coolants should be considered vulnerable.

5.4.2.4.3 Index 3 Results

The calculations for the coolant are completely parallel to those for the breeder/multiplier. The activation figures-of-merit are based on the first wall activity times the volume of coolant within the blanket. Actually, the activity would be diluted by the ratio of coolant in the entire coolant loop divided by the volume within the blanket. However, the total coolant activity is the same in either way of viewing the situation; hence, the comparison is not affected. The index scores are listed in Tables 5.4.2-14 and 5.4.2-15.

Since some of the breeders are also coolants, most of the Index 2 comments are equally valid here. For lithium and 17Li83Pb coolants, impurities are significant in activation. Nitrate Salt activation is

TABLE 5.4.2-14. INDEX SCORES FOR COOLANT SOURCE TERM CHARACTERIZATION, INDEX 3, FOR MIRROR BLANKETS

Blanket	Activation[a]	Tritium Inventory (g)	Index Score
Li/Li/V15Cr5Ti	5.3×10^{15}	336	0.08
Li/Li/HT-9	5.3×10^{15}	336	0.08
17Li83Pb/17Li83Pb/V15Cr5Ti	1.4×10^{15}	0.1	0.65
17Li83Pb/17Li83Pb/HT-9	1.4×10^{15}	0.1[b]	0.65[b]
Li/He/V15Cr5Ti	$\sim8.4\times10^{12}$(?)	Low[b]	1.00[b]
Li/He/HT-9	8.4×10^{12}	Low	1.00
17Li83Pb/He/HT-9	8.4×10^{12}	Low[b]	1.00[b]
FLIBE/He/HT-9/Be	8.4×10^{12}	Low	1.00
Li_2O/He/HT-9	8.4×10^{12}	Low	1.00
$LiAlO_2$/He/HT-9/Be	8.4×10^{12}	Low	1.00
$LiAlO_2$/He/PCA/Be	3.1×10^{13}	Low[b]	0.91[b]
17Li83Pb/NS/HT-9	1.2×10^{16}	500[b]	0.00[b]
$LiAlO_2$/NS/HT-9/Be	1.2×10^{16}	500	0.00
$LiAlO_2$/NS/PCA/Be	1.2×10^{16}	500[b]	0.00[b]
$LiAlO_2$/H_2O/HT-9/Be	7.4×10^{14}	58	0.32
$LiAlO_2$/H_2O/PCA/Be	1.4×10^{15}	58[b]	0.28[b]

a. Biological hazard potential of coolant material adjusted by the relative volatility of each element.

b. Modification cases, coolant tritium inventory estimates were scaled from blankets with the same coolant.

TABLE 5.4.2-15. INDEX SCORES FOR COOLANT SOURCE TERM CHARACTERIZATION, INDEX 3, FOR TOKAMAK BLANKETS

Blanket	Activation[a]	Tritium Inventory (g)	Index Score
Li/Li/V15Cr5Ti	6.0×10^{15}	490	0.09
Li/Li/HT-9	6.0×10^{15}	490[b]	0.09[b]
17Li83Pb/17Li83Pb/V15Cr5Ti	1.6×10^{15}	0.1[b]	0.65[b]
17Li83Pb/17Li83Pb/HT-9	1.6×10^{15}	0.1[b]	0.65[b]
Li/He/V15Cr5Ti	$\sim 8.4 \times 10^{12}$(?)	Low[b]	1.00[b]
Li/He/HT-9	8.4×10^{12}	Low	1.00
17Li83Pb/He/HT-9	8.4×10^{12}	Low[b]	1.00[b]
FLIBE/He/HT-9/Be	8.4×10^{12}	Low	1.00
Li_2O/He/HT-9	8.4×10^{12}	Low	1.00
$LiAlO_2$/He/HT-9/Be	8.4×10^{12}	Low	1.00
$LiAlO_2$/He/PCA/Be	3.1×10^{13}	Low[b]	0.91[b]
17Li83Pb/NS/HT-9	1.5×10^{16}	900[b]	0.00[b]
$LiAlO_2$/NS/HT-9/Be	1.5×10^{16}	900	0.00
$LiAlO_2$/NS/PCA/Be	1.5×10^{16}	900[b]	0.00[b]
$LiAlO_2$/H_2O/HT-9/Be	7.4×10^{14}	50	0.36
$LiAlO_2$/H_2O/PCA/Be	1.4×10^{15}	50[b]	0.31[b]

a. Biological hazard potential of coolant material adjusted by the relative volatility of each element.

b. Modification cases, coolant tritium inventory estimates were scaled from blankets with the same coolant.

controlled by its sodium and potassium. Water activation levels are determined by corrosion; the values in this study are based on the detailed water corrosion examination in References 2 and 5. Corrosion and impurities in helium are so low that only sputtering remains as a significant source of activation products in a helium coolant, the values used in the study are based on those of Bickford.(55)

As one expects, helium is the best coolant in the source term comparison, nil tritium and lowest level of activation products. Nitrate Salt is the worst, having the highest level of tritium and of activation products. Because water and helium activation levels are determined by structural-material-relevant processes, corrosion and sputtering, the score for those coolants is determined by which structural material is used. The modification case of Li/He/V15Cr5Ti would likely be better in this respect than Li/He/HT-9, but no calculations have been done.

5.4.2.5 <u>Results</u>.

The source term characterization scores are summarized in Tables 5.4.2-16 and 5.4.2-17. The relative comparison does not significantly differ from mirror concepts to among tokamak concepts.

It is interesting to divide all the materials by what process is controlling the level of activation relevant for accident analysis. In the first category are those materials dominated by isotopes caused by primary constituents: V15Cr5Ti, HT-9, PCA, Nitrate Salt, FLIBE, and beryllium (chemical toxicity). It was found that the "low-activation" steels have a small influence in accident-relevant activation, which may be either positive (likely for Modified HT-9) or negative (likely for Tenelon). Among the structural materials in this study, only V15Cr5Ti is a low activation material with a several point (∿4 out of 30 in source term) advantage over HT-9.

In the second category are those materials where activation relevant for accidents is dominated by impurities, lithium, 17Li83Pb, Li_2O, $LiAlO_2$, and beryllium (nonchemical). 17Li83Pb actually would belong in

TABLE 5.4.2-16. SUMMARY OF SOURCE TERM CHARACTERIZATION SCORES, INDICES 1-3, FOR MIRROR BLANKETS

Blanket	Index 1 Structure	Index 2 Breeder/ Multiplier	Index 3 Coolant	Total Source[a] Term
Li/Li/V15Cr5Ti	1.00	0.11	0.08	11.9
Li/Li/HT-9	0.56	0.11	0.08	7.5
17Li83Pb/17Li83Pb/V15Cr5Ti	0.84	0.69	0.65	21.8
17Li83Pb/17Li83Pb/HT-9	0.56	0.69	0.65	19.0[b]
Li/He/V15Cr5Ti	0.92	0.10	1.00	20.2[b]
Li/He/HT-9	0.56	0.10	1.00	16.6
17Li83Pb/He/HT-9	0.50	0.68	1.00	21.8
FLIBE/He/HT-9/Be	0.18	0.46	1.00	16.4
Li_2O/He/HT-9	0.54	0.56	1.00	21.0
$LiAlO_2$/He/HT-9/Be	0.54	0.30	1.00	18.4
$LiAlO_2$/He/PCA/Be	0.40	0.30	0.91	16.1[b]
17Li83Pb/NS/HT-9	0.50	0.73	0.00	12.3[b]
$LiAlO_2$/NS/HT-9/Be	0.33	0.15	0.00	4.8
$LiAlO_2$/NS/PCA/Be	0.19	0.15	0.00	3.4[b]
$LiAlO_2$/H_2O/HT-9/Be	0.43	0.06	0.32	8.1
$LiAlO_2$/H_2O/HT-9/Be	0.29	0.06	0.28	6.3[b]

a. Total Source Term = $\sum_{i=i}^{3} I_i W_i$, ranges from 0 to 30.

b. Modification cases for which tritium inventories were not calculated, values used were roughly scaled from the most similar blanket for which tritium inventories were available.

TABLE 5.4.2-17. SUMMARY OF SOURCE TERM CHARACTERIZATION SCORES, INDICES 1-3, FOR TOKAMAK BLANKETS

Blanket	Index 1 Structure	Index 2 Breeder/ Multiplier	Index 3 Coolant	Total Source[a] Term
Li/Li/V15Cr5Ti	1.00	0.10	0.09	11.9
Li/Li/HT-9	0.52	0.10	0.09	7.1[b]
17Li83Pb/17Li83Pb/V15Cr5Ti	0.92	0.68	0.65	22.5[b]
17Li83Pb/17Li83Pb/HT-9	0.52	0.68	0.65	18.5[b]
Li/He/V15Cr5Ti	0.97	0.12	1.00	20.9[b]
Li/He/HT-9	0.52	0.12	1.00	16.4
17Li83Pb/He/HT-9	0.49	0.69	1.00	21.8[b]
FLIBE/He/HT-9/Be	0.18	0.46	1.00	16.4
Li_2O/He/HT-9	0.51	0.58	1.00	20.9
$LiAlO_2$/He/HT-9/Be	0.51	0.27	1.00	17.8
$LiAlO_2$/He/PCA/Be	0.37	0.27	0.91	15.5[b]
17Li83Pb/NS/HT-9	0.49	0.72	0.00	12.1[b]
$LiAlO_2$/NS/HT-9/Be	0.35	0.05	0.00	4.0
$LiAlO_2$/NS/PCA/Be	0.21	0.05	0.00	2.6[b]
$LiAlO_2$/H_2O/HT-9/Be	0.47	0.06	0.36	8.9
$LiAlO_2$/H_2O/PCA/Be	0.33	0.06	0.31	7.0[b]

a. Total Source Term = $\sum_{i=i}^{3} I_i W_i$, ranges from 0 to 30.

b. Modification cases for which tritium inventories were not calculated, values used were roughly scaled from the most similar blanket for which tritium inventories were available.

the first category except for the fact that lead-related isotopes appear far less volatile than some caused by impurities. For the breeders, the key impurities are generally sodium and potassium. Significant improvements for fusion could be made by having cleaner lithium and Li_2O. Impurity reduction in 17Li83Pb will help only to a point because of the lead activation. Lithium and 17Li83Pb improvement is limited eventually by corrosion products. $LiAlO_2$ and beryllium improvement is limited by beryllium chemical toxicity. In fact, the breeder BHP for the $LiAlO_2$/Be is about two-thirds from $LiAlO_2$ and one-third from beryllium. Hence, the concept could be significantly improved only if both $LiAlO_2$ impurities were reduced and beryllium toxicity were found to be lower than currently assumed in setting Threshold Limit Values for industrial practice.

In the third category is the material dominated by corrosion products, water. It is interesting to note (here and under maintenance, Index 10) that PWR maintenance exposure is quite significant and water is the second-best coolant from the activation standpoint. Helium has nil activation problems, while lithium, 17Li83Pb, and Nitrate Salt appear worse than water.

In the fourth category is the material whose activation is dominated by sputtering, helium. Actually, sputtering is present at similar levels in any design with any coolant. However, for all other coolants than helium, some other process appears to contribute greater amounts of activation products than sputtering. In other words, sputtering products form the lower limit of coolant contamination, and only helium approaches the lower limit.

The rationale and discussion of tritium calculations is detailed in Section 6.6. Among breeders and coolants, lithium generally has the highest inventory, and FLIBE, the lowest. Later, under the effluent discussion, Index 9, the situation is reversed. The relatively high tritium solubility in lithium hurts here in the source term evaluation but makes tritium control far easier, helping in the effluent evaluation.

Several areas of source term uncertainties and sensitivities should be briefly mentioned. In the following discussion, the sensitivity will be put in terms of the total Safety Figure-of-Merit (SFM) on the 0-100 scale. Thus, a sensitivity of 1 point equals 1% of the total SFM. Note that because almost all of the components of the SFM involve parameters varying by orders of magnitude, a 1% SFM change does not mean the blanket design is 1% safer. In absolute terms a 1% SFM change could well be significant. Given the uncertainties in a relative comparison, a 1% SFM change is only moderately significant.

Use of "low-activation" steels was found to be a 1 point impact. Low impurity lithium compounds could be a several point improvement, the key impurities being sodium and potassium. Beryllium chemical toxicity is only a 0.2 point impact except for FLIBE/Be where it is a 3.2 impact. Late in the study some minor changes were made in the structural fractions in the $LiAlO_2$/NS and Li/He/HT-9 designs. These are not expected to significantly alter the results here.

The tritium parameters are more uncertain than activation. They appear to be uncertain by several SFM points, although the relative comparison is less uncertain. The most uncertain and sensitive tritium issues are the first wall implantation-relevant inventory, the Nitrate Salt tritium inventory, and the importance of tritium versus activation in the source term. In the first case, implantation inventory is poorly understood. If the inventory of one structural material were found to vary by orders of magnitude relative to others, up to 5 SFM points could be changed. However, a change in implantation understanding that did not change the relative inventories would not change the comparison results.

The Nitrate Salt tritium inventory is quite high (500 g, 900 g) and could be lowered. An order of magnitude improvement could increase its score by about 1.5 points with about 2 points additional improvement possible if its structural inventory fell by a corresponding amount, giving a total of 3.5.

The final issue is the relative importance between tritium and activation in the source term part of the evaluation. The reference case is a 50-50% split. Since many of the tritium inventories appear low, a lower weighting for tritium might be appropriate. To examine the potential change of scores if source term tritium were lower weighted, the limiting case of devoting none of the source term score to tritium was examined, see Table 5.4.2-18. It is seen that the impact is larger than any of the other uncertainties examined for source term characterization. Overall, the Li/He and Li/Li concepts would most benefit from devoting more weight to activation, +2.6, +1.5 points respectively. Most hurt would be LiPb/LiPb (-5.0) and $LiAlO_2$/He (-4.4). The basic reason is that lithium has the highest tritium inventory among breeders and would be the most helped by ignoring tritium in the evaluation; the opposite is true for LiPB/LiPb and $LiAlO_2$/He.

TABLE 5.4.2-18. CHANGE IN TOTAL SOURCE TERM CHARACTERIZATION SCORE IF TRITIUM WERE IGNORED FOR SELECTED MIRROR BLANKETS

Blanket	Reference Score	Tritium Ignored	Change
Li/Li/V	11.9	13.4	+1.5
Li/Li/HT-9	7.5	7.0	-0.5
LiPb/LiPb/V	21.8	16.8	-5.0
Li/He/HT-9	16.6	15.2	-1.4
FLIBE/He/HT-9/Be	16.4	14.6	-1.8
Li_2O/He/HT-9	21.0	23.6	+2.6
$LiAlO_2$/He/HT-9/Be	18.4	14.0	-4.4
$LiAlO_2$/NS/HT-9/Be	4.8	3.6	-1.2
$LiAlO_2$/H_2O/HT-9/Be	8.1	7.6	-0.5

5.4.3 Fault Tolerance

Based on current understanding of fusion reactors and early comparative risk assessments, several specific blanket-relevant potential accidents appear to be significant contributors to total accident risk, hence important enough for consideration in design and concept evaluation. The second part of the safety evaluation, fault tolerance, attempts to gauge how tolerant the various blanket designs are to various reference potential accidents. Only those accidents relevant to blanket concept choice were considered in this study.

The overall philosophy is that blankets should be inherently safe, depending only on passive features to survive transients. The basic intent of the fault tolerance evaluation is to measure how well each design does in avoiding the need for active protection systems. To the extent that a design deviates from a perfect fault tolerance score, increased active protection systems appear to be required.

5.4.3.1 General Approach

The purpose of the fault tolerance evaluation is to measure the relative risk of the blanket concepts because of specific potential fault conditions. Where the probability of the transient varies among blankets, e.g., coolant-tube break, both the probability and consequence parts of the risk are considered. Where the probability of the transient is blanket-independent, e.g., seismic events, only the consequence parts of the risk are considered.

The fault tolerance part of the evaluation is subdivided into five indices. Ideally, a probabilistic treatment would determine how much each transient contributes to total risk. Information and manpower resources in this study precluded that level of detail; rather, the potential transients were logically grouped into five categories and given equal weighting.

For each fault tolerant index, there are one or more potential transients. Appropriate design philosophy and goals corresponding to each were identified early in the study and were listed in the BCSS interim report.[1] The design teams therefore had goals to work toward. Through the design process there has been interaction between designers and safety, helping to lead to the final design version. Failure to meet one of these goals does not, in itself, constitute a fatal flaw for a design. However, in each case it necessitated an evaluation of whether a concept inherently had a fatal problem. These cases are described later on a case-by-case basis.

For each index, the response of the various blankets to the specific transient was compared. Those blankets clearly meeting the corresponding design goals were given the top index score of 1.0. For cases where there is no absolute criterion, the blanket(s) best meeting the identified design philosophy were scored 1.0. In like fashion, blankets that inherently did not meet the goal were scored 0.0. Remaining blankets were scored relative to the best.

The basis for the blanket-to-blanket comparisons was a combination of experimental data, analytical analysis, and engineering judgement. Because BCSS resources did not allow any transient calculations, heavy use was made of past studies, calculations, and engineering judgement.

The following subsections sequentially go through the five fault tolerant indices. For each index, the relevant transients are first mentioned along with the corresponding design philosophy or goals. Next, the evaluation scheme is detailed, explaining the relevant figures-of-merit and other bases for comparing the blankets. Then, supporting evidence and analysis for each blanket performance is given. Finally, the information is assembled and that index evaluation is summarized.

In most cases, there was no difference between the evaluation issues among tokamak blankets and the evaluation among mirror blankets. For these cases, no distinction between them is made in this presentation. In the areas of fault tolerance to external forces and to near-blanket system interaction (indices 6 and 7), the differences between the limiter and plasma disruptions in a tokamak and a choke coil and rapid plasma loss in a mirror causes differences in the evaluation.

As seen in the individual subsections to follow, the coolant choice appears most important to fault tolerance. The coolant choice determines the basic thermal-hydraulic and geometric aspects of the design, which are key determinants to how the design would respond to several transient conditions. Because detailed stress analysis and crack propagation studies were not available, some potential differences among structural material choices may have been missed. However, the afterheat, physical properties, general chemical behavior, and influence on design parameters as functions of structural material choices were considered.

5.4.3.2 Fault Tolerance to Breeder-Coolant Mixing, Index 4

One of the possible types of primary coolant losses is into the rest of the blanket, i.e. the breeder or multiplier zone. The water and salt-cooled designs use coolant tubes separating coolant and breeder. The helium-cooled designs instead have the breeder material placed in tubes or plates. However, for the liquid-metal self-cooled designs, there is no such separation. Except for the self-cooled designs, the surface area and number of pieces in the reactor separating the coolant and breeder are large, and failure is somewhat likely, hence the motivation for this index.

5.4.3.2.1 Design Philosophy

The design philosophy is that the first wall and blanket should be capable of withstanding complete severing of a coolant tube or breeder container, without (a) propagation of failure to other components, modules, or structures or (b) loss of radioactivity from the module and primary coolant system in excess of that normally entrained in blanket fluids. No new radioactivity should be mobilized. Relevant issues include possible coolant-breeder chemical reactions, resulting pressures and temperatures, mitigation potential, and the probability of occurrence.

5.4.3.2.2 Evaluation Scheme

The choice of structural material was not relevant in this index primarily because no distinction in the study was made among the crack propagation resistance of the various materials. Known differences among structural materials like afterheat, activation inventory, oxidation/chemical behavior, and thermophysical properties are not relevant to this index.

The evaluation scheme is built on two sets of utility points, for probability and for occurrence, of this transient. Blankets were given utility points from 0 to 2 for the relative probability, corresponding to the scheme in Table 5.4.3-1. A 0 indicates that the transient cannot exist for this concept. A 1 indicates the probability is very low because of the use of double wall piping. A 2 indicates the probability is modestly high; a single barrier separates the coolant and breeder.

Blankets were also given utility points based on the consequences of the transient, Table 5.4.3-1. A 0 indicates no transient exists for this

TABLE 5.4.3-1. SCORING SCHEME FOR FAULT TOLERANCE TO BREEDER-COOLANT MIXING, INDEX 4

<u>Relative Probability (0 to 2 utility points)</u>

0 - No probability of occurrence, breeder and coolant are the same fluid system.

1 - Low probability of occurrence, double wall separates breeder and coolant.

2 - Moderate probability of occurrence, single wall separates breeder and coolant; transient likely during reactor lifetime.

<u>Relative Consequences (0 to 3 utility points)</u>

0 - Design goal met, no transient exists

1 - Design goal met, continued operation possible.

2 - Design goal met, but requires reactor shutdown; potentially requiring changeout and corresponding occupational exposure and waste generation.

3 - Design goal not met, module fails.

<u>Relative Risk (0 to 6 utility points)</u>

Relative Risk = Relative Probability x Relative Consequences.

design. A 1 indicates the module completely meets the design philosophy, even to the point of allowing some continued operation in the event of failure. A 2 was given to those designs where the design philosophy appears to be met, but reactor shutdown appears necessary. Such a shutdown could lead to a sector changeout and corresponding occupational exposure and waste generation. A concept clearly failing the design philosophy receives a 3 and is a candidate for having a potential fatal flaw.

5.4.3.2.3 Liquid-Metal Cooled Concepts

The liquid-metal cooled designs score the best possible: 0 on relative probability and 0 on relative consequence. As MHD understanding and modeling improves, the effect of a channel-to-channel wall tearing should be examined. Any perturbation in the blanket geometry could influence current-flow paths, which could influence MHD forces and liquid metal flows.

5.4.3.2.4 Helium-Cooled Concepts

The helium-cooled designs all have a single-wall separating coolant and breeder, meriting a relative failure probability of 2. Because the pods are initially at full helium pressure, they will survive failure of a breeder plate or tube wall. A breeder wall failure would lead to helium coolant (5.0 MPa) entering the breeder zone, slightly increasing its pressure (4.9 MPa initially). While this would not be a desirable operating condition, limited operation would be possible, so helium-cooled blankets score a 1 on relative consequence.

The designs were also checked to see if they tolerated a loss of breeder zone pressure, which would increase the pressure gradient between coolant and breeder. The solid breeder plates should withstand this crushing force. The liquid breeder tubes would have to withstand a buckling force. As indicated in the helium design sections, the tubes were found to have about a 20% margin in avoiding buckling.

Beryllium use entails three general chemical reaction questions. First is the question of beryllium in contact with impure helium coolant or

$LiAlO_2$. At sufficiently high temperature, oxygen will cause oxidation of beryllium with resulting swelling. At lower temperatures, the oxide layer on the beryllium will protect the interior from this oxidation process. The temperature at which extensive swelling occurs, "breakaway swelling," is 850°C, see Chapter 6. The beryllium temperature in the $LiAlO_2$/He/Be concept, where beryllium rods are in contact with helium coolant, is at or below 550°C, far below the "breakaway swelling" point. The same is true for the FLIBE/He/Be concept, where beryllium balls are in contact with the helium coolant. The other beryllium designs, $LiAlO_2$/NS/Be and $LiAlO_2/H_2O$/Be, do not have beryllium in contact with the coolant. Thus, beryllium-helium compatibility should be acceptable from the safety viewpoint in all designs.

It should be noted that the latter two designs use beryllium in contact with $LiAlO_2$ at temperatures as high as 1000°C. It is not known if the combination is compatible, but this seems plausible and is assumed in the study. An appropriate R&D item would be further experiments and analysis on beryllium compatibility with helium impurities and relevant solid breeders, including behavior during abnormal thermal transients.

The second beryllium question is contact with FLIBE. Normally in the FLIBE/He/Be design the two are not in contact. A breeder tube break would not cause much FLIBE to contact the beryllium because the FLIBE is at lower pressure than the helium coolant. However, even if FLIBE-Be contact were to occur there should be no problem since the FLIBE is kept in the reducing state by addition of excess beryllium directly to the salt.

For the above reasons, present helium-cooled designs do not appear to have any chemical reaction problems among coolant-breeder-multiplier.

The third issue is the beryllium-air reaction. As mentioned in the breeder source term characterization, subsection 5.4.2.3, beryllium can ignite in air, especially when in powder form. However, high temperatures, >1200°C, may be required. Because carbon dioxide was used for all beryllium-containing designs as the building atmosphere, no penalty was associated with beryllium-air reactions. This is a similar position to not directly penalizing lithium-air reactions, a more serious issue. If carbon dioxide reacts with beryllium or if air is used, one should consider beryllium reactivity. The most worrisome case would be when beryllium is used in powder form ($LiAlO_2$/NS/Be, $LiAlO_2/H_2O$/Be) rather than as bulk pieces (FLIBE/He/Be, $LiAlO_2$/He/Be).

5.4.3.2.5 Salt-Cooled Concepts

The Nitrate Salt-cooled designs also have a single wall, meriting a relative failure probability of 2. The issue is the chemical compatibility of NS-$LiAlO_2$, NS-Be, and NS-17Li83Pb. In the first case, no chemical reaction is expected thermodynamically between Nitrate Salt and $LiAlO_2$. However, the high temperature, up to 1000°C, in the solid breeder zone would cause rapid decomposition of the salt. Experimental studies with these salts to date [57,58] are not sufficient to determine how the salt would behave. The assumption used here is that some decomposition would occur with some volatile products introduced into the breeder zone and possibly some pressure increase.

The second case is NS-Be. Qualitative scoping tests at 500°C have been conducted with wood, graphite, and beryllium exposed to the salt,[64] as follows:

> "In the case of wood, a sliver of fir weighing ∿10 mg was attached to the end of a length of platinum wire and lowered to the surface of the molten salt. The wood immediately burst into flames and was completely consumed."
>
> "Three experiments were conducted with graphite. To test for an immediate reaction a 100 mg piece of graphite was immersed in the molten salt. No reaction was evident over a period of ∿5 minutes. The graphite was then washed with water to remove residual salt and reweighed. There was a gain in weight of about 2 mg. The same piece was then dropped into the molten salt and left there for four hours. When checked the piece had disappeared either by reaction with the salt or by break-up into small particles. There was a layer of dark particles on the bottom of the crucible. In the final graphite experiment two rods 4.6 mm in diameter were tested at the same time in the same crucible."
>
> "One of the rods was 23 mm in length and was completely immersed in the molten salt at >500°C for 1 1/2 hours. The initial

weight of the rod was 625 mg and the weight after testing was 632 mg. The second rod was about 100 mm in length. It was suspended so half its length was submerged in the salt and half in the atmosphere. The submerged portion of this rod looked much the same as the shorter rod, however the portion exposed to the atmosphere and particularly the area at the salt/atmosphere interface had changed diameter significantly. The weight of the rod before testing was 2.7444g after testing was 3.224g."

"A special stainless steel fixture was built to clamp the ends of a beryllium wire securely. The dimensions of the wire were ∿0.36 mm in diameter by 37.3 mm long and it weighted 6.7 mg. A cutter was built into the assembly so the wire could be broken at its center to provide an oxide free beryllium surface within the molten salt bath. The wire was exposed to ∿540°C molten salt for 60 hours then the wire was broken remotely. No reaction was noted. The weight of the wire after testing was unchanged at 6.7 mg. The cutting operation stretched the wire approximately 1 mm."

"In summary, wood reacted immediately and violently with the molten salt. Graphite exhibited no fast reaction, but a possible slow, longer term reaction. Beryllium showed no reaction whatsoever with the molten salt."

It should be noted that temperatures in a blanket would be higher (up to 1000°C) than the 500°C test condition and that the surface area for reaction would be substantially higher. Thus, blanket conditions could lead to more substantial reactions. Given the above information, the $LiAlO_2$/NS/Be blanket is judged to require shutdown in the event of salt tube breakage, likely requiring changeout and corresponding occupational exposure and waste generation. Therefore the blanket receives a consequence score of 2. Additional information could change this to a 1 or 3.

A 17Li83Pb-Nitrate Salt scoping testing has been performed.[63] Gram-scale quantities of NS and 17Li83Pb have been reacted at 450°C. In one test 17Li83Pb was added to excess NS and temperatures rose only 20°C. In the second test NS was added to excess 17Li83Pb and temperatures rose 68°. In both cases the reaction products included Li_2O and PbO. Both lithium and lead participate in the reaction. Gas bubbles were observed although the reaction was mild. Sparks and glowing were observed in the second test where excess 17Li83Pb was used.

Thermodynamically, a 1 to 1 mixture of Li from 17Li83Pb with $NaNO_3$ should produce a temperature rise of about 100°C if all heat were confined to the reaction products. However, if a 5 to 1 mixture of 17Li83Pb with $NaNO_3$ is assumed, thermodynamics predict a 1100°C temperature rise. The difference is the ability of excess Li atoms to further reduce NO_3 to NO_2 to Li_2O. The experimental finding agrees that the 17Li83Pb rich reaction is the more serious case. Although the experimentally observed temperature rises for gram-scale quantities were not serious, the fact that the reactions went to completion and the high thermodynamic temperature rise for 17Li83Pb-rich reactions do not promote optimism. Based on the existing information, the 17Li83Pb/NS blanket is judged to fail the design goal because of potentially substantial chemical reaction. Therefore, the blanket receives a consequence score of 3. Additional information could change this to a 1 or 2.

5.4.3.2.6 Water-Cooled Concepts

The water-cooled designs use double-wall tubing, which should substantially lower the probability versus a single-wall design; they are given a 1 on relative probability. A single-wall design, like STARFIRE, could experience failures on the order of once per year.[66] Although use of a double-walled design reduces the probability of failure, it does not reduce the probability to zero because (a) potential common mode failures exist, and (b) the design concept calls for continued operation if only the inner tube or outer tube fails. Indeed, there is no provision to detect only an inner or outer tube failure, since the intent is to maximize availability by continuing operation unless breeder-coolant contact occurs as the result of failure of both the inner and outer tube for a single double-wall-tube assembly. The outer and inner tube are each designed to withstand full coolant pressure so that an inner tube failure of one tube does not automatically lead to failure of the outer tube. The pod geometry is designed to survive accidental pressurization by the water; thus, it meets the design goal. However, continued operation is not possible, leading to a 2 on relative consequences. Water would enter and highly pressurize the tritium system, normally at ~0.1 to 0.6 MPa pressure.

Also, it would possibly slowly dissolve the $LiAlO_2$.[67] A Li_2O/H_2O concept would be worse since the Li_2O would more actively react with water.

5.4.3.2.7 Index 4 Results

The resulting index scores for the concepts are listed in Table 5.4.3-2. The self-cooled designs have no risk in this area and get a perfect index score of 1.0. Only 17Li83Pb/NS appears to fail the design goal. The remaining designs have intermediate scores based on the above discussion of single versus double barrier (probability) and the possibility for continued operation. It should be noted that this potential transient greatly influenced the water-cooled design. A single-wall, flat box water-cooled design, like STARFIRE, was initially considered in this study. Analysis[66] of the STARFIRE design indicated that the module would fail if a coolant tube broke. The present water-cooled design avoided failing the design goal by adoption of double-wall tubing and a pod geometry.

Future work relevant to this transient should concentrate on four issues. First, researchers should acquire and use additional probability failure rate information. Second, transient codes that examine those transients should be verified by experiment, particularly in the area of pressure drop and flow through solid breeder material. Third, crack propagation studies should be conducted on the reference fusion alloys, particularly in the irradiated state. Fourth, additional understanding of relevant chemical reactions is needed.

5.4.3.3 Fault Tolerance to Cooling Transients, Index 5

Several cooling transients are possible--LOCA (loss of primary cooling, large break), LOFA (loss of coolant flow), and LOSP (complete loss of site power). All blankets require cooling during operation. Past studies have shown, e.g. References 2-3, 5-6, 69-70, that typical blankets will only survive several seconds without cooling if the plasma stays on.

TABLE 5.4.3-2. INDEX SCORES FOR FAULT TOLERANCE TO COOLANT-BREEDER MIXING, INDEX 4

	Utility Points			
Blanket[a]	Relative Probability	Relative Consequence	Relative Risk[b]	Index Score[c]
Li/Li	0	0	0	1.00
17Li83Pb/17Li83Pb	0	0	0	1.00
Li/He	2	1	2	0.67
17Li83Pb/He	2	1	2	0.67
FLIBE/He	2	1	2	0.67
Li_2O/He	2	1	2	0.67
$LiAlO_2$/He/Be	2	1	2	0.67
17Li83Pb/NS/Be	2	3	6	0.00
$LiAlO_2$/NS/Be	2	2	4	0.33
$LiAlO_2$/H_2O/Be	1	2	2	0.67

a. Within the limits of the analysis in this study, the selection of the structural material was not relevant for this index.

b. Relative risk = relative consequence times relative probability.

c. Index score = 1.0 - (relative risk points/6).

This appears true for both tokamak and mirror blankets, although relatively few mirror cases have been studied. The only likely exceptions are materials with very high melting points and high thermal conductivity like TZM[5], which are not being considered in this study. The differences in the available time to action between mirror and tokamak are not particularly relevant[70] and thus the basic plasma shutdown need is blanket independent--unimportant to this study. It is therefore assumed in this study that either a mirror or tokamak plasma can be safely shutdown on the order of a few seconds following a blanket cooling transient. However, it should be emphasized that such plasma shutdown systems are only at a very early stage in reactor design. Commercial reactors, perhaps even long-pulse (~minutes) test devices of mirrors and tokamaks, will require such systems.

Given the plasma shutdown assumption, the blanket response to these transients is a strong function of choice of structural material, coolant, breeder, and basic geometry. The structural material is particularly relevant since it is generally the dominant source of afterheat. The breeder is of interest because of its influence on thermal-hydraulics, sometimes (a) serving as a heat transfer medium, (b) serving as a heat sink, (c) being an additional source of afterheat, or (d) influencing heat transfer from structure to other blankets or other ultimate heat sinks.

5.4.3.3.1 Design Philosophy

The reference LOCA is assumed to be a loss of primary coolant from at least one entire blanket sector. Where there are dual coolant loops, this applies to only one loop. The design philosophy is that blanket modules should be capable of withstanding a major loss of coolant accident without (a) significant structural failure or (b) loss of radioactivity from the module and primary coolant system in excess of that normally entrained in blanket fluids.

The reference LOFA is assumed to result in primary coolant flow stoppage within at least one blanket sector, most likely due to pump stoppage or flow blockage. Where there are dual coolant loops, this applies to only one loop. The design philosophy is the same as that of the LOCA. Unlike the reference LOSP case, auxiliary systems like purge flow and breeder circulation are assumed to be working.

The reference LOSP, loss of site power, is assumed to cause complete loss of electricity at the site, i.e. a station blackout. The design philosophy is that the first wall and blanket should be capable of withstanding complete loss of site power for four hours without failure of components or structures. The motivation for four hours is that the power recovery occurs for current plants within four hours for 77% of LOSP events.[71] A concept requiring auxiliary power generators is less satisfactory than one that does not need backup power. The common theme in this index is that it is desired that blanket concept designs be inherently safe with respect to cooling transients, i.e. be able to survive severe transients with only passive design features, not active engineered safety systems.

5.4.3.3.2 Evaluation Scheme

The probability of these transients is assumed independent of blanket choice. Since a large LOCA would occur outside of the blanket area, the amount of piping involved, hence the failure probability, is only weakly dependent on coolant choice. Interestingly, the data base for pipe failures do not show a dependence on pressure. Rather, pipe failure data[73-75] per length of pipe for water, helium, and sodium vary by only an order of magnitude, helium being the best. Perhaps this reflects similar quality goals. In any case, given the existing pipe failure data and the similarities in pipe-lengths external to the blanket, it is not possible to differentiate among large-break LOCA probabilities for the various concepts. The possible initiators to large LOCA where coolant choice is relevant, e.g. seismic events, are covered separately in Index 7.

Likewise, pump redundancy (a LOFA probability factor) could be used equally well, with associated costs, for any coolant. Flow blockage (part of the LOFA probability) may be coolant dependent, but cannot be assessed here. The primary concern is that corrosion products in the liquid metal systems could deposit and block flow. Corrosion-related penalities are addressed in other indices relating to the associated activation concerns. The chance of loss-of-site-power is blanket independent.

Based on the above, the probability of these transients appears fairly blanket-independent; the relevant cooling-transient comparisons are associated only with consequence. Relevant measures include the allowable time to action before damage or radioactivity release, mitigation schemes, and afterheat levels. For most types of blanket concepts, there are adequate transient codes to calculate the actual blanket response to these transients. For example, the ATHENA code[72] can handle helium and water and will shortly have lithium capability, initially for flow in constant magnetic fields. The main uncertainties in this area are the behavior and associated modeling of liquid metal or molten salts in a magnetic field, either due to fluid flow variations or magnetic field variations or both. Another type of uncertainty is that associated with lack of property data, especially for 17Li83Pb.

Extensive use was made of past studies in the evaluation. Because past results for somewhat different geometries are not quantifiably transferable to this study and its exact designs, quantified blanket responses to the reference transients were not obtainable. Furthermore, the focus of the study is less on exact detailed designs and more on inherent characteristics of the various concepts. Also, designers had inadequate time to fully consider transient behavior in their reference BCSS designs. For all these reasons, the intent for this index was to evaluate whether each blanket design could meet the stated design goals, based on past studies and engineering judgment. For each transient, blankets with a high chance of meeting the design goals in their basic current configuration were given two points. Blankets that appeared capable of meeting the goals (a) with some redesign, (b) by removing a key uncertainty, or (c) are border-line cases were given one point. Blankets that appeared unable to meet the design goals without active protection systems were given zero points. Since there did not seem to be any cases where active emergency systems could not be used to solve these problems, no blanket concept was deemed to have a fatal flaw. The scoring scheme is summarized in Table 5.4.3-3.

5.4.3.3.3 Liquid-Metal Cooled Concepts

Overall the liquid-metal cooled designs appear fairly tolerant to cooling transients.[5, 6, 69, 70] The low afterheat in the Li/Li/V concept allows it to survive several hours (LOCA) or even days (LOSP, LOFA), based on past results.[5] 17Li83Pb has a volumetric specific heat 1.5 times lower than lithium so a 17Li83Pb concept would be somewhat worse than lithium but still meet the design goal. The afterheat in 17Li83Pb does not appear to be a significant driver.[70] Neither concept depends on natural circulation, which is unlikely if the magnets are on, but at least the Li/Li concept should be designed for flow up through the blanket in order to get any benefit possible, especially if magnets are off. The Li/Li/V and LiPb/LiPb/V blankets get perfect scores for this index.

TABLE 5.4.3-3 SCORING SCHEME FOR FAULT TOLERANCE TO COOLING TRANSIENTS, INDEX 5

<u>LOCA--Loss of primary cooling, or LOFA--Loss of primary coolant flow (0 to 2 utility points for each transient)</u>

2 - Blanket meets design goal.

1 - Blanket has the potential to meet the design goal, likely with some redesign.

0 - Blanket does not meet design goal, requires emergency cooling.

<u>LOSP--Loss of site power (0 to 2 utility points)</u>

2 - Blanket meets design goal of four-hour survival without active means.

1 - Blanket has the potential to meet design goal, especially with some redesign.

0 - Blanket does not meet design goal, requires emergency backup power.

The HT-9 concepts are somewhat poorer. Using a 300°C temperature increase to define the time to failure, the HT-9 concepts still survive at least four hours for the LOFA and LOSP cases.[5, 6, 70] The LiPb/HT-9 concept would experience such a temperature rise in a LOCA after 3 hours;[70] similarly a different Li/316SS concept lasts only 2 hours.[5] Therefore the Li/Li/HT-9 and LiPb/LiPb/HT-9 designs are seen as borderline cases in the event of LOCA, but they clearly pass the LOSP and LOFA design goals. An auxiliary volume associated with the surge tank could provide enough additional liquid metal to achieve the design goal.

It should be mentioned that the time scale for survivability for LOCA and LOFA is approximately 4 hours, as for LOSP. However, it has deliberately been left somewhat vague. One reason is that almost all of the calculations to date do not model the details concerning heat being rejected to an ultimate heat sink, which is the building at shorter times, at longer times, outside the building. Thus at longer times (several hours), the calculations become more uncertain. Therefore, the primary intent is for the design to appear to survive several hours without special active systems.

5.4.3.3.4 Helium-Cooled Concepts

The behavior of the helium-cooled designs appears to be a strong function of the breeder material because of the breeder's role as heat sink, heat removal medium, and heat transfer agent. Basically one expects to find that He/liquid-metals have more tolerance than He/solids, and this is in fact the case.

For LOCA transients, a key advantage of helium is that it is single-phase. Thus, as the coolant loop depressurizes, no pump cavitation takes place. Instead there is a steady loss of cooling capacity as the pressure drops. Since continued use of depressurized cooling was found to be adequate for higher afterheat fission-fusion hydride blankets,[76] it is expected to work well for all of the helium-cooled blankets in this study.

For LOFA and LOSP transients, a key disadvantage of helium is that it is gaseous and thus has a very low specific heat capacity. Based on the He/Li/steel cases in Reference 5, the He/Li and He/LiPb designs should still survive a LOFA or LOSP, the liquid metals being good heat sinks. This is .lso in agreement with the finding that the Li/Li and LiPb/LiPb designs can meet the LOFA and LOSP design goals.

The He/solid cases do not appear capable of meeting the LOFA and LOSP design goals. A Li_2O/He/Inconel design experienced first wall melt in 2 1/4 hours with a 1.2 MW/m^2 wall load.[77] Even though a HT-9 version would have less afterheat than Inconel, the Reference 5 MW/m^2 wall loading in this study makes it probable that the Li_2O/He and $LiAlO_2$/He designs would fail in about an hour. Another calculation[5] puts first wall failure at about one hour for $LiAlO_2$/He. No credit was given for purge stream circulation during a LOFA.

The FLIBE/He design appears to be in-between the liquid metal and solid breeder cases, but no calculations have been done. Fortunately, the activation of flourine adds only a modest amount of afterheat: 0.53% of operating power at shutdown and 0.022% of operating power at 5 minutes.[78] On the negative side, its heat conduction is poor, about 1 W/m°K, even slightly lower than the solid breeders. On the positive side, one should get some cooling from the continued circulation of flibe, either pumped (LOFA) or natural (LOSP). Since FLIBE/He transient calculations have not been done, one can only say that it is a borderline case for LOFA and LOSP.

One possible solution for LOFA and LOSP transients is natural circulation, which appears to work for small size HTGR's.[79] However, an HTGR has a much easier geometry for heat transfer by natural circulation than does a fusion blanket. In the absence of any calculations, it does not appear promising to depend on natural circulation for a helium-cooled blanket.

5.4.3.3.5 Salt-Cooled Concepts

No cooling transient calculations have been done for any Nitrate Salt designs; however, one can estimate their behavior from other coolant cases. Similar to the water-cooled designs, the salt-cooled designs depend on use of dual coolant loops to protect against LOFA and LOCA. Thus even if one cooling loop fails, another is still operating and capable of adequate removal of afterheat, perhaps indefinitely. Dual cooling loops may work for the salt and water cases because the manifolding restrictions are lax enough to provide two sets of cooling lines to each blanket module whereas this appears excessively difficult for helium or liquid-metal cooled designs.

Rough examination of density change with temperature indicate sufficient buoyancy to drive natural circulation. A rough estimate of the required coolant velocity and pressure drop for afterheat removal is compared with an estimate of the pressure head available due to temperature (density) differences in the coolant. Since the available head greatly exceeds the required head, it appears that natural convection cooling is feasible even though the analysis is approximate.

Afterheat in the coolant itself reaches an equilibrium value of $\sim 0.25\ W/cm^3$. This will account for more than half of the afterheat in the blanket and coolant loop, since the most highly activated parts of the structure should produce roughly $0.1\ W/cm^3$ and the entire coolant inventory will generate heat at $0.25\ W/cm^3$. During reactor operation the heat removed in the coolant is $\sim 170\ W/cm^3$. Near the first wall, the coolant velocity is 5 m/s during operation. Thus, to maintain the 75°C coolant ΔT, a coolant velocity of 0.7 cm/s or ~ 1 cm/s would be required.

The Nitrate Salt Reynold's number is less than 100 in the blanket coolant tubes at 1 cm/s and well within the laminar region throughout the loop. The resulting pressure drop in the blanket coolant tubes is ~ 40 Pa. Velocities in the coolant loop external to the blanket are lower than in the coolant tubes near the first wall but turns, entrances, etc., and much longer flow paths cause most of the coolant loop pressure drop to occur outside the blanket. Based on the above estimates of afterheat and the pressure drop in the blanket coolant tubes, a total loop pressure drop of ∿100 to 200 Pa is expected to be adequate to remove the afterheat at a coolant ΔT of 75 to 100°C.

Sandia data[57, 58] indicates that a temperature change of 100°C results in a density change of 70 kg/m^3 in the draw salt. Thus a simple loop oriented vertically with one vertical leg 100°C hotter than the other would have a potential pressure difference of ∿700 Pa times the height of the vertical legs, in meters. Since the coolant generates most of the heat in the system, the ideal loop would have heat exchangers continuously along the down leg.

According to this estimate, a ten meter high loop with 100°C ΔT would have an available pressure difference of 7000 Pa. This compares rather well with the 100-200 Pa pressure difference required to remove the afterheat. Several deficiencies in this analysis are apparent. However, since the estimated available pressure exceeds the required pressure by more than an order of magnitude it is believed that the removal of afterheat by natural convection in Nitrate Salt blankets is feasible.

For all three transient cases with this salt, a prime concern is thermal decomposition at elevated temperatures, potentially leading to a severe pressure-increase in the loop and/or degraded cooling ability as the salt's composition changes. For evaluation purposes it is <u>assumed</u> that the above cooling methods keep the salt sufficiently cool to avoid these problems. If decomposition is rapid at low (∿600°C) temperatures, then not only would the salt concept score lower in Index 5, but it might have a fatal flaw. It would not be acceptable if a mild temperature increase caused rapid pressure buildup in the coolant. Data presented in Reference 57 suggests that this is indeed the case.

5.4.3.3.6 <u>Water-Cooled Concepts</u>

Dual coolant loops appear to be acceptable solutions to LOCA and LOFA transients for water-cooled designs. However, dual loops do not help if site power fails (LOSP). A possible solution is natural circulation. Analysis[72] of the STARFIRE[2] design indicated that natural circulation was not established and the design experienced multiplier melting at about 70 minutes after shutdown. PWR work on "feed and bleed"[80] is relevant. This technique refers to adding more water to the system and bleeding steam from it to remove heat. The amount of water required is large. At 15 MPa, about 6×10^5 kg ($\sim$800 m^3) of water would boil to remove 1% of operating power (40 MW_{th}) for four hours. Since the coolant loop holds about 500 m^3, then excess, passive storage of 200-300 m^3 of water would be needed. To avoid the need for an active high-pressure injection pump, that volume of water would need to be stored above whatever pressure one would try to operate in "feed and bleed" mode. Normal operating pressure is 15 MPa. If one trys to operate "feed and bleed" above 15 MPa, then (a) the water would need to be stored at very high pressure, (b) the design would have to withstand the higher pressure, and (c) a mass-energy balance must be sustainable. Based on Reference 80, it appears one cannot operate such a mass-energy balance above 15 MPa, rather a lower pressure is used. If one trys to operate "feed and bleed" below 15 MPa, there is no passive arrangement to perform the "bleed" part. That is, any passive valving to allow steam loss below 15 MPa would also attempt to work during normal operation. Normally the technique calls for operator-controlled values, which is an active system. Finally, it should be mentioned that cooling fails if steam pockets develop somewhere within the blanket. For these reasons, the water-designs appear to fail the design goal for LOSP protection.

5.4.3.3.7 Index 5 Results

The results for the cooling transient evaluations are listed in Table 5.4.3-4. In most cases, the design appears to meet the design approach. The poor thermal characteristics of the solid breeders made those cases more difficult. Overall, the designers depend strongly on inherent features of the design, especially natural circulation and/or using dual coolant loops.

For liquid-metal cooled designs, the toughest cooling transient to protect against is the LOCA transient since it involves loss of the main

TABLE 5.4.3-4. INDEX SCORES FOR FAULT TOLERANCE TO COOLING TRANSIENTS, INDEX 5

	Utility Points[b]			
Blanket	LOCA	LOFA	LOSP	Index[a] Score
Li/Li/V15Cr5Ti	2	2	2	1.00
Li/Li/HT-9	1	2	2	0.83
17Li83Pb/17Li83Pb/V15Cr5Ti	2	2	2	1.00
17Li83Pb/17Li83Pb/HT-9	1	2	2	0.83
Li/He/V15Cr5Ti	2	2	2	1.00
Li/He/HT-9	2	2	2	1.00
17Li83Pb/He/HT-9	2	2	2	1.00
FLIBE/He/HT-9/Be	2	1	1	0.67
Li_2O/He/HT-9	2	0	0	0.33
$LiAlO_2$/He/HT-9/Be	2	0	0	0.33
$LiAlO_2$/He/PCA/Be	2	0	0	0.33
$LiAlO_2$/NS/HT-9/Be	2	2	2	1.00
$LiAlO_2$/NS/PCA/Be	2	2	2	1.00
17Li83Pb/NS/HT-9/Be	2	2	2	1.00
$LiAlO_2$/H_2O/HT-9/Be	2	2	0	0.67
$LiAlO_2$/H_2O/PCA/Be	2	2	0	0.67

a. Index score = one-sixth times the sum of the individual LOCA, LOFA, and LOSP utility points.

b. See Table 5.4.3-3 for explanation of scoring scheme.

heat sink. For all other coolants a breeder would remain in the event of a LOCA so that something may serve as a heat sink other than the afterheat-producing structural material.

For helium-cooled designs, LOCA's are relatively easy to protect against. Instead, LOSP's and LOFA's are the toughest problems. It appears that the liquid-metal breeders designs can cope with these transients but that solid breeders cannot, without the help of an active protection system.

For the salt and water-cooled designs the use of dual, parallel coolant loops is taken to provide adequate protection from LOCA and LOFA transients. The toughest problem appears to be LOSP. The key approach would seem to be natural circulation, which may be viable for the salt but may not be credible for water because of its two-phase, pressurized nature.

In the general area of cooling transients, several R&D-type issues are relevant. First, as is stated elsewhere, magnetic effects on liquid metal flow is a key R&D item. However, it is vital that those studies include transient fluid flow and transient magnetic field cases. Likewise, transient code capability for these cases is required. Second, the behavior of Nitrate Salt during a transient involving thermal transients needs further study. Excessive decomposition could be a fatal flaw. Third, modeling and some experiments are needed to better determine the ability for natural circulation to cool the various blankets. Experiments are needed in the molten salt cases to study buoyancy and thermal stability and in the liquid metal cases to study the hindrance from the magnetic field. Fourth, the thermal emissivity of structural material after operation is important in loss-of-coolant cases. The emissivity of these cases is poorly known and should be studied using representative metal surface conditions: plasma-side walls (clean, low emissivity), liquid metal-corroded walls, and helium impurity-oxidized walls.

The use of "low-activation" steels could have an influence if the afterheat levels for the first several hours differed appreciably from the reference steels. Specific calculations have not been done, but it is known that manganese, e.g. Tenelon, leads to several times higher afterheat levels at shutdown than iron. Tungsten, e.g. modified HT-9, may pose a similar problem. Thus, "low-activation" steels are not expected to help in cooling transients, but rather to hurt. It is possible that a "low-activation" steel version blanket would not meet the cooling transient design goals even though the reference version would. Thus the impact of "low-activation" steels could be 0 to 6 SFM points negative.

Use of a low-activation alloy like V-alloy does help, as indicated in the index scores. Use of a low-activation ceramic like SiC would totally solve afterheat problems (and most other radiological problems).

5.4.3.4 *Fault Tolerance to External Forces, Index 6*

Several externally-caused forces may accidentally occur. For BCSS purposes, these are plasma disruption (tokamak), rapid loss of plasma (mirror), off-normal magnetic fields due to magnet disturbances, and seismic events. The probability of these initiators occurring is blanket independent, unless first wall choice somehow influences the frequency of plasma disturbances. Therefore, only the relative consequences due to the transients are relevant for the comparison in this study.

5.4.3.4.1 Design Philosophy

The overall design philosophy is that blanket structures should be capable of withstanding external forces without (a) significant structural failure or (b) loss of radioactivity from the module and primary coolant system. Additional specifics are indicated in the following sections.

5.4.3.4.2 Evaluation Scheme

The index is equally divided into thermal, electromagnetic, and mechanical coupling of the external forces with the blankets. This type of separation directly highlights the material and geometrical properties relevant to the various external forces. The only external force directly giving a thermal effect is a plasma disruption. The eddy currents in blankets caused by electromagnetic transients can only cause modest temperature increases.[5] Both plasma disturbances and coil-related transients give electromagnetic forces. Finally, seismic events cause a direct mechanical impact to the blankets.

The evaluation scheme is summarized in Table 5.4.3-5, more detailed discussions are given in the following subsections. Basically, 0 to 2 utility points are assigned for each concept for each of the three types of external coupling-thermal, electromagnetic, and mechanical. The primary thermal coupling in a tokamak would come from a major plasma disruption. The relevant figure of merit is the relative thermal stress generated in the first wall normalized by the structural material's yield strength. Rapid mirror plasma losses do not appear to have a significant thermal influence on blankets so the issue of thermal coupling does not appear relevant for mirror blankets.

TABLE 5.4.3-5 SCORING SCHEME FOR FAULT TOLERANCE TO EXTERNAL FORCES, INDEX 6

<u>Thermal Impact of Plasma Distribance (2 utility points)</u>

Mirror - Transient causes nil thermal impact, all blankets get 2 points

Tokamak - Blankets get 0 to 2 points on basis of relative values for the figure of merit:[a] $\dfrac{E\alpha}{k(1-\nu)\sigma_y}$

<u>Electromagnetic Impact of B-field Disturbance, plasma or coil induced (2 utility points)</u>

Mirror - All blankets get 2 points

Tokamak - All blankets get 0 points

<u>Mechanical Impact of Seismic Disturbances (2 utility points)</u>

Blankets get 0 to 2 points on basis of relative values of figure of merit:[b] M_s

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a. $E\alpha/k(1-\nu)\sigma_y$ is the relative thermal stress divided by the yield stress, see text.

b. M_s is the mass of a blanket sector.

The electromagnetic fields in the vicinity of the blanket can be directly perturbed from either a plasma disturbance or magnet coil transient calculations. However, significant variances among blankets were not found.

A seismic event would cause direct mechanical shaking of the blanket. In lieu of detailed dynamic seismic force calculations, the relevant figure-of-merit is the total blanket mass.

5.4.3.4.3 Mirror Plasma Loss

The reference plasma dump has a time scale of 100 msec. The plasma would escape to the end cell, with nil thermal impact on the blanket. The magnetic impact is potentially important. Following a rapid plasma dump, a 10 mm first wall will experience a pressure of about 0.04 MPa in the direction toward the plasma. The pressure in a one meter thick 17Li83Pb blanket is predicted to be 0.3 MPa. The pressure for a blanket of 10% steel is predicted to be 0.1 MPa. All of these results are pressures small in comparison to operating conditions. The stresses corresponding to these pressures are of order 8 MPa. Further, since the EM pressure loading as well as the blanket geometry are radially symmetric, there is no net load transferred to the support of the blanket.

It is important to realize that a rapid plasma loss for mirrors is not a likely event. They are hypothetical, not having been observed in mirror experiments. Because such a transient does not have a direct thermal impact to the blanket, mirror blankets should not and are not relatively ranked in the thermal-coupling area. All blankets get the best score of 2 utility points for thermal-coupling. Likewise, it appears that the direct magnetic coupling is minor. All blankets get the best score of 2 utility points for magnetic-coupling.

Basically, a hypothetical rapid plasma loss appears to pose a small risk, low probability and low consequences. The issue should not be forgotten since the predicted pressures are not totally insignificant. Future reactor designers should at least consider the problem. For the above reasons, this issue does not influence the relative comparison. Future work is needed to verify that any variance among blankets, potentially influenced by the presence of liquid-metals, is not sufficient to make this transient a significant risk.

5.4.3.4.4 Tokamak Plasma Disruption

Inadequate examination of the impact of tokamak plasma disruptions on the various blankets has been performed in this study. For comparison purposes the impact is divided into thermal and magnetic. The state-of-knowledge indicates that the risk from both thermal or magnetic effects is significant. In lieu of detailed thermal effects calculations, the most appropriate basis to judge the relative thermal effects is a thermal stress figure of merit: $E\alpha/k(1-\nu)\sigma_y$, where these physical properties of the first wall structural material are relevant: Young's modulus (E), thermal expansion coefficient (α), thermal conductivity (k), poisson's ratio (ν), and yield stress (σ_y). The result is HT-9 is midway between PCA and V15Cr5Ti, which vary by a factor of three. Actually, several slightly different figures of merit are relevant,[5,81] but the rank ordering is always V15Cr5Ti(best), HT-9, and PCA. The 2 thermal-coupling utility points are assigned on this basis. Coolant and breeder choice is not relevant.

The magnetic impact appears significant. The stress analysis of both the liquid-metal-cooled and the helium-cooled blankets are based on a static pressure of the order of 0.6 MPa on the first wall directed towards the plasma. This pressure causes stresses in the blankets in two ways. First, the pressure causes additional bending of the first wall and the blanket over and above that due to coolant pressure. Typically this causes additional bending stress of 50 MPa in the liquid-metal-cooled-designs and a maximum of 152 MPa at the base of the lobe for the helium-cooled designs. When added to the stresses due to the coolant pressure, the total stresses are within the allowable limits for both designs. Secondly, this pressure causes additional stresses due to the global bending of the blanket and these stresses are critically dependent on how the blanket is supported. Based on one support configuration (not necessarily the optimum configuration), the maximum additional stress in the liquid metal cooled blanket occurs at the first wall near the inboard support and equals -169 MPa. When added to the stresses due to coolant pressure, dead weight of the coolant and structure and seismic loading, the total stress is still within the allowable bending stress limit. The torque due to the EM loading causes a shear stress of only 14 MPa in the first wall. This analysis also shows that a radial thrust of 11 MN per sector has to be supported by the inboard wall support. A parallel analysis for the helium cooled designs has not been carried out. However, since the pressure loadings due to the disruption and the first wall areas are similar, a thrust of similar magnitude will also come on the supports of the helium cooled designs. This thrust is not negligible and will require careful design of particularly the inboard wall supports where available space is limited in both designs. It should be remembered that all disruption stress analyses to date have been based on a static pressure whereas in reality the pressures are applied and removed dynamically in tens of ms. The resultant dynamic stresses could be higher or lower than those computed so far dependent on the mass, stiffness and damping of the blankets. However, based on the stress analyses conducted to date there is no basis to conclude that the liquid-metal-cooled blankets are more critical than the helium-cooled designs as far as EM loading because of disruptions are concerned.

Given the level of analysis to date, variance of magnetic-coupling risk among blanket concepts has not been discovered. Since this problem appears quite significant, all blankets were given 0 utility points in this area.

5.4.3.4.5 Coil-Related Magnetic Transients

Plasma disturbances are not the only potential source of magnetic-coupling to a blanket and associated structures. Failures of the various magnets may provide another source of magnetic disturbance. No analysis of any such coupling was performed for this study, but it is felt that any impact would not be highly blanket dependent and is not included in the evaluation.

Another issue is the possibility of any induced stresses on ferromagnetic HT-9 piping either during normal or off-normal conditions. Although such forces do exist, based on previous work [82] they appear to be quite manageable and are not included in the evaluation.

5.4.3.4.6 Seismic Events

Like the case of a magnetic field, a seismic disturbance has the potential to couple with several parts of the reactor and plant simultaneously. A preliminary examination of the level of the problem started with the STARFIRE[2] reference operational-basis earthquake with ground shaking of 0.13 g horizontal and 0.09 g vertical. A conservative estimate to determine blanket survivability is to then apply a 4.4 g acceleration in all three directions simultaneously. At least one blanket, 17Li83Pb/17Li83Pb/V, does not meet this conservative criterion, indicating that a problem might exist. A detailed dynamic structural analysis would be needed to verify that heavier designs could withstand seismic loading. Such an analysis is beyond the scope of this study.

Since it appears that a problem exists, it is relevant to grade blankets in this area. The chosen figure of merit is simply the total blanket mass.

5.4.3.4.7 Index 6 Results

Details of EM interactions and stress analysis can be found in Chapter 6. Based on the above discussions, the individual utility points for each blanket for thermal, electromagnetic, and seismic coupling have been determined and are summarized in Tables 5.4.3-6 and 5.4.3-7 along with the corresponding Index 6 scores.

5.4.3.5 Fault Tolerance to Near-Blanket System Interactions, Index 7

The blanket does not exist by itself; it is surrounded by several other components and potentially interacting systems. These include the reflector (if separate from the blanket), shield, near-plasma components,

TABLE 5.4.3-6. INDEX SCORES FOR FAULT TOLERANCE TO EXTERNAL FORCES, INDEX 6, FOR MIRROR BLANKETS

Blanket	Utility Points[a] Thermal	Electromagnetic	Seismic	Index[b] Score
Li/Li/V15Cr5Ti	2	2	1.51	0.92
Li/Li/HT-9	2	2	1.48	0.91
17Li83Pb/17Li83Pb/V15Cr5Ti	2	2	0.42	0.74
17Li83Pb/17Li83Pb/HT-9	2	2	0.38	0.73
Li/He/V15Cr5Ti	2	2	1.67	0.95
Li/He/HT-9	2	2	1.60	0.93
17Li83Pb/He/HT-9	2	2	0.00	0.67
FLIBE/He/HT-9/Be	2	2	1.66	0.94
Li_2O/He/HT-9	2	2	1.93	0.99
$LiAlO_2$/He/HT-9/Be	2	2	2.00	1.00
$LiAlO_2$/He/PCA/Be	2	2	2.00	1.00
17Li83Pb/NS/HT-9	2	2	1.00	0.83
$LiAlO_2$/NS/HT-9/Be	2	2	1.98	1.00
$LiAlO_2$/NS/PCA/Be	2	2	1.98	1.00
$LiAlO_2$/H_2O/HT-9/Be	2	2	1.81	0.97
$LiAlO_2$/H_2O/PCA/Be	2	2	1.81	0.97

a. See Table 5.4.3-5 for explanations of scoring scheme.

b. Index score = sum of utility points divided by six.

TABLE 5.4.3-7. INDEX SCORES FOR FAULT TOLERANCE TO EXTERNAL FORCES, INDEX 6, FOR TOKAMAK BLANKETS

Blanket	Utility Points[a] Thermal	Electromagnetic	Seismic	Index Score[b]
Li/Li/V15Cr5Ti	2	0	1.56	0.59
Li/Li/HT-9	1	0	1.51	0.42
17Li83Pb/17Li83Pb/V15Cr5Ti	2	0	0.05	0.34
17Li83Pb/17Li83Pb/HT-9	1	0	0.00	0.17
Li/He/V15Cr5Ti	2	0	1.75	0.63
Li/He/HT-9	1	0	1.67	0.45
17Li83Pb/He/HT-9	1	0	0.21	0.20
FLIBE/He/HT-9	1	0	1.77	0.46
Li_2O/He/HT-9	1	0	1.93	0.49
$LiAlO_2$/He/HT-9/Be	1	0	2.00	0.50
$LiAlO_2$/He/PCA/Be	0	0	2.00	0.33
17Li83Pb/NS/HT-9	1	0	1.00	0.33
$LiAlO_2$/NS/HT-9/Be	1	0	2.00	0.50
$LiAlO_2$/NS/PCA/Be	0	0	2.00	0.33
$LiAlO_2$/H_2O/HT-9/Be	1	0	1.87	0.48
$LiAlO_2$/H_2O/PCA/Be	0	0	1.87	0.31

a. See Table 5.4.3-5 for explanations of scoring scheme.

b. Index score = sum of utility points divided by six.

primary heat exchangers/steam generator, magnets outside the shield, and the reactor building. The following subsections, 5.4.3.5.1-6, describe these and limits the scope of the evaluation. subsections 5.4.3.5.7-8 indicate the design goals and evaluation scheme. Results appear in subsection 5.4.3.5.9.

5.4.3.5.1 Potential Interactions

The number of possible interactions between the blanket and surrounding systems is large, too large for all to be fully considered in this study. There are three reasons for this. First, the allowable manpower to analyze the interactions is limited. Second, many of the issues and mechanisms involved are not understood. Third, the BCSS is focused on blanket choice comparisons. These near-blanket system interactions are only relevant to the extent that it appears that there is a considerable difference in how blankets might interact with other components. This is particularly of interest in the cases where the potential interaction carries sufficient associated risks that the particular blanket choice forces some major change in the near-blanket systems. The approach for the BCSS has been to screen the possible interactions and then to focus the evaluation on only the most critical near-blanket systems. The following subsections contain brief discussions of the possible interactions.

5.4.3.5.2 Reflector and Shield

Early in the BCSS, some of the blankets had a separate reflector. The final designs, however, have evolved to where none of the concept designs uses a reflector with a coolant different from the blanket coolant. Hence there are no reflector-blanket interactions.

The shield is a low pressure, low temperature system, but several interactions may be of interest. First, its potential role as a heat-sink in the event of blanket cooling loss is sometimes significant, but this has already been covered in Index 5. Since the shield normally removes a few percent of the total heat load, it would only represent a very modest additional heat source on the blanket/reflector should shield cooling loss occur. The probability for a shield cooling loss is likely to be blanket-independent and not relevant to this study.

Second, if a shield were to fail, it could interact with the blanket via shield-coolant/blanket-material chemical interactions. The worrisome cases are when water is the shield coolant with reactive metals, lithium or vanadium, in the blanket. Recent work [46] indicates the V15Cr5Ti oxidation is rapid for temperatures over 650°C coupled with oxygen partial pressures over 10 Pa. This oxidation causes formation of a eutectic V_2O_5-Cr_2O_3 oxide that is molten above ∿655°C and causes significant vanadium volatility. Thus, such a pressure-temperature combination could lead to both V15Cr5Ti wall failure and radioactivity mobilization. The MARS and STARFIRE shields have water at under 100°C and about 100 kPa pressure. It is possible that a shield water leak could lead to oxidizing potential over 10 Pa in the vacuum chamber. The uncertainty is whether the cold water would cool the first wall below 650°C, initially at or above 750°C, sufficiently fast. The chance of a oxidation transient occurring is likely controlled by the exact details of a given blanket-shield arrangement as well as the time history of the leak. Another complication is the issue of how the plasma would behave as steam enters. In a mirror, a loss of plasma would not further heat the first wall, so no special problem exists. In a tokamak, a steam-triggered disruption could further heat the first wall with associated increased oxidation of the wall and/or could directly fail the wall.

If the first wall fails either from wall oxidation (if V15Cr5Ti) or from an induced plasma disruption (tokamak), then the steam may interact with the rest of the blanket. In the case of the self-cooled blankets, first wall failure causes lithium or 17Li83Pb to mix with water. In the case of Li/He or 17Li83Pb/He, a second failure of the breeder tube is necessary before water-liquid metal reaction could occur.

Third, shield failure could cause pressurization of the plasma chamber. Depending on the exact boundary of the plasma chamber vacuum boundary, the location of the break, and volume and type of shield coolant, such a transient may pressurize and fail the vacuum chamber. Table 5.4.3-8 summarizes the possible reflector and shield interactions with the blanket.

TABLE 5.4.3-8. REFLECTOR AND SHIELD INTERACTIONS WITH THE BLANKET

Blanket Type	Reflector/Shield	Potential Interactions	Comment
All	Reflector is integral with blanket	None	None
All	Water-cooled shield	Shield as heat source	Not evaluated, minor problem
		Shield as heat sink	Evaluated for blanket cooling transients, Index 5
		Chemical reactions	Concern for Li, V15Cr5Ti blankets
		Plasma disruption triggered, first wall fails	Concern for tokamak
		Vacuum chamber pressurization	Concern
		Building pressurization	Evaluated in Index 8

5.4.3.5.3 Near-Plasma Components/Mirror

Of all the components near the blanket, it appears that the most important interactions are those between the blanket and near-plasma components. The higher importance stems from several factors. First, ignoring any design restrictions on these components imposed by blankets, these components are more difficult to design. Thus, it is more important to know if blanket choices further restrict the limiter or choke coil design than it is to know if the shield would need to be redesigned. For example, nonwater-cooled shields are certainly feasible, but nonwater-cooled limiters are more problematical. Second, the mere proximity of the limiter and choke coil to the plasma and to the blanket raises the probability of interactions among the three. For example, a leak from a limiter could likely lead to a plasma disruption, thereby subjecting the blanket to the limiter coolant and the plasma disruption.

Several interactions appear plausible between blanket and near-plasma components in a mirror, summarized in Table 5.4.3-9. Relevant components include the restive part of the choke coil and its shielding, the direct converter, and the halo scraper. The first issue is the possibility of thermal shock to the first wall from an impinging jet of water. For a leak of the direct converter or halo scraper, there is only about a 0.001% chance that a line-of-sight from the leaked jet to part of the first wall would occur. In addition the water would have to travel tens of meters. As indicated later, thermal shock is only possible if some condensed water, rather than steam, strikes the wall. Thus there is a nil probability that a leak of the direct converter or halo scraper could thermal shock a blanket first wall. For the case of the choke coil and its shielding, the solid angle subtended by the first wall is larger so that there is a 5% chance that a line-of-sight from a leaked jet to part of the first wall would occur. The water would have to travel a distance of meters. It appears to be low probability that such a leak could thermal shock a first wall.

Second, any intrusion of a fluid could disturb the plasma, thereby leading to a blanket impact. However, in a mirror plasma losses do not appear to impact the blanket, removing this type of possible interaction.

Third, intrusion of water could pressurize the plasma chamber. The volume of the plasma chamber of MARS appears to be about 700 m^3. Assuming sufficient heat transfer from the first wall to steam, about 500 kg of water is needed to over pressurize the chamber. This is of the same order as the amount of water in the entire choke coil and associated shielding so that such a single leak would not be expected to over pressure the chamber. The direct converter and halo scraper have 1-2 orders of magnitude more water, initially at $\sim$300°C and 22 MPa; a leak could pressurize the chamber.

Fourth, as explained for the case of shields, an oxidizing environment could oxidize and fail a V15Cr5Ti first wall and the issue is whether the first wall can be cooled fast enough. Further analysis would be required to answer this question with key aspects of (a) how large a leak occurs,

TABLE 5.4.3-9. MIRROR NEAR-PLASMA COMPONENT INTERACTIONS WITH BLANKETS

Blanket Type	Component	Potential Interactions	Comment
All	Water-cooled direct converter and halo scraper	Over pressure of plasma chamber	Concern
		Thermal shock of first wall	Nil probability
	Water-cooled choke coil and shielding	Over pressure of plasma chamber	Unlikely, insufficient water
		Thermal shock of first wall	Very low probability
V first wall	Water-cooled direct converter	Oxidation and failure of wall, volatility	Concern
	Water-cooled choke coil and shielding	Oxidation and failure of wall, volatility	Concern
Li bearing	Water-cooled components	Violent chemical reaction	Possible if lithium released
17Li83Pb bearing	Water-cooled components	Mild chemical reaction	Possible if 17Li83Pb released

(b) how long does it take for the oxygen potential to reach at least 10 Pa, and (c) does the blanket first wall cool below 650°C in that amount of time?

Fifth, should a Li/Li blanket wall fail while water is in the vacuum chamber, lithium-water reactions could occur. The same would be true for a Li/He blanket if the lithium breeder tubes also failed. This should/must be avoided since the Li-H_2O reaction is quite exothermic and could lead to temperatures over 1000°C, depending on the amount of material available for reaction, the geometry, and any heat sinks. 17Li83Pb-water reactions are milder than Li-H_2O but still of some concern based on recent tests at HEDL,[86] which are discussed in the steam generator subsection, 5.4.3.5.7.

5.4.3.5.4 Near-Plasma Components/Tokamak

The key near-plasma component in a tokamak is a limiter, and several limiter-blanket interactions appear possible, see Table 5.4.3-10. The first issue is the possibility of thermal shock of the first wall from a jet of water from a leak of limiter water coolant. Because of the tokamak geometry, any limiter leak is aimed at some part of the first wall, in contrast to the mirror case. Using a criterion based on Reference 83 of

$$\sigma = (0.1\ qd)\ \frac{E\alpha}{k(1-n)} > \sigma_{max}$$

where

σ	=	stress
d	=	wall thickness
q	=	heat flux

to determine if a thermal impulse fails the wall and a wall thickness of 5 mm, it is found that heat fluxes of order $4 - 12 \times 10^6\ W/m^2$ are

TABLE 5.4.3-10. TOKAMAK NEAR-PLASMA COMPONENT INTERACTIONS WITH BLANKETS

Blanket Type	Component	Potential Interactions	Comment
All	Water-cooled limiter	Overpressure of plasma chamber	Concern
		Thermal shock of first wall	Concern
	Helium-cooled limiter	Overpressure of plasma chamber	Very low probability
	Lithium-cooled limiter	Thermal shock of first wall	Concern(?)
V first wall	Water-cooled limiter	Oxidation and failure of wall	Concern
Elemental Li bearing	Water-cooled limiter	Chemical reaction	Concern

required to thermal shock the wall. The above expression is valid for short durations, under about 0.2 sec for these materials. The lower end of the range corresponds to 316SS and the higher one to V15Cr5Ti.

A second criterion was based on Reference 81:

$$\sigma = qt^{1/2} \left(\frac{E\,\alpha\,K}{k(1-\nu)}\right)^{1/2} > \sigma_{max} .$$

where

t = time

K = thermal diffusivity

At the longest time period (generally a few msec) for which the expression is valid and assuming a 5 mm wall, one finds that heat fluxes of 1×10^6 (316SS) to 3×10^6 (V) W/m^2 could cause thermal shock. Overall, heat fluxes for short times of $10^6 - 10^7$ W/m^2 are cause for concern. Based on heat flux expressions from References 84 and 85, it appears that fairly cold water (~100°C) from a limiter leak striking a hot first wall can achieve such a heat transfer. Steam or helium could not achieve sufficiently fast heat transfer, but lithium might. The chance of a blanket coolant thermal shocking a limiter wall seems lower since the temperature difference is lower than between limiter coolant and blanket first wall.

The second issue is whether a fluid would disturb the plasma, thereby leading to a blanket impact. In a tokamak such a disruption could lead to thermal and/or magnetic effects on the first wall leading to either elevated temperatures or wall failure or both.

Third, intrusion of water could pressurize the plasma chamber. One sector of the limiter has about 2×10^3 kg water associated with it, sufficient to raise pressures to 600-800kPa as the first wall heats the water/steam to 300-500°C. A helium or lithium-cooled limiter could not cause this problem.

Fourth, an oxidizing environment in the vacuum chamber could oxidize and eventually fail a V15Cr5Ti first wall, also causing radioactivity mobilization. As in other cases the issue is how rapid 10 Pa oxygen partial pressure is reached versus first wall cool-down.

Fifth, there are Li-H_2O and 17Li83Pb-H_2O reaction concerns if liquid metal in the blanket were released. The probability of this occurring if a water-cooled limiter ruptures appears high enough to merit special concern. Because of the possibility of thermal shock, adverse plasma disruption effects, and severe wall oxidation (if V15Cr5Ti), leak of cold (<100°C) limiter water could well cause blanket first wall failure (initially at $\gtrsim$600°C for steels, $\gtrsim$750 for V-alloy).

5.4.3.5.5 Primary Heat Exchanger/Steam Generator

In principle any steam generator or intermediate heat loop transient is of concern to the blanket since it might imply loss of heat removal function. For evaluation purposes, however, this is not considered here. Rather, one can consider it to be a case in-between LOCA and LOFA, which were already considered in the cooling transient evaluation, Index 5. Other than loss of heat removal, some special issues should be mentioned, see Table 5.4.3-11.

The Li/Li blankets use a sodium intermediate loop in order to avoid the possibility of radioactive, primary-loop lithium combining with steam/water in a steam generator, similar to LMFBR policy. To be consistent, one might make the case that if having water next to a primary Li(or Na) loop, but away from the nuclear island, is unacceptable, then having water in the nuclear island with lithium in the blanket is unacceptable. In any case, use of a sodium loop takes the Na-H_2O steam generator problem out of the reactor building so that any chemical reaction

TABLE 5.4.3-11. STEAM GENERATOR INTERACTIONS WITH BLANKETS

Blanket	Potential Interactions	Comment
All	Loss of heat removal function	Indirectly accounted for in cooling transient evaluation, Index 5
Li/Li with Na intermediate loop	Na-H_2O reactions, external to reactor buildings	From blanket perspective, only serves as one of many ways to lose cooling
17Li83Pb/17Li83Pb	17Li83Pb-H_2O reactions	Concern
All helium-cooled designs	Pressure increase in helium	Pressure relief valve required
Li/He/HT-9	Li-H_2O reactions	Concern
Li/He/V	Li-H_2O reactions V-H_2O oxidation	Concern Concern
All salt-cooled designs	Pressure increase in salt	Pressure relief valve required
All H_2O-cooled designs	None	None

cannot harm the blanket. From the blanket perspective, the intermediate loop Na-H_2O reaction only serves as one of many ways to lose steam generator function.

In future design studies, designers should consider eliminating the intermediate loop and using a double-wall steam generator instead. The double-wall with leak detection reduces the probability of interaction. Here it is important to draw a distinction from the LMFBR case. In LMFBR's, a primary loop sodium/water interaction would mobilize highly activated sodium. In fusion, a primary loop lithium/water iteration would only mobilize lithium impurities, corrosion products, and tritium. The increased risk of eliminating the intermediate loop for lithium self-cooled designs may be warranted given the savings in economics (∿4% decease in COE) and engineering complexity.

The 17Li83Pb/17Li83Pb blankets do not use an intermediate loop, so a steam generator break leads to 17Li83Pb-H_2O reactions. A recent test at HEDL[86] used a steam-generator-type geometry. Steam at 1MPa, 350°C was injected into 200 kg of 17Li83Pb at 500°C at a rate of about 5 g/s for 325 s. The temperature rose to 870°C and may have started to level off, perhaps because of lithium depletion. The lithium content fell from 0.0068 weight percent to 0.0022 weight percent. Material collected from the aerosol and from the top of the reaction chamber included LiOH, Li_2O, and Pb.

These results are consistent with recent modeling.[87] The results indicate that thermodynamics allow a Li + H_2O steam generator temperature over 2000°C for possible reaction zone pressures; 17Li83Pb + H_2O temperatures are thermodynamically limited to about 1200°C. Kinetics calculations indicate that reaction zone temperatures for Li + H_2O could be high enough to melt steel. More detailed, more realistic calculations are needed to determine if fault propagation is possible for a Li + H_2O case. Similar kinetics calculations for 17Li83Pb + H_2O indicate upper temperatures of 1100°C, too low to melt near-by tubes.

The maximum 17Li83Pb + H_2O temperatures lie somewhere between 870°C (experiments) and 1100°C (modeling). These temperatures are those projected for the liquid metal reaction zone. If nearby tubes are subjected to these temperatures for long times they may fail, although it should be emphasized that nearby tubes would still be cooled, removing heat from the reaction zone and lowering nearby tube temperatures. At this time it is not known if propagating tube failures are possible or likely in a 17Li83Pb-water steam generator. Thus the viability of that combination is unknown. If failure propagation is found to be reasonably likely, it appears that the 17Li83Pb cooled blanket would have to have an intermediate loop with associated economic and thermal efficiency penalities.

In summary, the reference cases are to use an intermediate loop with lithium-cooled blankets and not to use an intermediate loop with 17Li83Pb-cooled blankets. The key issues include the possibility of fault propagation following a single water tube break and the radioactivity consequences of metal-water interactions. Given what is known at the end of the BCSS, it appears that the pressure response in the steam generator is about the same, regardless of the liquid-metal being sodium, lithium, or 17Li83Pb. The primary loop radioactivity is higher in sodium (LMFBR) and 17Li83Pb but lower in lithium. Future analysis might conclude that a primary 17Li83Pb steam generator had higher risk than a primary lithium steam generator, possibly leading to the conclusion of having an intermediate loop for a 17Li83Pb-cooled blanket, but not for a lithium-cooled blanket, the opposite of the reference BCSS case.

The helium-cooled designs operate with about 5.2 MPa pressure helium. The steam pressure is 8.3 MPa. Thus a steam generator leak could raise pressure in the helium coolant. A simple pressure relief valve system should be an acceptable passive safety safeguard. One would also probably use some active detection means to monitor for leakage into the helium and shutdown if the leakage were too high. The He/Li/HT-9 and He/Li/V blankets are special cases. At any given time there may be some level of leakage from helium coolant to lithium breeder so that they may be a pathway from steam generator to lithium. However, the time to respond appears long (hours?) and the amount of moisture (and associated $Li + H_2O$ consequences) involved is likely to be small. Thus the steam generator-lithium breeder risk seems fairly low. The He/V case has an additional worry, attack of the V15Cr5Ti by steam or air. If V15Cr5Ti operating temperatures are above 650°C as seems likely, then oxidation appears a concern for even modest oxygen partial pressures, perhaps as low as 10 Pa. This is one motivation for the Li/He/V15Cr5Ti design using a double wall steam generator, to reduce the probability of water in-leakage.

The only special steam generator concern for the Nitrate Salt blankets is pressure. The salt is initially at 0.4 MPa with steam at 10.4 MPa. As in the case of helium designs, a pressure relief valve system should fulfill the need for passive safety. The salt does not react with water.

5.4.3.5.6 Magnets and Building

The resistive portion of the choke coil magnet for the mirror is considered with near-plasma components. The electromagnetic interaction between all magnets and blankets is considered under external forces, Index 6, and is not repeated here. The possibility of missile generation from any magnet failure cannot be ruled out at this time, but in any case would have blanket-independent failure probability. If missiles were generated, the blanket transient would be a loss of coolant into the vacuum chamber, considered under near-plasma components in this index; or a loss of coolant into the breeder zone, considered under coolant-breeder interactions (Index 4); or a major loss of cooling, considered under cooling transients (Index 5). Further examination of missiles is beyond the scope of this study. If magnets were to experience a cryogen release, the building would be somewhat pressurized; this is discussed in Index 8. Thus, other than choke coil involvement via the release of coolant into the plasma chamber, magnet interactions are either not considered or considered in other indices.

The building, in a sense, transmits a seismic disturbance to the reactor and blanket. The seismic issue is evaluated under external forces, Index 6. The possibility of blanket-dependent failures to the containment function of the reactor building is considered in Index 8. Magnet and building concerns are summarized in Table 5.4.3-12.

TABLE 5.4.3-12. MAGNETS AND BUILDING INTERACTIONS WITH BLANKETS

Component	Interaction	Comments
Magnets	Electromagnetic	Discussed in Index 6
	Missile damage	Resulting damage translates into transients considered elsewhere
	Cryogen pressurization of building	Discussed in Index 8
Building	Seismic forces	Evaluated in Index 6
	Blanket-dependent failure modes to containment function	Evaluated in Index 8

5.4.3.5.7 Design Philosophy.

The first two design goals were established early in the study. The third came later. The first design goal is that the first wall and blanket should withstand failures in nearby components. These include limiters, choke coils, and shields. This prevents failure propagation. The second design goal deals with chemical reactions and says that lithium-water reactions, in or near the blanket should be eliminated by design. For liquid-lithium modules, this eliminated the option of water-cooled blanket modules. Early in the study it was agreed that water-cooled shield, limiters, divertors, or choke coils may be acceptable with a liquid lithium module, provided that the probability of substantial lithium-water reaction is $\sim 10^{-4}$ per year or lower. A third design goal needed to be added later in the study after EG&G experiments indicated the seriousness of V15Cr5Ti-air reactions.[46] The third design goal is that V15Cr5Ti-steam reactions should be eliminated by design whenever the V15Cr5Ti temperature is above 650°C and the equivalent oxygen partial pressure is over 10 Pa (10^{-4} atmosphere). These matters will now be discussed.

5.4.3.5.8 Evaluation Scheme

Via the above discussion, the wide-ranging area of near-blanket system interactions has been considerably narrowed with cooling, electromagnetic, and building concerns being evaluated elsewhere and minor issues being eliminated from further attention. The remaining problems are all associated with the plasma chamber, specifically chemical reactions, chamber pressurization, and thermal shock.

The technical issues associated with blanket/plasma chamber/limiter and blanket/plasma chamber/choke coil chemical reactions and accidental pressurization are exceedingly complex. Tables 5.4.3-13 and 5.4.3-14 list factors influencing the risk of simultaneously using water and the highly reactive metals, lithium and V15Cr5Ti.

The key issues with V15Cr5Ti are temperature and oxidizing potential. EG&G experiments indicate that 100 kPa (one atmosphere) air in contact with

TABLE 5.4.3-13. PROBABILITY FACTORS INFLUENCING THE RISK OF SIMULTANEOUSLY USING REACTIVE METALS AND WATER IN A FUSION REACTOR

Scenario: Water-cooled component failure induces liquid-metal-cooled component failure

--Probability of water leaking from water-cooled component

--Probability of water reaching the liquid metal-cooled component given a water leak

--Probability of liquid metal-cooled component failing given contact of water

Scenario: Common cause failure of both components

--Probability of one initiator failing both components, e.g., plasma disruption or seismic event

Scenario: Liquid-metal-cooled component failure induces water-cooled component failure

--Probability of liquid metal leaking from liquid-metal-cooled component

--Probability of liquid metal reaching the water-cooled component given a liquid metal leak

--Probability of water-cooled component failing given contact of liquid metal

All Scenarios:

--Probability of activated material being mobilized given reactive metal-water reaction

--Probability of mobilized material being transported away from the reactor

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TABLE 5.4.3-14. SOME CONSEQUENCE FACTORS INFLUENCING THE RISK OF SIMULTANEOUSLY USING REACTIVE METALS AND WATER IN A FUSION REACTOR

--Amount of water that is leakable

--Volume of vacuum chamber

--Amount of liquid metal that is leakable

--Liquid metal choice: lithium vs. 17Li83Pb

--Structure choice: V15Cr5Ti vs. steel, copper[a]

--Distance between water-cooled component and liquid-metal-cooled component[a]

--Water pressure and temperature[a]

--Liquid metal-cooled component plasma side temperature[a]

--Liquid metal temperature[b]

--Water-cooled component plasma side temperature[b]

a. Influences ability of water/steam to fail liquid-metal-cooled component, e.g., thermal shock (likely if water still in condensed state when strikes the liquid metal-cooled component) or V15Cr5Ti rapid oxidation.

b. Influences ability of liquid metal to fail water-cooled component, e.g., thermal shock (depending on temperatures involved).

V15Cr5Ti above 650°C produces a V_2O_5-Cr_2O_3-TiO_2 molten oxide (m.p. ~655°C) with substantial V_2O_5 volatility. 100 kPa carbon dioxide gas exposure only pushed the vanadium oxidation state to VO_2 (m.p. 1545°C) rather than V_2O_5. Thus the lower oxidizing potential of carbon dioxide avoided the formation of molten oxide and caused about an order of magnitude less vanadium volatility. It is expected that the primary issue for the steam case is which oxide state, VO_2 or V_2O_5, is reached. This should be a function of steam pressure and temperature, but the function is unknown at present. Based on the carbon dioxide and air results, steam with oxygen partial pressures below ~10 Pa at ~650°C should not produce the V_2O_5 specie, allowing a V15Cr5Ti wall to survive. Steam with oxygen partial pressure $\gtrsim$100 Pa at >650°C is likely to produce the V_2O_5 specie, causing rapid wall failure and radioactivity mobilization.

The key issue with the steel case is whether exposure to water/steam would cause a first wall failure. Operational temperatures are insufficient for rapid oxidation to occur, but thermal shock is another pathway.

A Li-H_2O reaction in the vacuum chamber would likely have serious, unacceptable, consequences.

Although a 17Li83Pb-H_2O reaction should be avoided, its consequences are orders of magnitude less serious than the Li-H_2O case. Similarly, the Li_2O-H_2O reaction should be avoided but is mild compared with lithium. The breeders $LiAlO_2$ and FLIBE do not react with water, but $LiAlO_2$ does apparently dissolve.

It is beyond the scope of this study to consider all the above complexities in this evaluation. Initially an attempt was made to consider some second-order differences among designs, e.g. Li/Li/V versus Li/He/V, in the evaluation, but this became unworkable. Only first-order differences, considering whether lithium or V15Cr5Ti are present at all, were finally included.

The first stage is to determine if any concepts have a fatal flaw, Table 5.4.3-15. Two cases appear unacceptable, Li/V and Li/steel with a water-cooled limiter. The reason is that a single failure of the limiter cooling might lead to (a) first wall failure, (b) Li + H_2O reaction, (c) plasma chamber pressurization and rupture, and (d) radioactivity transport. All other combinations appear to lack one of these elements. For example, a LiPb/V blanket with a limiter could experience elements (a), (c), and (d) but not (b), Li + H_2O reactions. A Li/V blanket with a water-cooled choke coil could experience (a) and (b) but there is not sufficient water to rupture the chamber. For the above reasons, nonwater-cooled limiters were adopted for Li or V15Cr5Ti designs. It

TABLE 5.4.3-15. INDICATIONS OF WHICH COMBINATIONS SHOULD BE JUDGED AS HAVING A FATAL FLAW

Blanket	Water-Cooled Component	Fatal Flaw?	Comments
Li/V	Limiter	Yes	Assumes T_{wall} >650°C
Li/V	Choke Coil[c]	No	Water supply not adequate to pressurize vacuum chamber,[a] design makes FW thermal shock unlikely; V-wall failure possible
Li/steel	Limiter	Yes	FW can be thermal shocked[b]
Li/steel	Choke Coil	No	Design makes FW thermal shock unlikely
LiPb/V	Limiter	No	V-wall failure possible
LiPb/V	Choke Coil	No	Water supply not adequate to pressurize vacuum chamber,[a] design makes FW thermal shock unlikely; V-wall failure possible
LiPb/steel	Limiter	No	Only mild reactions
LiPb/steel	Choke Coil	No	Only mild reactions

a. Issue is whether water can pressurize and rupture vacuum chamber as well as raise oxidation high enough to push V15Cr5Ti to final oxidation state, V_2O_5, which melts >650°C.

b. Issue is whether water leak necessarily fails FW.

c. Choke coil refers to the resistive part and its associated shielding.

should be noted although passive means to eliminate these flaws have not been found (other than removing water or lithium), active protection schemes appear possible. For example, it has been proposed[88] that the water inventory be dumped rapidly to an external tank if one detects water leaking into the vacuum chamber, requiring several active measures to work.

The second stage is a relative comparison among concepts. The scoring scheme for mirror blankets is given in Table 5.4.3-16. All designs use water in the choke coils (and associated shielding), direct converters, halo scrapers, and shields. As explained above, the evaluation is only based on the first-order questions of which reactive metal, vanadium or lithium, is present. 17Li83Pb should probably be slightly penalized, but its risk appears much lower than lithium or vanadium.

The scoring scheme for tokamak blankets is given in Table 5.4.3-17. The final designs all employ water-cooled shields. However, at early stages of the study, some were helium-cooled and the evaluation scheme had been established to account for the benefit of completely removing water from the nuclear island.

5.4.3.5.9 Index 7 Results.

Given the above evaluation scheme, the scores are straightforward. Table 5.4.3-18 lists the scores for the mirror blankets. Although water is present in the direct converter, halo scraper, choke coil and its associated shielding, and the shields, no one single failure could obviously lead to large scale Li-H_2O reactions. Given the level of analysis in this study, tagging any mirror case as having a fatal flaw is not defendable. However, future reactor designers must seriously consider the risk of water in these components, especially for blankets with reactive metals. From the safety perspective it is strongly recommended that no water be used in any of these components if lithium or V-alloy is used in the blanket. A softer version of this recommendation is appropriate for 17Li83Pb.

Table 5.4.3-19 lists the scores for the tokamak blankets. Because a somewhat plausible case can be made that a single failure of a water-cooled limiter could lead to H_2O - Li reactions if lithium or V is in the blanket, use of a water-cooled limiter with lithium was judged a fatal flaw. Of the elemental Li-bearing blankets, Li/Li/V and Li/He/HT-9 are official tokamak cases for evaluation. Fatal flaw was avoided by use of a lithium-cooled limiter. Such a limiter would also appropriate for Li/He/V

TABLE 5.4.3-16. SCORING SCHEME FOR METAL-WATER REACTIONS IN FAULT TOLERANCE TO NEAR-BLANKET SYSTEMS INTERACTIONS, INDEX 7, FOR MIRROR CONCEPTS

Relative Consequence (0 to 2 utility points)

2 - Both reactive metals, vanadium and lithium, present.

1 - One reactive metal present.

0 - Neither reactive metal present.

Relative Probability

All concepts are approximately the same. All have water-cooled choke coils, direct convertors, halo scrapers, and shields.

Relative Risk (0 to 2 utility points)

Relative Risk = Relative Consequences.

TABLE 5.4.3-17. SCORING SCHEME FOR METAL-WATER REACTIONS IN FAULT TOLERANCE TO NEAR-BLANKET SYSTEMS INTERACTIONS, INDEX 7, FOR TOKAMAK CONCEPTS

Relative Probability (0 to 2 utility points)

2 - Shield and limiter are both water-cooled.

1 - Shield is water-cooled.

0 - Neither shield nor limiter are water-cooled.

Relative Consequence (0 to 2 utility points)

2 - Both reactive metals, vanadium and lithium, present.

1 - One reactive metal present.

0 - Neither reactive metal present.

Relative Risk (0 to 4 utility points)

Relative Risk = Relative Probability x Relative Consequence.

TABLE 5.4.3-18. INDEX SCORES FOR FAULT TOLERANCE TO NEAR-BLANKET SYSTEM INTERACTIONS, INDEX 7, FOR MIRROR BLANKETS

Blanket	Utility Points Relative Consequence	Index[a] Score
Li/Li/V15Cr5Ti	2	0.0[b]
Li/Li/HT-9	1	0.5
17Li83Pb/17Li83Pb/V15Cr5Ti	1	0.5
17Li83Pb/17Li83Pb/HT-9	0	1.0
Li/He/V15Cr5Ti	2	0.0[b]
Li/He/HT-9	1	0.5
17Li83Pb/He/HT-9	0	1.0
FLIBE/He/HT-9/Be	0	1.0
Li_2O/He/HT-9	0	1.0
$LiAlO_2$/He/HT-9/Be	0	1.0
$LiAlO_2$/He/PCA/Be	0	1.0
17Li83Pb/NS/HT-9	0	1.0
$LiAlO_2$/NS/HT-9/Be	0	1.0
$LiAlO_2$/NS/PCA/Be	0	1.0
$LiAlO_2$/H_2O/HT-9/Be	0	1.0
$LiAlO_2$/H_2O/PCA/Be	0	1.0

a. Index Score = 1 - 1/2 (utility points).

b. Neither of these cases is a fatal flaw since a single failure would not automatically lead to a substantial chemical reaction.

TABLE 5.4.3-19. INDEX SCORES FOR FAULT TOLERANCE TO NEAR-BLANKET SYSTEM INTERACTIONS, INDEX 7, FOR TOKAMAK BLANKETS

	Utility Points			
Blanket	Relative Probability	Relative Consequence	Relative Risk	Index[a] Score
Li/Li/V15Cr5Ti	1	2	2	0.0
Li/Li/HT-9	1	1	1	0.5
17Li83Pb/17Li83Pb/V15Cr5Ti	1	1	1	0.5
17Li83Pb/17Li83Pb/HT-9	2	0	0	1.0
Li/He/V15Cr5Ti	1	2	2	0.0
Li/He/HT-9	1	1	1	0.5
17Li83Pb/He/HT-9	2	0	0	1.0
FLIBE/He/HT-9/Be	2	0	0	1.0
Li_2O/He/HT-9	2	0	0	1.0
$LiAlO_2$/He/HT-9/Be	2	0	0	1.0
$LiAlO_2$/He/PCA/Be	2	0	0	1.0
17Li83Pb/NS/HT-9	2	0	0	1.0
$LiAlO_2$/NS/HT-9/Be	2	0	0	1.0
$LiAlO_2$/NS/PCA/Be	2	0	0	1.0
$LiAlO_2$/H_2O/HT-9/Be	2	0	0	1.0
$LiAlO_2$/H_2O/PCA/Be	2	0	0	1.0

a. Index score = 1 - 1/2 (relative risk points).

and Li/Li/HT-9. The LiPb/LiPb/V case is interesting. Use of a lithium-cooled limiter with this case defeats a main value of using LiPb in the blanket. A 17Li83Pb-cooled limiter looks significantly more difficult to design than a lithium-cooled limiter. A helium-cooled may be a better idea, if feasible.

Elimination of water-cooled limiters for lithium blankets designs removed a fatal flaw but still leaves water in the shield. Future reactor designers must seriously consider the risk posed by use of water, especially with reactive metals. From the safety perspective, it is strongly recommended that no water be used in any component in the nuclear island if lithium or V-alloy is present. A softer version of the recommendation is appropriate for 17Li83Pb.

5.4.3.6 Fault Tolerance of the Reactor Building to Blanket Transients, Index 8

Reactor accidents generally cannot harm the public unless radioactivity or chemical toxins escape from the reactor building. Therefore, a fusion reactor building will have some type of containment function. The probability of containment failure is a very important part of total reactor accident risk. Since the blanket choice can greatly influence the containment failure probability, the potential of blanket accidents to impair the containment function is a valid and important part of blanket comparisons and is the subject of Index 8.

5.4.3.6.1 Design Philosophy

Several design goals were established. First, the concept design should prevent severe accidental pressure sources from compromising reactor building containment integrity. Such accidents include loss of primary coolant, e.g. water over pressure, and rupture of magnet helium integrity. Second, concepts should be designed to prevent fires from chemical spills, e.g. lithium-air reactions. Such fires could damage building equipment, pressurize the building, and cause expensive (economic and occupational-radiation exposure) cleanup. Third, concept design should prevent concrete reactions, e.g. lithium-concrete reactions. Again, such reactions could pressurize the building, fail concrete, and/or cause expensive cleanup.

5.4.3.6.2 Evaluation Scheme

The building issues divide into those directly associated with (a) building atmosphere choice and over pressurization and (b) building floor and equipment. For the former, the key designs are building atmosphere choice and blanket fluid choices discussed in subsection 5.4.3.6.3. For the latter, only blanket fluid choices are relevant, discussed in subsection 5.4.3.6.4. The general scoring scheme is indicated in Table 5.4.3-20 and explained below.

5.4.3.6.3 Building Atmosphere and Pressurization

The possible overpressures caused by various transients are strong functions of the chosen blanket fluids and building atmosphere. These

TABLE 5.4.3-20. SCORING SCHEME FOR FAULT TOLERANCE OF REACTOR BUILDING TO BLANKET TRANSIENTS, INDEX 8

Relative Probability (0 to 2 utility points)

2 - Coolant spill causes high pressure or equipment damage.

1 - Breeder spill causes high pressure or equipment damage.

0 - Neither coolant nor breeder can cause high pressure or equipment damage.

Relative Consequences (0 to 3 utility points)

3 - High pressure ($\sim$100 kPa) in building (water).

2 - High pressure in building if secondary failure occurs (lithium).

1 - Equipment damage or very difficult cleanup required (17Li83Pb, Nitrate Salt, FLIBE).

0 - No serious damage or difficult cleanup required (Helium).

Relative Risk (0 to 6 utility points)

Relative Risk = Relative Probability x Relative Consequence.

differences should be considered in both the safety and economic evaluations of the various blanket concepts. This discussion details the calculation of the maximum building over pressures for design purposes for the various BCSS concepts.

The calculation of overpressure from release of helium coolant or water coolant is straightforward and explained in Reference 5. For water, overpressure scales approximately as (mass of water/volume of building) 0.85. The result for the STARFIRE case is 80 kPa, which is within the 70 to 100 kPa range indicated in the STARFIRE report.[2] The analysis for MARS[4] indicated an overpressure of 19 kPa from release of the water coolant for the shield/reflector/direct converter systems. For helium coolant, the overpressure scales approximately linearly with the mass of helium and inversely with building volume, with a more complicated dependence on initial helium temperature.

The cryogenic helium release case must be considered since all reactor designs must include magnets. If the blanket fluids cannot cause overpressures higher than that resulting from a cryogen release, the building overpressure design can ignore the blanket fluid choice. There is one possible exception. If common-mode failures between cryogen and blanket can be ruled out so that cryogen releases do not involve radioactivity, then the building can simply release the expanding helium rather than confine the pressure. For BCSS purposes, we can not rule out cryogen releases that simultaneously involve radioactivity. Therefore, it is assumed that the building must be able to confine cryogen-related overpressures.

For a STARFIRE size building, equilibrium calculations[5,24] indicate a final overpressure of ∿2.4 kPa/Mg He released. Here, it is assumed that the overpressure scales inversely linearly with building volume. A transient calculation for the specific FED design[15] indicates a final overpressure of ∿2.5 kPa/Mg He in failed coils or ∿1.4 kPa/Mg He released.

Appropriate cryogen system design for a tokamak should limit the amount released to that associated with one TF coil. The BCSS tokamak base case is a tokamak with 10 TF coils rather than 12 as in STARFIRE. The total TF liquid helium is assumed to be the same as for STARFIRE. The release of one TF coil helium causes $\sim$7 kPa overpressure. There could be an initial underpressure, depending on the transient, before the final state is reached. For MARS, a value of 4 kPa overpressure for an end cell magnet cryogen system rupture is expected.(4)

Existing overpressure results(89-91) for a lithium fire using the LITFIRE code (92) in a STARFIRE-size building range from 100 to 220 kPa with the lower value more recent. Temperatures in the lithium reaction zone could reach 1260°C.(20) Lithium-water or lithium-concrete reactions could cause similar pressures, although the modeling is not sufficiently advanced to know details.

Because of the lithium-air fire problem, use of another gas was desired. Experiments indicate that the Li-CO_2 reaction is even worse than Li-air, so carbon dioxide gas is not an option for lithium cases. Another approach would be to use fire-suppression techniques. From the safety perspective, helium or another inert gas would be best. Considering economics, nitrogen was chosen. (Section 5.4.4 includes some activation-relevant discussions.) Nitrogen does react with lithium, but only mildly. 10-kg of lithium at 530°C, when exposed to nitrogen, just cooled down.(20) 10-kg of lithium at 840°C reacted with nitrogen to form Li_3N and the temperature rose to 960°C. Thermodynamically the free energy change for Li + N_2 going to Li_3N is unfavorable above about 1000°C,(5) inherently limiting the reaction. Because the highest lithium temperature (600°C) in the BCSS is only modestly above the 530°C data point, it is felt that lithium-nitrogen reactions pose only a minor risk.

Although V15Cr5Ti does not react with air to cause a pressure increase and thus influence containment, it should be mentioned that an air leak into a V15Cr5Ti structure while it is above 650°C could lead to catastrophic failure. Use of nitrogen or carbon dioxide should solve this problem.

Unless elemental lithium (or sodium) is present, no significant reactions or overpressures are expected with FLIBE. Nitrate salt could pose a problem if reactive materials were available, but which materials are relevant and the corresponding resulting pressures are unknown. For now, if known reactive materials like lithium are avoided, possible building overpressures from use of Nitrate Salt should be modest. Localized pressures in the blanket, however, could still be high due to salt decomposition.

The 17Li83Pb case is also not fully known. The major concern is water by itself or as part of concrete. H_2 is generated from the 17Li83Pb/water reaction, at ∿0.25 moles H_2 per kg of 17Li83Pb reacted.[20] Thus, 170 m^3 of 17Li83Pb would have to completely react with water for the level of hydrogen to reach the 4% lower hydrogen inflammability limit in the building. This is unlikely. There is insufficient 17Li83Pb to reach the 18% lower hydrogen detonation limit in the building. 17Li83Pb appears fairly inert in air.[54, 93, 94] A recent experiment[65] showed that 17Li83Pb is not inert in carbon dioxide. In fact, 17Li83Pb-carbon dioxide reactions appear to be worse than 17Li83Pb-air reactions. In the BCSS, nitrogen was used as the cover gas for 17Li83Pb. For these reasons, 17Li83Pb does not appear to present a building overpressure problem; although, like Nitrate Salt, localized high pressures might be possible.

The final question is what happens in those cases where the steam generator is located in the building. Appropriate design, including check valves, could probably reduce the problem, but it might be worse than the cryogen release case. The special case of a lithium/steam generator is ruled out: the intermediate sodium loop will go from the lithium primary loop to a steam generator outside the reactor building. The special case of a 17Li83Pb/steam generator is less clear, particularly since a recent HEDL experiment indicates some volatilization of Pb from the 17Li83Pb-water reaction.[86]

Without water and lithium being present, the cryogen release case is limiting. If substantial amounts of pressurized water are present, the overpressure from water/steam blowdown dominates. If elemental lithium (or sodium) is present along with air, water, or carbon dioxide, that case dominates. Selection of helium, solid breeders, molten salts, or 17Li83Pb does not appear to cause overpressure concerns for the building beyond that from the cryogen release case. The major possible exception is Nitrate Salt in the presence of reactive materials. The calculated maximum overpressures are listed in Table 5.4.3-21 for the cases discussed above.

5.4.3.6.4 Concrete and Equipment Protection

With the exception of helium, a spill of other blanket fluids would entail nontrivial problems. Since all will be activated, some occupational exposure and waste disposal are likely to result.

TABLE 5.4.3-21. MAXIMUM OVER-PRESSURES (kPa)[a]

	Starfire Building	Mars Building
One TF Coil	7	--
End Cell	--	4(4)
One Helium Coolant Loop	5(5)	6
One Water Coolant Loop	80(5)	97
Shield/Reflector/Direct Converter Water	--	19(4)
Lithium Fire (Lithium-Air Reaction)	100	100
FLIBE	0	0
Nitrate Salt	low [b]	low[b]
17Li83Pb	0	0

a. 1 kPa = 0.145 psi; numbers in parentheses are reference numbers.

b. Depends on which reactive materials are present.

Hot lithium could easily damage any equipment it spilled on, but it should be fairly easy to clean up. After lithium solidified, some could by physically removed in pieces, the remainder could be steam-cleaned.

17Li83Pb or FLIBE could also damage any equipment they fell on. Proper cleanup technique is not known, especially for 17Li83Pb. Development of proper solvents to remove trace amounts of these materials would be welcome. Disposal catch pans would be useful for highly vulnerable places outside of the blankets. However, a solvent appears to be required to remove 17Li84Pb . FLIBE if they fell between blanket sectors. The 17Li83Pb might self-weld.

Nitrate Salt could damage equipment. Water could serve as a cleaning solvent.

A small water spill should be fairly easy to clean. However, a large water/steam release could damage unprotected equipment throughout the plant, a finding from TMI experience.[95]

5.4.3.6.5 <u>Index 8 Results</u>

The key assumption in the area of how the building reacts to blanket transients is that the risk of chemical reactions has been reduced or eliminated by design, especially in the case of lithium. For lithium designs, a nitrogen atmosphere is used and assumed practical. Air and carbon dioxide react very strongly with lithium while it appears that nitrogen reacts only mildly. For lithium and 17Li83Pb, concrete reactions are taken to be solved by use of a steel liner. To reduce any problems of hot fluids striking concrete, steel liners were also used with FLIBE and Nitrate Salt.

Given the above assumptions and design details, the Index 8 evaluations are somewhat straightforward and listed in Table 5.4.3-22. The Li_2O/He and $LiAlO_2$/He designs are best because no transient has been identified by which these blankets impact the containment function or impose a difficult cleanup. The FLIBE/He and 17Li83Pb/He blankets pose a

TABLE 5.4.3-22. INDEX SCORES FOR FAULT TOLERANCE OF THE REACTOR BUILDING TO BLANKET TRANSIENTS, INDEX 8

	Utility Points			
Blanket[a]	Relative Probability	Relative Consequence	Relative Risk	Index[b] Score
Li/Li	2	2	4	0.33
17Li83Pb/17Li83Pb	2	1	2	0.67
Li/He	1	2	2	0.67
17Li83Pb/He	1	1	1	0.83
FLIBE/He	1	1	1	0.83
Li_2O/He	0	0	0	1.0
$LiAlO_2$/He	0	0	0	1.0
17Li83Pb/NS	2	1	2	0.67
$LiAlO_2$/NS	2	1	2	0.67
$LiAlO_2/H_2O$	2	3	6	0.0

a. The blanket structural choice is not relevant to this index.

b. Index Score = 1 - 1/6 (relative risk points).

difficult cleanup of spilled breeder. In fact it is not clear how one would remove either highly radioactive material from wherever it spilled, building or vacuum chamber. The Nitrate Salt and 17Li83Pb-cooled blankets pose a difficult cleanup problem of spilled coolant.

The Li/He blanket has a potential pathway to fail containment if spilled lithium breeder were to contact concrete that has been covered with steel. The same holds true for the Li/Li blanket except that a coolant spill is more likely than a breeder spill. In either lithium case, lithium fires are assumed eliminated by use of nitrogen gas in the building.

The water-cooled blankets pose a severe pressurization problem for the containment function of the building.

5.4.3.7 Summary.

The blanket designs have been examined to see how fault tolerant they are to breeder-coolant mixing (Index 4), cooling transients (Index 5), external forces (Index 6), and failures of near-by systems (Index 7). The impact of blanket failures on the containment ability of the reactor building (Index 8) has also been examined. The resulting scores are listed in Tables 5.4.3-23 and 5.4.3-24. Some brief comments follow.

For liquid-metal cooled blankets, the main problems are chemical reactivity of lithium, 17Li83Pb, and V15Cr5Ti. It has been assumed that nitrogen atmospheres are used with lithium and 17Li83Pb to prevent fires.

This assumption requires that, first, nitrogen-lithium reactions are mild; experimental evidence supports this. Second, this probaility of air from outside the building influencing chemical reactions must be insignificantly small. This probability should indeed by insignificantly low. For air from outside the building to react with lithium requires (a) the lithium being spilled, (b) the containment function of the building being massively failed, (c) a driving force for air to displace nitrogen in the building being present, and (d) sufficient time occurring to allow the oxygen concentration in a volume of 2×10^5 m^3 to raise to near normal atmospheric levels. Reactor building design will, however, have to preclude pathways for air to bypass the containment function. For example, it would not be wise if the RF ducts that lead into the plasma chamber come from outside the reactor building, as for STARFIRE, as this provides a path for air from the outside directly into the plasma chamber.

It has also been assumed that the risk reactions with water have been reduced because of the use of nonwater-cooled limiters and assumed adequate separation of liquid-metal and water. Even with these favorable assumptions, use of lithium or vanadium has some penalties in the fault tolerance evaluation. For heavier blankets, specifically 17Li83Pb, it has not yet been verified that the blanket could be seismically qualified.

TABLE 5.4.3-23. SUMMARY OF FAULT TOLERANCE SCORES, INDICES 4-8, FOR MIRROR CONCEPTS

Blanket	Index 4	Index 5	Index 6	Index 7	Index 8	Total Fault[a] Tolerance
Li/Li/V15Cr5Ti	1.00	1.00	0.92	0.00	0.33	19.50
Li/Li/HT-9	1.00	0.83	0.91	0.50	0.33	21.42
17Li83Pb/17Li83Pb/V15Cr5Ti	1.00	1.00	0.74	0.50	0.67	23.46
17Li83Pb/17Li83Pb/HT-9	1.00	0.83	0.73	1.00	0.67	25.38
Li/He/V15Cr5Ti	0.67	1.00	0.95	0.00	0.67	19.74
Li/He/HT-9	0.67	1.00	0.93	0.50	0.67	22.62
17Li83Pb/He/HT-9/Be	0.67	1.00	0.67	1.00	0.83	25.02
FLIBE/He/HT-9	0.67	0.67	0.94	1.00	0.83	24.66
Li_2O/He/HT-9	0.67	0.33	0.99	1.00	1.00	23.94
$LiAlO_2$/He/HT-9/Be	0.67	0.33	1.00	1.00	1.00	24.00
$LiAlO_2$/He/PCA/Be	0.67	0.33	1.00	1.00	1.00	24.00
17Li83Pb/NS/HT-9	0.00	1.00	0.83	1.00	0.67	21.00
$LiAlO_2$/NS/HT-9/Be	0.33	1.00	1.00	1.00	0.67	24.00
$LiAlO_2$/NS/PCA/Be	0.33	1.00	1.00	1.00	0.67	24.00
$LiAlO_2$/H_2O/HT-9/Be	0.67	0.67	0.97	1.00	0.00	19.86
$LiAlO_2$/H_2O/PCA/Be	0.67	0.67	0.97	1.00	0.00	19.86

a. Total Fault Tolerance = $\sum_{i=4}^{8} I_i W_i$, ranges from 0 to 30.

TABLE 5.4.3-24. SUMMARY OF FAULT TOLERANCE SCORES, INDICES 4-8, FOR TOKAMAK CONCEPTS

Blanket	Index 4	Index 5	Index 6	Index 7	Index 8	Total Fault Tolerance[a]
Li/Li/V15Cr5Ti	1.00	1.00	0.59	0.00	0.33	17.52
Li/Li/HT-9	1.00	0.83	0.42	0.50	0.33	18.48
17Li83Pb/17Li83Pb/V15Cr5Ti	1.00	1.00	0.34	0.50	0.67	21.06
17Li83Pb/17Li83Pb/HT-9	1.00	0.83	0.17	1.00	0.67	22.02
Li/He/V15Cr5Ti	0.67	1.00	0.63	0.00	0.67	17.82
Li/He/HT-9	0.67	1.00	0.45	0.50	0.67	19.74
17Li83Pb/He/HT-9	0.67	1.00	0.20	1.00	0.83	22.20
FLIBE/He/HT-9/Be	0.67	0.67	0.46	1.00	0.83	21.78
Li_2O/He/HT-9	0.67	0.33	0.49	1.00	1.00	20.94
$LiAlO_2$/He/HT-9/Be	0.67	0.33	0.50	1.00	1.00	21.00
$LiAlO_2$/He/PCA/Be	0.67	0.33	0.33	1.00	1.00	19.98
17Li83Pb/NS/HT-9	0.00	1.00	0.33	1.00	0.67	18.00
$LiAlO_2$/NS/HT-9/Be	0.33	1.00	0.50	1.00	0.67	21.00
$LiAlO_2$/NS/PCA/Be	0.33	1.00	0.33	1.00	0.67	19.98
$LiAlO_2$/H_2O/HT-9/Be	0.67	0.67	0.48	1.00	0.00	16.92
$LiAlO_2$/H_2O/PCA/Be	0.67	0.67	0.31	1.00	0.00	15.90

a. Total Fault Tolerance = $\sum_{i=4}^{8} I_i W_i$, ranges from 0 to 30.

For helium-cooled designs, the main issues are reactive metals (Li/He/V, Li/He/HT-9), cooling transients (FLIBE/He, Li_2O/He, $LiAlO_2$/He), and breeder container breaks (all). Other than chemical reactions, no very serious flaws appear likely. The capacity of these designs to survive (a) pod depressurization, (b) breeder depressurization, and (c) breeder container failure would have to be verified for a specific design. The chemical reaction risk of beryllium in the FLIBE and $LiAlO_2$ cases seems low because the beryllium is used as bulk pieces rather than powder.

The main problems with Nitrate Salt involve either salt decomposition with pressure buildup or salt chemical reactivity with beryllium or 17Li83Pb. It has been assumed that the salt will not undergo severe pressure buildup with a modest temperature increase. It would not be acceptable if a mild (∿100°C) temperature increase above operating conditions caused high pressure increases. The data in Reference 57 suggest that this is the case. Similarly, although beryllium pieces do not appear to react quickly with the salt, it is feared that beryllium powder would be more reactive. The viability of the 17Li83Pb/NS concept depends on demonstration that large-scale mixtures of these materials can produce only mild consequences. This has been observed for gram-scale quantities but thermodynamic considerations and the observation that the reaction goes to completion are not causes for optimism.

The main problem with water-cooled designs involve its high pressure and two-phase nature. Although the tube break problem has been largely solved by design, emergency cooling is needed for loss-of-power transients because of the difficulty of keeping the water from flashing to steam in the coolant lines.

Clearly, the fault tolerance results depend on several assumptions; almost all deal with chemical reactions, either their seriousness or the practicality of design solutions. Table 5.4.3-25 lists some key potential serious flaws for blankets in the area of fault tolerance. These have already been mentioned, and they represent the key uncertainties in the area of fault tolerance. All blankets except Li_2O/He, $LiAlO_2$/He, and FLIBE/He appear at least once on the list.

TABLE 5.4.3-25. LIST OF POTENTIAL VERY SERIOUS FLAWS IN THE AREA OF FAULT TOLERANCE

Issue	Blanket Type	Comment
Lithium-air fires	Li/Li, Li/He	Nitrogen atmosphere assumed to solve problem
V15Cr5Ti-air reaction	Li/Li/V, LiPb/LiPb/V, Li/He/V	Nitrogen or carbon dioxide atmosphere assumed to solve problem
Lithium-water reactions	Li/Li, Li/He	Assumed reduced by design, especially use of non-water-cooled limiters
V15Cr5Ti-water reactions	Li/Li/V, LiPb/LiPb/V, Li/He/V	Assumed reduced by design, especially non-water-cooled limiters
17Li83Pb-water reactions	LiPb/LiPb/V, LiPb/LiPb/HT-9	Assumes steam generator tube break does not propagate
Nitrate Salt decomposition and/or pressure increase	All NS blankets	Assume salt sufficiently stable to avoid problem
Salt-beryllium reactions	$LiAlO_2$/NS/Be	Assume reaction mild
Beryllium-air reactions	$LiAlO_2$/NS/Be, $LiAlO_2/H_2O$/Be	Assume probability reduced because of carbon dioxide building atmosphere
Seismic qualification	17Li83Pb	Assumed do-able

5.4.4 Normal Radioactivity Effluents, Index 9

The preceding sections include the parts of the safety evaluation associated with accident risks. The remaining sections are associated with expected behavior and corresponding risks: normal radioactivity effluents (considered here), occupational exposure (Subsection 5.4.5.1), and waste management (Subsection 5.4.5.2). Normal operation of a fusion facility is expected to result in a small amount of radioactive effluents being released to the environment. The subject of this index is a comparison among blanket concepts on that basis. Several sources and types of effluents are conceivable, but only one, tritium loss through the steam generator, has received much attention in this study.

5.4.4.1 Sources of Effluents

A summary of the potential sources of effluents is given in Table 5.4.4-1. The basic types of radionuclides of interest are atmosphere activation products, tritium, and coolant activity. These are discussed in more detail in the following subsections.

5.4.4.1.1 Atmosphere Activation Effluent Source

The building gases that were considered in this study were air, carbon dioxide, nitrogen, and helium. There was inadequate time for all the ramifications of cover gas choice to be considered. Some aspects of the choice are indicated in Subsection 5.4.3.6 where the choice influences the possible severity of various chemical reactions. Because the STARFIRE[2] and MARS[3,4] studies selected carbon dioxide as the cover gas, it was adopted in this study, except for the blankets using liquid lithium or 17Li83Pb. Experimental results indicate that the Li-CO_2 reaction is more violent than lithium-air. The same is now known to be true for 17Li83Pb reactions with carbon dioxide and air; carbon dioxide is worse.[65] Thus, some other gas than air or carbon dioxide should be selected for liquid lithium and 17Li83Pb blankets to prevent fires. Because of the cost and perceived difficulty of using helium as the cover gas, nitrogen was selected.

TABLE 5.4.4-1. POTENTIAL SOURCES OF EFFLUENTS

Source	Included in BCSS Safety Evaluation	Comments
Atmosphere Activation Products	NO	Designer choices, N_2 and CO_2, do not entail significant risk to public
Tritium permeation/leakage to steam loop via steam generator	YES	Very difficult to limit, sufficient information available for analysis
Tritium permeation/leakage into the building, ultimately to the environment	SOME	Very design detail dependent, beyond scope of this study; two barriers to release
Coolant activity leakage to steam loop via steam generator	NO	Appears serious mainly for water-cooled systems
Coolant activity leakage into the building, ultimately to the environment	NO	Very design detail dependent, beyond scope of this study; two barriers to release
Coolant activity losses from Coolant Processing	SOME	Appears particularly a problem for Nitrate Salt

It should be noted that it is not an absolute requirement that nonair gases be used. Even if such gases are used to cover more vulnerable components, there is no requirement for the nonair gas to be used throughout the building. The selections of carbon dioxide (solid breeder and FLIBE blankets) and nitrogen (lithium and 17Li83Pb blankets) had the advantages of (a) being consistent with the more recent Tokamak[2] and mirror design[3,4] studies, (b) imposing similar economic and practicality constraints among the blanket concepts, (c) largely removing fires as a safety concern, and (d) avoiding significant atmosphere activation problems, the subject relevant here.

The public doses from building activation can be estimated for some isotopes from past studies. For the case of nitrogen cover gas in STARFIRE, the predominant concern is the ^{14}N (n,p) ^{14}C reaction. However, if all ^{14}C produced is released, the site boundary dose commitment is of the order of nrem/year (see, e.g. Reference 7), i.e. totally insignificant. As the STARFIRE report[2] points out, if the nitrogen is not slowly vented continuously during operation, then the ^{14}C level would reach the MPC-worker level after about five years.

The possible production of ^{14}C from carbon dioxide gas is orders of magnitude less than from air or nitrogen. The production of ^{16}N from air or carbon dioxide and venting of the building would lead to releases of order 1-10 mCi/yr;[7] however, the short half-life (7.13s) of ^{16}N would lead to dose commitments of order nrem/yr or less, totally insignificant.

Thus, use of either nitrogen or carbon dioxide cover gases <u>with STARFIRE-type reactor shielding</u> should not lead to significant public exposures from atmosphere activation; therefore, the building atmosphere activation source term is not relevant to the BCSS.

5.4.4.1.2 <u>Tritium Effluent Source</u>

No possible fusion reactor effluent has received more attention in the fusion program than tritium. Adequate tritium control has long been a major concern. However, much more information is needed to predict with confidence the complete control of tritium effluents. The most recent study of the various tritium effluents is the GEIS technical report,[7] which divided sources into these general areas: (a) plasma, vacuum, and fueling systems, (b) blanket tritium removal system, (c) building atmosphere cleanup and waste handling, and (d) permeation and leakage from the coolant. These are summarized in Table 5.4.4-2 and detailed below.

The upper bound, conservative limits for the tritium release rate for the plasma, vacuum, and fueling systems in the GEIS was 10 Ci/day. A lower bound was not estimated. For BCSS purposes, this category is not relevant because the magnitude of releases is blanket independent, not important to the blanket comparison.

The tritium effluents from the blanket tritium removal systems appear to be smaller, less than 1 Ci/day.[7] Significantly, this was the result even though the GEIS considered both the $LiAlO_2/H_2O$/PCA and Li/He/316SS cases, i.e. both liquid and solid breeders. These are two completely

TABLE 5.4.4-2 POSSIBLE TRITIUM EFFLUENT SOURCES

Components	Upper Bound Release Estimate (Ci/d)	Comment
plasma, vacuum, fueling systems	10[a]	blanket-independent, release value could be significantly lower
blanket tritium removal system	1[a]	
building atmosphere cleanup, waste handling	1[a]	
permeation and leakage from coolant	100[b]	could be the most difficult to solve in the long run; most blanket-dependent

a. Estimates from GEIS, Reference 7.

b. Maximum allowable value in BCSS for transfer to steam generator tritium Report.[1]

separate types of systems. All solid breeders, $LiAlO_2$ and Li_2O, are assumed to use a helium purge stream to carry tritium from the blanket to the tritium processing equipment. The 17Li83Pb self-cooled design concept uses a similar approach, a helium purge stream to carry tritium from the 17Li83Pb coolant to the tritium processing equipment. The other liquid breeders, elemental lithium and FLIBE, use completely different approaches. Because (a) the details of the exact processing scheme for the various breeders are somewhat undefined, (b) the releases are probably small, near 1 Ci/day, (c) existing studies (the GEIS) do not show a breeder dependence, and (d) BCSS project resources did not allow much design of the breeder processing systems, it was decided not to include the tritium breeder processing pathway in the comparison.

The GEIS also indicated that the category of <u>building atmosphere cleanup processing and waste material handling</u> should contribute a tritium effluent of under 1 Ci/day. Dependence on blanket material choice could come from two sources: (a) differences in leakage from blanket systems to the building atmosphere and (b) differences in the tritium content and volatility in waste material generated from the various blanket systems. The waste material pathway was not considered further in the BCSS because the effluent contribution appears fairly small, because there was not an adequate figure-of-merit to compare concepts, and because there were not resources available to try to determine the dependence on blanket material choice, if any. The atmosphere cleanup pathway would seem to scale directly as the load on the atmosphere cleanup unit, which in turn would scale as the leakage from tritium-related systems into the building. Blanket choice influence on the amount of tritium leakage into the building stems from either the breeder choice or the coolant system choice. As discussed above, the breeder system does not seem to have a strong impact. The influence of the coolant system on the amount of tritium getting into the building is relevant and discussed below.

Given the above discussion among the possible tritium effluent sources, the potentially largest and most blanket-dependent effluent appears to be <u>permeation and leakage from the coolant</u>. The effluent rate could easily be as high as the 100 Ci/day value adopted as a screening value in the early stages of the BCSS. This release category can be subdivided for further consideration.

Coolant-related releases could come from either the permeation or leakage mechanisms. Either of these could result in tritium going from the coolant to either the building atmosphere or the steam cycle (sodium intermediate loop in the case of the lithium self-cooled design), see Table 5.4.4-3. Because the GEIS only considered two coolants, water and helium, and much of the relevant data have been updated since then, the GEIS discussions are less relevant or valuable to this comparison than they were in the preceding discussions for noncoolant effluent sources.

For the case of the permeation release mechanism, the actual losses should depend on the surface area and the possibility of downstream side tritium recovery, once the coolant, coolant wall material (external to the blanket), and associated tritium inventory are fixed. The area in the steam generator appears at least an order of magnitude higher than the

TABLE 5.4.4-3. TRITIUM PERMEATION AND LEAKAGE FROM COOLANT

Release Mechanism	Downstream Environment	Comments
Permeation	Building atmosphere	Should be less important than permeation to steam cycle because (a) lower surface area involved, (b) released tritium could be recovered via building atmosphere processing, (c) more flexibility in adding permeation barriers.
Permeation	Steam cycle, via steam generator	Likely a significant release pathway.
Leakage	Building atmosphere, via valves, pumps, etc.	Likely a significant release pathway; however, released tritium could be recovered via building atmosphere processing.
Leakage	Steam cycle, via steam generator	Possibly a significant release pathway, especially for water coolant.

coolant wall area exposed to the building. Also, since nonsteam-generator walls are not heat transfer surfaces, one has much more flexibility to add barriers, if needed, to reduce permeation losses. The GEIS estimates that 99% to 99.9% of the tritium released to the building can be recovered by building atmosphere processing,[7] or even secondary enclosures. The traditional view is that tritium that gets into the steam side is lost to the environment,[7] which is also the case for CANDU reactors.[96, 97] Although this view is assumed here, an important area of future research would be to see if there were cost-effective means to prevent some of the tritium in the steam cycle from ultimately reaching the environment. For the above reasons, permeation into the building should be several orders of magnitude less important than permeation in the steam generator (lithium-to-sodium heat exchanger in the case of the lithium self-cooled design), and is not further considered here.

Thus, the one permeation pathway considered in some detail in the BCSS is the permeation from coolant to steam cycle. Adequate control of this pathway was not easy in the study for most of the various design teams, confirming that it is an important area of comparison.

The <u>leakage mechanism</u> is not so easily discussed as permeation. Although some estimates are possible for the leakage from water and helium systems, the lack of any operating experience with the liquid metal or molten salt coolants makes those cases very difficult to evaluate. The GEIS goes into the most detail for the case of pressurized water--"typical leakage rates through LWR steam generator tubes may be approximately 45 kg/d." They go on to note that fusion reactors will probably have to do better because of tritium concerns. Thus they state, "it appears reasonable to assume leakage rates near the lower part of the range estimated for current and projected near-term CANDU reactors, approximately 10 kg/d."[7] Just as for permeation, leakage across the steam generator into the steam side is more important, per amount lost, than that leaked into the building.

The EPRI technical risk assessment[98] came to similar conclusions and adopted 24 kg/day-GWe as the reference value for water leakage into the steam generator. This value is used for the BCSS. It should be noted that primary water coolant would also be lost via leakage directly to the building atmosphere. Scaling the results from CANDU[96, 97] experience and the INTOR discussions,[24,25] leads to a value of ~200 kg/day-GWe from primary coolant to building. The latter pathway would be less important to atmospheric effluents since most would be recovered by the building atmospheric cleanup system.

For the case of helium-cooled designs, HTGR experience and analysis are adopted. The base case is that 1% of the helium coolant will leak into the building each year.[99] Standard design practice of using clean helium back-fill gases with valves and circulator seals is responsible for this low value. Thus as valves leak, clean helium at higher pressure leaks <u>into</u> the helium coolant. In similar fashion, because the steam pressure (8.3 MPa) is higher than the helium coolant pressure (5.2 MPa), leakage at the steam generator will be into the helium. It should be noted that the GEIS[7] assumed 1% helium loss per year into the steam generator, rather than into the building. This appears to be an incorrect citing of Reference 99.

Unfortunately, there are not GEIS estimates for liquid metal or salt systems. However, these should be very small. In all these cases, the pressure gradient is from a high pressure steam cycle to lower primary coolant pressure. In the case of lithium, tritium has to leak or permeate into the sodium intermediate loop and then leak or permeate into the steam side. Because of the sodium-water reaction, that steam generator is designed for extremely low leakage. The LMFBR design called for double wall tubing with leak detection. Thus the tritium leakage pathway should be small, at least relative to permeation.

The Nitrate Salt case has the least experience. There is no chemical reaction between the salt and water to motivate very low leakage steam generators. However, there is a strong motivation due to tritium and activation products in the salt. Fortunately, the total driving pressure gradient is very much against salt (0.4 MPa) leaking into the steam (10 MPa). For BCSS purposes, no salt-to-steam leakage will be assumed.

In summary, the coolant leakage is somewhat uncertain. The tritium leakage loss from the water coolant to the environment is taken as 10 kg of coolant per day. 200 kg/day-GWe of water is assumed lost to the building atmosphere. The leakage loss from the helium coolant to the building is 1% of the helium inventory per year. The liquid metal and salt coolants were assumed to have insignificant leakage losses, relative to permeation.

5.4.4.1.3 Coolant/Breeder Activity Effluent Source

Coolant or breeder activity include both the presence of activated corrosion products as well as activation of the coolant material itself. Such coolant activity could be the source of radioactive effluents in several ways: (a) leakage from primary coolant to the steam generator, (b) leakage from primary coolant to the building atmosphere whereupon a small fraction would escape from the building, or (c) effluent stream from coolant processing systems. Similar items are relevant for breeder fluid activity. It is possible that nontritium effluents from the coolant or breeder could be more radiologically significant than tritium effluents, as was mentioned in Reference 7. Three key items must be known to adequately estimate coolant/breeder activity effluents; (a) the steady state activity levels, by isotope and chemical form, in coolants and breeder fluids, (b) location and amounts of leakage, and (c) the details of coolant and breeder processing systems down to the level of precise chemistry control. Severe data gaps exist for these items for most of the fluids in this study. Knowing steady state activity levels requires adequate prediction of corrosion/sputtering rates, transport, deposition, and processing systems. The details of most coolant or breeder fluid systems are sparse. Of all the fluids in this study, meaningful estimates of nontritium effluents could only be made for helium and water. Thus, it was not possible to directly compare blanket concepts on the basis of coolant/breeder radioactive effluents. However, some indirect comparisons have been included.

First, the steady state activity levels in fluids have been estimated and are a basis for comparison in indices 2,3,10, and 11, source term characterization, occupational exposure, and waste management. Thus a major aspect of comparing nontritium effluents among blankets has already been explicitly included in the safety evaluation. Second, the concepts that would be expected to be the best from this activation product effluent aspect, helium-cooled blankets, already appear top safety candidates in the evaluation. Third, the concept that would be expected to be the worst, Nitrate Salt, is explicitly treated as a special case in the effluent evaluation, Index 9, and has been given a zero score (Subsection 5.4.4.3.3). Overall, except for the special case of Nitrate Salt, it is not felt that leaving out nontritium effluents has a strong impact on the final safety evaluation. Given the knowledge available, it is felt that the effect, if any, on the safety rankings if one could include nontritium effluents in the comparison would be to (a) further increase the safety attractiveness of helium designs and (b) decrease the attractiveness of lithium-cooled steel and 17Li83Pb-cooled concepts.

5.4.4.2 Design Philosophy

The design philosophy for controlling effluents that was adopted during the early stages of the study was not particularly useful. "Concept design should minimize normal tritium losses to the environment. Concepts requiring fewer tritium control barriers/systems are preferable because of the higher assurance of success and higher flexibility in meeting possible future standards."[1] Although this design suggestion was initially written for the case of tritium, it applies equally well to the other sources of effluents.

The problem with the above statement from the interim report is that there is inadequate guidance on how the designer should accomplish this minimization. The early guidance did, however, provide numerical targets. The screening value for acceptance of alternate blanket concepts was 100 Ci/day. The initial utility function for determining the relative value was established such that concepts releasing over 10 Ci/day were given zero points in the comparative evaluation. As discussed in the following section, the utility function was later relaxed.

5.4.4.3 Evaluation Scheme

As discussed above, it was decided to evaluate blankets on the basis of tritium loss through the steam generator, with the one special case of activity from the processing of Nitrate Salt. Thus, there are three aspects to the evaluation scheme, detailed in the following subsections: appropriate tritium release limits, the utility function to actually compare concepts, and how to handle the case of Nitrate Salt.

5.4.4.3.1 Tritium Guidelines for Evaluation

Any design guidance or limits on tritium effluents from a fusion reactor are necessarily speculative. It is not known what future standards might exist. Neither is it known just how well releases will be controlled or at what cost. Ideally, the eventual standards will themselves be a function of the cost versus benefit of controlling tritium releases.

On the other hand, it has long been acknowledged by the fusion community that adequate tritium control will be required. Since the blanket choice can have so much influence on the amounts, mobility, partial pressures, and chemical forms of tritium, blankets have to be compared on the basis of difficulty of controlling releases. Thus, designers need some guidance concerning what level of releases is "good" and what level is "bad."

The original intent of the design philosophy also included the idea that concepts requiring fewer tritium control barriers and systems are preferable because of the higher assurance of success and higher flexibility in meeting possible future standards. However, the concept designs have all evolved to the point where multiple control features are necessary. In the concepts with separate coolant and breeder, both the tritium breeder helium purge system and the coolant must be processed to remove tritium. The lithium self-cooled concept requires both the primary and the intermediate sodium coolant loops to be processed for tritium removal. The 17Li83Pb self-cooled concepts require processing of the entire coolant stream as it leaves the blanket. All designs also depend on various oxide barriers. Thus, the difficulty of controlling tritium has led all the design teams to use multiple barriers and control systems. Therefore, the relevant comparison among blankets in the area of control of tritium loss into the steam generator is simply the rate of loss.

The question then is how well should tritium release be controlled. Three approaches are summarized in Table 5.4.4-4. First, a more traditional approach in the U.S. fusion community has been to start with the 5 mrem/yr maximum individual regulatory dose limits from atmospheric releases, imposed on LWRs by the U.S. NRC. Using existing dose calculation codes like TREM[100] and AIRDOS,[101, 102] the maximum individual dose for releases from a 100 m stack and 500 m exclusion zone is about 0.05 mrem/yr per Ci/day released atmospherically. Thus the corresponding regulatory limit would be 100 Ci/day, neglecting allotment for nontritium contribution to dose. Given the desire to design below the regulatory limit and giving some allotment to nontritium contributions to dose, an appropriate design goal is approximately 10 Ci/day of tritium released to the atmosphere. Had a shorter stack of 10 m been used, the regulatory limit would drop from 100 Ci/day to 10 Ci/day.

Second, a Canadian approach could be used. The Canadian regulatory limit is 500 mrem/yr with a design goal of 5 mrem/yr. If one uses their dose experience for releases at their Pickering and Bruce facilities, the average maximum individual dose is about 0.018 mrem/yr per Ci/day released atmospherically. This is one-third less than that calculated for the assumptions of 100 m stack and 500 m exclusion zone. Using their 5 mrem/yr design goal and their dose experience, one could design to release 280 Ci/day to the atmosphere assuming no allotment for nontritium contribution to dose. Using parameters of 100 m stack and 500 m exclusion zone would result in a 100 Ci/day design goal, assuming no allotment for nontritium contribution to dose.

Third, one could use a modified, more up-to-date U.S. approach. NRC limits do not apply to fusion. However, EPA regulations most likely will. The EPA is currently proposing a limit on maximum individual dose from atmospheric radioisotope releases of 10 mrem/yr, it may be raised to

TABLE 5.4.4-4. POSSIBLE TRITIUM RELEASE GUIDELINES

(1) More traditional approach in U.S. fusion community

- Adopt 5 mrem/yr maximum individual regulatory dose limit from atmospheric releases[a]
- Assume all tritium releases are atmospheric
- 100 m stack, 500 m exclusion zone
- Result--100 Ci/day regulatory limit, neglecting nontritium contribution to dose limit[b]
- Result--10 Ci/day approximate design goal for atmospheric tritium effluents[c]

(2) Canadian Approach

- Adopt 500 mrem/yr as maximum individual regulatory dose limit
- Adopt 5 mrem/yr as maximum individual design limit
- Canadian dose experience at their facilities[d]
- Result--280 Ci/day appropriate design goal, assuming no allotment for nontritium contributions
- Result--100 Ci/day appropriate design goal, assuming 100 m stack, 500 m exclusion zone, and no allotment for nontritium contributions to dose

(3) Modified fusion U.S. assumptions

- Adopt 10 mrem/yr maximum individual regulatory dose limit from atmospheric releases[e]
- Assume 75% of tritium is released to atmosphere, 25% to aquatic systems
- 100 m stack, 500 m exclusion zone
- Result--267 Ci/day regulatory limit on coolant/steam generator tritium loss, neglecting nontritium contributions to dose and other tritium contributions to dose
- Result--30 Ci/day design goal for coolant/steam generator tritium loss[f]

---

a. U.S. NRC limit for LWRs is 5 mrem/yr.

b. Based on dose calculations(103,104) using both the TREM(100) and AIRDOS-MIT(102) codes.

c. Design limit should be lower than regulatory limit to allow for nontritium dose contribution as well as uncertainty. Note, a 10 m stack would lead to a 10 Ci/day regulatory limit.

d. Based on References 96, 97.

e. 10 mrem/yr is the current U.S. EPA proposed limit, which may be raised to perhaps 25 mrem/yr, which would raise the design goal to perhaps ~70 Ci/day.

perhaps 25 mrem/yr. Using parameters of 100 m stack and 500 m exclusion zone, the limit can be converted to atmospheric tritium release regulatory limits of 200 Ci/day, without allotment for nontritium contributions to dose. The GEIS included the assumption that of the tritium getting into the steam cycle, 75% would be released atmospherically. Thus the regulatory-based limit for the coolant/steam generator pathway would be 267 Ci/day, without allotment for nontritium dose or noncoolant/steam generator tritium pathways. Considering doses other than tritium from coolant/steam generator pathway, an appropriate design goal would be of order 30 Ci/day to the steam generator.

The preceding discussion indicates that the appropriate design goal for limiting tritium getting into the steam cycle is uncertain, but appears to be in the 10-100 Ci/day range. It is interesting to see if CANDUs would meet the above goals. Normalized to a 1000 MWe size plant, Pickering-A and Bruce-A experience an average of 71.2 Ci/day tritium effluent, 46.5 Ci/day to atmospheric, 24.7 Ci/day to aquatic. Thus, CANDUs would fall in the BCSS acceptable design range.

5.4.4.3.2 Tritium Utility Function for Evaluation

In the BCSS interim report, the initial allowable tritium loss rate via the steam generator was 100 Ci/day. The initial design goal was 10 Ci/day. The preceding discussion indicates that the appropriate design limits for tritium effluents are uncertain. "Good" and "bad" values, however, needed to be established for the BCSS comparison. The upper bound, "bad," for the utility function should be the highest likely design limit for the coolant/steam generator pathway. The lower limit, "good," for the utility function should be the release rate such that the coolant/steam generator pathway is an insignificant contributor to total effluents.

In regards to the upper limit for comparison purposes, several factors should be noted. First, current total tritium releases from CANDUs[96, 97] average 71 Ci/day normalized for 1000 MWe. The discussion in Subsection 5.4.4.3.1 indicated appropriate design limits of some value less than 100 Ci/day via the steam generator pathway. 100 Ci/day is selected as the upper bound.

In regards to the lower bound, it appears that noncoolant tritium effluents could be as high as 12 Ci/day, the sum of noncoolant estimates in Table 5.4.4-2. The lower values for these noncoolant pathways is, of course, not known. Because it is not known how well noncoolant effluents can be controlled, nor how low standards will be, it was deemed appropriate to continue to reward blankets that appear to offer the potential of having "insignificant" coolant/steam generator releases. "Insignificant" in this context means that blankets should be rewarded for showing the potential of coolant/steam generator releases being only 10% of the upper bound estimates for noncoolant tritium releases, 12 Ci/day. Therefore, blankets estimated to release 1.0 Ci/day via the coolant/steam generator should be rewarded.

With 100 Ci/day as the upper bound and 1.0 Ci/day as the lower bound, the appropriate utility function is therefore

$$
\begin{aligned}
I_9 &= 0.0 && R > 100 \text{ Ci/day} \\
&= 0.5 \log (100/R) && 1.0 \text{ Ci/day} < R < 100 \text{ Ci/day} \\
&= 1.0 && R < 1.0 \text{ Ci/day}
\end{aligned}
$$

where R = release rate in Ci/day via steam generator

I_9 = index score for the ninth index

Because the mirror and Tokamak blanket versions are sized to two different reactors, the values of the total power, inventory, and effluent differ. This generally causes the use of two utility functions for a given figure-of-merit, one for the range of mirror versions and one for the range of Tokamak versions. In the case of effluents, however, the utility function is independent of plant size and does not differ between mirror and Tokamak. The reason is that all of the U.S. regulatory guidelines and limits apply to a given site, independent of facility size. A 1200 MWe reactor is not allowed to release twice as much as a 600 MWe reactor.

Whether intended or not, this provides a built-in regulatory bias in favor of smaller reactor sizes.

5.4.4.3.3 Special Case--Nitrate Salt

The above utility function is adequate to compare blankets where the tritium loss from the coolant/steam generator pathway appears the most difficult to control. As discussed above, that assumption was made for all coolants except Nitrate Salt. The corrosion products and radioactivity, other than tritium, in all other coolants are nongaseous. Nitrate Salt activation, however, leads to two key gaseous products, ^{39}Ar and ^{41}Ar. Several other isotopes are also produced, but they do not appear to be in gaseous form.

The calculated inventory of ^{39}Ar and ^{41}Ar in the various breeders, multiplier, and coolant materials is indicated in Table 5.4.4-5. Because

TABLE 5.4.4-5. ARGON ISOTOPE PRODUCTION PER m^3 IN BREEDER, MULTIPLIER, AND COOLANT MATERIALS FOR 2-YEAR IRRADIATION AT THE FIRST WALL

Material	^{39}Ar (269 yr)	^{41}Ar (1.83 h)
Lithium	4.9×10^{1}	2.1×10^{2}
17Li83Pb	1.3×10^{3}	4.1×10^{3}
FLIBE	3.3×10^{-10}	3.6×10^{-5}
Li_2O	3.4×10^{2}	1.1×10^{3}
$LiAlO_2$	5.8×10^{2}	1.7×10^{3}
Beryllium	8.3×10^{-1}	1.6×10^{2}
Nitrate Salt	1.7×10^{5}	4.7×10^{5}
Helium	0	0
Water	0	0

the reference activation calculations include potassium impurity, a source of argon isotopes, most materials have some amount of these isotopes. Because the Nitrate Salt has potassium as a prime constituent, rather then an impurity, it is seen to have two orders of magnitude more of the argon isotopes per volume as any other material. The seriousness of these isotopes was examined via calculation of a required reduction factor, defined as the yearly dose from instantaneous release of all isotopes as they are produced divided by 10 mrem/yr (whole body dose) or 30 mrem/yr (skin dose). For $\sim 55\ m^3$ of Nitrate Salt in the blanket, the required ^{39}Ar reduction factor is 60. The required ^{41}Ar reduction factor is 1.7×10^7. While ^{41}Ar (1.83 h) control can be accomplished by insuring that a 2 day wait occurs before any off-gas is released, the control of ^{39}Ar (269 yr) appears difficult and must be almost 99% efficient.

The next worst case is 17Li83Pb when used as a coolant (360 m^3 in the blanket). The ^{39}Ar required reduction factor is about 3; ^{41}Ar, 10^6. Again ^{41}Ar control only requires a holdup; in this case it is 36 hours. Either a ^{39}Ar capture efficiency of 33% or a 3-fold reduction of potassium impurity in 17Li83Pb is needed. The latter is probably the easier path; 17Li83Pb will not be penalized in the evaluation. The other cases do not appear to have a significant argon problem; however, impurities will have to be kept limited, especially any elements leading to highly releasable gaseous isotopes.

Because capture and retention of argon from the salt, especially from the salt processing system, would appear to be difficult and probably expensive, it appears that activation product control for the sodium-potassium nitrate salts like Nitrate Salt will be a major feasibility concern. In the spirit of the 3CSS comparative evaluation, the Nitrate Salt appears the worst case from the operational effluent viewpoint and thus merits a zero score for the normal radioactivity effluent comparison, Index 9.

5.4.4.4 Index 9 Results

The expected tritium effluent for the designs has been calculated with a consistent set of ground rules, detailed in Subsection 6.6. The most important of these are as follows: (a) all tritium being released from solid breeders is in the highly mobile T_2 form, (b) tritium conversion kinetics from T_2 to T_2O in helium breeder purge streams and helium coolants is two slow to be of much help, (c) all oxidized HT-9 surfaces have a factor of 100 lower tritium permeation than that calculated classically ("oxide barrier factor" = 100), and (d) tritium going through oxidized metal is not oxidized. It is interesting to note that the tritium control analysis for HTGRs[99] gave no credit for tritium potentially being in the oxide form, identical to the assumptions in this study.

It is very important to note that (a) these assumptions are far less favorable for solid breeders than assumed in past design studies, (b) these assumptions are based on the most current information and technical understanding, (c) more data and understanding is still needed, and (d) the net effect of the high calculated tritium effluents for most of the concepts (including $LiAlO_2$/He, $LiAlO_2/H_2O$, Li_2O/He, FLIBE/He, and 17Li83Pb/He) provides a very strong advantage for the Li/He and Li/Li concepts.

Given the high importance (20%) of the effluent evaluation and the many uncertainties, special mention is needed of the assumptions, and their impacts, for the various designs. The reference tritium loss rates and corresponding index scores are given in Tables 5.4.4-6 and 5.4.4-7.

The amount of tritium entering the lithium self-cooled blankets is controlled by the lithium breeder itself rather than an external source. There is no separate breeder to be a source of tritium getting into the coolant. The calculated tritium flux through a vanadium wall is 162 g/day (mirror) and 168 g/day (Tokamak). The HT-9 values are almost 3 orders of magnitude lower, about 0.2 g/day. In neither case does the first wall flux appear important. This is indeed fortunate for the Li/Li/V blanket, removing one of the main disadvantages of vanadium versus steels, its higher permeability. Because both Li/Li/V and Li/Li/HT-9 are assumed to use a steel heat exchanger, there is no permeability problem for vanadium there either. Thus, although V is several orders of magnitude more permeable than HT-9, the estimated tritium loss rate for Li/Li/V and

TABLE 5.4.4-6. INDEX SCORES FOR NORMAL RADIOACTIVITY EFFLUENTS, INDEX 9, FOR MIRROR BLANKETS

Blanket	Tritium Steam Generator Loss Rate, Ci/day	Index Score
Li/Li/V15Cr5Ti	2	0.85
Li/Li/HT-9	2	0.85
17Li83Pb/17Li83Pb/V15Cr5Ti	10	0.50
17Li83Pb/17Li83Pb/HT-9	10	0.50[a]
Li/He/V15Cr5Ti	high?	0.00[a]
Li/He/HT-9	10.2	0.50
17Li83Pb/He/HT-9	high?	0.00[a]
FLIBE/He/HT-9/Be	191[b]	0.00
Li_2O/He/HT-9	18	0.37
$LiAlO_2$/He/HT-9/Be	41	0.19
$LiAlO_2$/He/PCA/Be	41	0.19[a]
17Li83Pb/NS/HT-9	0(?)[c]	0.00[a]
$LiAlO_2$/NS/HT-9/Be	0(?)[c]	0.00
$LiAlO_2$/NS/PCA/Be	0(?)[c]	0.00[a]
$LiAlO_2$/H_2O/HT-9/Be	59	0.11
$LiAlO_2$/H_2O/PCA/Be	59	0.11[a]

a. Modification cases for which tritium losses were not calculated, values and scores used here were based on extrapolations of other designs.

b. High value partially caused by renormalization of tritium generation rate late in the study.

c. Tritium effluent may be nil, but ^{39}Ar control appears sufficiently difficult to warrant a zero score for this index.

TABLE 5.4.4-7. INDEX SCORES FOR NORMAL RADIOACTIVITY EFFLUENTS, INDEX 9, FOR TOKAMAK BLANKETS

Blanket	Tritium Steam Generator Loss Rate, Ci/day	Index Score
Li/Li/V15Cr5Ti	3	0.76
Li/Li/HT-9	3	0.76[a]
17Li83Pb/17Li83Pb/V15Cr5Ti	13	0.44[a]
17Li83Pb/17Li83Pb/HT-9	13	0.4[illegible][a]
Li/He/V15Cr5Ti	high?	0.00[a]
Li/He/HT-9	16.1	0.40
17Li83Pb/He/HT-9	high?	0.00[a]
FLIBE/He/HT-9/Be	260[b]	0.00
Li_2O/He/HT-9	24	0.31
$LiAlO_2$/He/HT-9/Be	44	0.18
$LiAlO_2$/He/PCA/Be	44	0.18[a]
17Li83Pb/NS/HT-9	0(?)[c]	0.00[a]
$LiAlO_2$/NS/HT-9/Be	0(?)[c]	0.00
$LiAlO_2$/NS/PCA/Be	0(?)[c]	0.00[a]
$LiAlO_2$/H_2O/HT-9/Be	76	0.06
$LiAlO_2$/H_2O/PCA/Be	76	0.06[a]

a. Modification cases for which tritium losses were not calculated, values and scores used here were based on extrapolations of other designs.

b. High value partially caused by renormalization of tritium generation rate late in the study.

c. Tritium effluent may be nil, but ^{39}Ar control appears significantly difficult to warrant a zero score for this index

Li/Li/HT-9 are the same. The lithium-cooled designs use an intermediate coolant, sodium, providing another stage to remove tritium. Overall, because the tritium in lithium has a low partial pressure and because an intermediate loop is used, the tritium effluent from either a Li/Li/V or Li/Li/HT-9 blanket should be quite small, 2-3 Ci/day in the reference cases, and should be known with a good degree of confidence. If necessary the numbers could probably be lowered.

The 17Li83Pb-cooled cases are qualitatively similar to the lithium-cooled cases with two exceptions. First, there is no intermediate loop providing another stage to remove tritium. Second, the tritium solubility is lower so that tritium partial pressures are harder to control. Both differences mean that the removal of tritium from the primary coolant is far more difficult then lithium. The result is that the entire 17Li83Pb primary coolant goes through a helium-purged counter-flow separator to strip tritium from the liquid metal. Some heroic scheme like this appears to be required for 17Li83Pb, and its viability is an uncertainty. If viable, the concept looks attractive from the effluent standpoint. If not viable, the concept may not be acceptable. As for the lithium-cooled designs, the 17Li83Pb designs would not show a dependence on the blanket structural material between V15Cr5Ti and HT-9.

The helium-cooled designs divide into three types. The tritium entering the helium coolant in Li/He blanket primarily comes from the first wall, none from the breeder. The tritium entering the helium coolant in the LiPb/He and Li_2O/He cases has significant amounts from both first wall and breeder zone. The $LiAlO_2$/He and FLIBE/He concepts may also have tritium from beryllium entering the coolant.

For the reference designs, it appears that the tritium pressure in the Li/He breeder can be kept low enough so that tritium flows from the helium into the breeder. The tritium source is the first wall. The calculated tritium fluxes through the first wall indicated above are assumed the same for all other blankets. The Li/He/HT-9 uses hydrogen addition in the coolant to dilute tritium and reduce the effluent. The loss of oxide layers on the coolant side is acceptable because (a) it allows tritium to more readily flow into the breeder lithium and (b) the steam-side oxide layer in the steam generator is not affected so that the total steam generator is reduced only from 200 to 100 (none on coolant side) while the oxide layer on the breeder tubes is reduced to perhaps only 2. If tritium could be readily oxidized in the helium coolant, hydrogen addition would not be used and would not be needed; the effluent would drop well below 1 Ci/day. Future research for all helium designs should include some way to oxidize tritium before it gets to the steam generator so that it will not permeate, assuming oxidized surfaces. On the negative side, a substantially higher first wall tritium flux could pose problems. For example a Li/He/V design could have tritium control problems. Both the tritium flux through the first wall and the tritium flux into the breeder would increase. However, unless the tritium partial pressure in the lithium breeder were able to be correspondingly reduced, the new equilibrium tritium pressure in the coolant would be substantially higher, increasing tritium loss into the steam generator.

The Li_2O/He and 17Li83Pb/He cases would have tritium in the coolant from both first wall and breeder zone. Present understanding of solid breeders indicates that much of the tritium is released from the solid breeder in elemental form, contrary to past studies. Furthermore, it is now believed that hydrogen addition to the purge stream is needed to assist tritium atoms on the solid breeder surface to form HT molecules and leave the surface. Otherwise, kinetics limitations appear to cause a surface inventory buildup. For those two reasons, the tritium in all solid breeder cases is assumed to be 100% in the elemental form with no oxide barriers on the breeder side. Coolant-side oxide layers are still expected. Both the Li_2O/He and 17Li83Pb/He cases would be dramatically helped if an on-line tritium catalyst for the helium coolant or improved permeation barriers were developed. Reducing the tritium from the breeder zone does not solve the effluent by itself since the first wall source is still present. Either blanket would be hurt if the first wall flux were found to be substantially higher. The best defense for either first wall or breeder zone uncertainties would be a tritium oxidizer in the helium coolant or a special tritium barrier at the steam generator.

The $LiAlO_2$/He/Be and FLIBE/He/Be cases are similar to the preceding two cases. The $LiAO_2$ case is very similar to Li_2O in having the same solid breeder issues. FLIBE, which follows Henry's law rather than Sievert's, is similar to 17Li83Pb in that the tritium solubility in the breeder is very low and associated partial pressure very high. Even with special, untested tritium barriers in the FLIBE/He design, its effluent rate is still near 200 Ci/day, probably unacceptable. Besides the uncertainties for the Li_2O/He and 17Li83Pb/He cases, the $LiAlO_2$/He/Be and FLIBE/He/Be cases have one more, the presence of unclad beryllium directly in the helium coolant. Late in the study it was realized that tritium is generated in the beryllium. Time and data uncertainties did not permit estimation of resulting inventory in the beryllium or how much tritium will enter the coolant. Either could be large. Thus the $LiAlO_2$/He/Be and FLIBE/He/Be designs have three possible sources of tritium entering the coolant: first wall, breeder zone, and beryllium. If beryllium is found to be a problem, design fixes may be possible; for example, put beryllium in the $LiAlO_2$ breeder plates.

It should be noted that one area where PCA may have an advantage over HT-9 is in tritium control. As explained in Section 6.6, the permeability of HT-9 is higher than PCA but the "oxide barrier factor" also appears to be higher so that the net permeation through oxidized HT-9 and oxidized PCA would be similar. Even if this is true for normal permeation, it may not be true for implantation-driven permeation through the first wall. For those concepts were first wall flux is a driver, PCA may help versus HT-9.

The biggest problem for Nitrate Salt is ^{39}Ar, not tritium. In fact, tritium control might be fairly easy in the salt. If tritium entering the salt from either the breeder zone or first wall oxidizes before the salt reaches the steam generator (few seconds), then the tritium effluent should be quite small, perhaps $\sim$1 Ci/day. Beryllium is not an issue here since it is in the breeder zone. Rapid oxidation of tritium in water coolant is also assumed. However, significant water-to-steam leakage is expected whereas salt-to-steam leakage is not possible because of the pressure differential. If tritium oxidation in the salt is not sufficiently rapid, then tritium control would be progressively more difficult.

The bigger potential improvement for Nitrate Salt would be a replacement for potassium, eliminating the argon problem.

The water-cooled designs benefit from the expected assumed rapid oxidation of tritium once it enters the coolant from either the first wall or breeder zone. As with the salt, beryllium as a source of tritium is not relevant since it is already in the breeder zone, not in contact with the coolant. It is assumed that the economic penalty will be paid to keep tritium concentration in the water to 1 Ci/liter. The reference 24 kg/day-GWe water-to-steam loss rate means that substantial tritium leaks to the steam side. Because hydrogen is added to the water for corrosion/water chemistry control, there is an equilibrium amount of HT in the water, which permeates. For the reference design, the water-to-steam permeation was calculated at about 2 Ci/day.

A generic problem with many of the designs may be to actually operate the designs as intended. In theory, steel corrosion in PWRs can be kept very low. However, water chemistry control is tricky and touchy. Actual operation sometimes leads to corrosion increasing by an order of magnitude as chemistry slips from optimum or even something is left inside piping during maintenance. Similarly, actual operation of some of these blankets to provide tritium control and recovery may also be difficult. The solid breeder cases will depend on some purge stream composition. An air or water leak could cause tritium inventory buildup on the solid breeder surface. Excess hydrogen in a helium coolant could reduce oxide layers on walls.

Finally, two alternatives to the reference effluent scores are proposed. First, if one optimistically believes new tritium control ideas will be found and workable, e.g. tritium barriers or tritium oxidizers; or that tritium release limits are too strict, then all designs should get a perfect index score of 1.00. Second, if one pessimistically believes that some of the assumptions were not conservative enough, then most designs have a severe effluent problem. Examples of possible problems include (a) an order of magnitude or more increase in first wall permeation, (b) oxide barriers not as good (100 for HT-9) as assumed, or (c) the hydrogen dilution trick does not work. In these cases, the Li/Li concept still appears able to control tritium and would get a perfect index score of 1.0. There is some reason to believe that Li/He might be able to demonstrate tritium control; besides, it is less susceptible to the uncertainties mentioned for the remaining designs. Li/He/HT-9 would be scored 0.0, but not tagged as having a fatal flaw. All other designs might find it difficult to meet public acceptance because of tritium control and would be scored 0 for the effluent evaluation. The impact of these optimistic and pessimistic cases will be discussed with the final results.

5.4.5 Occupational Exposure and Waste Management

The preceding sections of the safety evaluation have dealt with accident risks and operational radioactivity releases. Additional risks to society are compared here, i.e. occupational exposure and waste management risks.

The radiological and chemical toxicity of fusion plants will lead to some level of worker exposure. As seen below, that level can reasonably be expected to be a function of the blanket concept choice. The occupational exposure comparison is designed to examine that influence.

The radiological and chemical toxicity of fusion plants will ultimately have to be either disposed of or reprocessed. Risks associated with waste management include transportation of material to the disposal/reprocessing site (if not at the reactor), exposure during disposal/reprocessing, release of additional radioisotopes or chemical toxins during disposal processing/reprocessing, risks from waste once buried, and exposure from reprocessed material.

The complexity of occupational exposure and waste management risk assessments is well beyond the scope of this study. Rather, existing studies, like the Generic Environmental Impact Study[7-9] and MARS,[3-4, 105] have been used to identify appropriate bases of comparison among the blanket options. These are discussed in the following subsections.

5.4.5.1 Occupational Exposure, Index 10

Several factors relevant to occupational exposure were indicated in the early stages of the study and were mentioned in the BCSS interim report.[1] The basic design philosophy was that

> "Work force radiation exposure should be minimized through increased component/blanket reliability (fewer maintenance operations needed), increased blanket simplicity (easier and faster maintenance), lower radiation fields, increased use of remote maintenance techniques, and lower mobility of radioactive and toxic chemical contaminants."

Unfortunately, time and manpower resources of design teams to address occupational concerns was severely limited.

5.4.5.1.1 Sources of Exposure

The number of factors relevant to the comparison was narrowed from those in the interim report by adoption of the basic analysis in the Generic Environmental Impact Study (GEIS)[7-9] and by making some key simplifying assumptions, see Table 5.4.5-1.

A significant area in a complete evaluation of occupational exposure would be the number and duration of needed maintenance operations. This would be affected by such factors as blanket simplicity, blanket reliability, module lifetime, and the time actually needed for the required maintenance operations. These are not well established for fusion concepts at this time. It was deemed beyond the scope of this study to identify (a) all the possible needed types of maintenance and the time required for each, or (b) the probability of different fault conditions that might require maintenance. Within this area, the only quantifiable issue was the module lifetime. The qualitative aspects of blanket simplicity/reliability are accounted for in the engineering feasibility evaluation. Likewise, the qualitative nature of potentially difficult maintenance operations, e.g. cleanup of a 17Li83Pb spill, are accounted for in the fault tolerance part of the safety evaluation. These various qualitative aspects are not repeated in the occupational exposure evaluation.

TABLE 5.4.5-1. FACTORS POTENTIALLY RELEVANT TO OCCUPATIONAL EXPOSURE

Factor[a]	Considered in BCSS Safety Evaluation?	Comment
Blanket simplicity, reliability	No	Difficult to quantify. Blanket reliability indirectly included in fault tolerance evaluation, where the difficult types of maintenance after accidents are considered. Considered somewhat in engineering feasibility evaluation
Module lifetime or number of sector changeouts	Yes	
Time required for maintenance	No	Cannot quantify in this study. Indirectly considered in engineering feasibility evaluation
Degree of remote maintenance	No	Actual degree would be a tradeoff between occupational exposure and economics. Since this is not considered in economics evaluation, the degree of remote maintenance is assumed the same for all concepts (see text)
Activity/dose levels, near serviced components	Yes	
Total tritium Inventory	Yes	
Amounts of operationally generated wastes	No	GEIS(7) indicates it is a small, 5-7%, contributor

a. Indicated in the BCSS interim report.

A second key area is the degree of remote maintenance. In principle, occupational exposure could be reduced to zero if human workers were not near any of the maintenance procedures. On the other hand, remote operation has an associated economic cost. Thus, the degree of remote maintenance in a fusion reactor would be determined by a trade-off between the economic cost and the human occupational exposure costs. Insufficient prior analysis has been done to know that tradeoff, and the resources available in this study are too limited to allow that calculation here. However, to first order, one does not need to know the degree of remote maintenance to compare blanket concepts. In a sense, increasing the degree of remote maintenance simply transfers an occupational exposure penalty to an economic one. Thus, the penalty due to increased occupational exposure potential of a blanket concept cannot be eliminated, merely transferred. Because the economic evaluation in this study did not consider remote maintenance penalities among blankets, the appropriate comparison among blankets is to include the penalty of increased occupational exposure potential as occupational exposure. Therefore, the degree of remote maintenance techniques was assumed fixed among concepts.

A third major area was the actual potential for radiation exposure, i.e., the activity/dose levels near serviced components and the amount of tritium present. These are considered in the evaluation in the basic procedure as in the GEIS, coolant piping and component (blanket) changeout.

A fourth area of comparison is the occupational exposure caused by processing and handling of operationally-generated wastes. Various components will become contaminated during reactor lifetime, e.g. resin beds, pumps, filters, pump oils. These will cause occupational exposure from either their servicing or disposal. This area was not explicitly considered in the evaluation because (a) the GEIS found that it contributes only a small amount, 5-7%, of exposure at PWRs, (b) it could not be calculated at the present time for fusion, and (c) it is indirectly accounted for in the matter of the activity levels near coolant piping and blankets.

Having limited the large number of occupational exposure issues to a more manageable value, it is possible to specify the precise details of the comparison.

5.4.5.1.2 Evaluation Scheme

The basic approach in this study is to follow the approach in the GEIS study in terms of identifying figures-of-merit. In this spirit, the evaluation is divided into coolant system exposure, blanket changeout exposure, and tritium exposure, Table 5.4.5-2.

The discussion in the GEIS indicates that the key determinant in coolant system exposure is the corrosion product deposition levels in the coolant piping and steam generator. Only gamma emitters are relevant. Likewise, the presence of beryllium chemical toxicity in the $LiAlO_2$ designs should not be a major factor. The basis of comparison is the radiation field around the coolant piping. Since it would be desired to do much coolant system maintenance quickly, the evaluation is conducted assuming one does not take the time to drain the coolant first. The activation calculations, Subsection 6.12, determine the Remote Maintenance Rating (RMR) for blanket materials as defined by Maninger and Dorn[105] for MARS. "The RMR is defined as the radiation dose rate at the surface of a uniformly activated, think, infinite slab with the same composition and density as the specific machine component."[105] For occupational exposure purposes, the RMR is evaluated at one day after shutdown, constant for all blankets. As is the case for the issue of the degree of remote maintenance, the exposure for a specific blanket concept design could be reduced by waiting a longer period of time before maintenance, but this increases reactor downtime and increases costs. Since variation of reactor downtime among concepts due to maintenance is not considered in the economic evaluation, it is assumed fixed for the safety evaluation.

For the case of coolant system exposure, the RMR (units of mrem/h) of the coolant is used. The radiation field caused by corrosion products is added to the RMR.

TABLE 5.4.5-2. SCORING SCHEME FOR OCCUPATIONAL EXPOSURE, INDEX 10

Coolant and Steam Generator Exposure (one-third of the index)

Basis:	F_{T1}	$\equiv$	Radiation field around coolant piping
	F_{T1}	=	RMR_c
	RMR_c	=	Remote Maintenance Rating for the coolant, evaluated at 1 day after shutdown

Blanket Changeout Exposure (one-third of the index)

Basis:	F_{T2}	$\equiv$	The number of sector changeouts over the reactor lifetime
	F_{T2}	=	$40/L_b$
	L_b	=	Blanket lifetime, based on structure lifetime only

Tritium-Related Exposure (one-third of the index)

Basis:	F_{T3}	$\equiv$	Total blanket and coolant tritium inventory

Index Score

$$I_{10}(TMR) = 1.3951 - 0.01927 \ln(F_{T1}) - (0.0325)F_{T2} - 0.0558 \ln(F_{T3})$$

$$I_{10}(TOK) = 1.4727 - 0.01927 \ln(F_{T1}) - (0.0302)F_{T2} - 0.0595 \ln(F_{T3})$$

The second general area is the exposure associated with major reactor system maintenance and changeout. For BCSS purposes where matters are restricted to blankets, this translated to blanket changeout exposure. The basis for comparison is the number of sector changeouts per year over the blanket lifetime, the inverse of the blanket module lifetime. The radiation field quantified by the RMR for the structural material did not vary significantly over the metals at 1 day after shutdown. Because lifetimes vary between mirrors and Tokomak concepts, the figure-of-merit and associated utility function differ between the mirror evaluation and Tokamak evaluation.

The third comparison area is tritium-related exposure. Much of the maintenance on blanket components will involve tritium. Remote or fully protected maintenance operations would not lead to significant tritium exposure. However, CANDU reactor experience[96, 97] indicates that in an actual operating reactor there will be tritium-related exposure. The problem for this study is how to compare blanket concepts. The selected approach was to use the total blanket tritium inventory, which is calculable, on the judgment that the more tritium that is present, the more exposure that will eventually be caused by it.

5.4.5.1.3 Index 10 Results

The first step in this evaluation was the Remote Maintenance Ratings (RMR) for various materials at 1 day after shutdown. For the case of the RMR for fluids the potential impact of corrosion had to be investigated. As indicated in Index 3, it does not appear that corrosion or sputtering is a significant contributor, except for water and helium. The results depends on the assumed impurity levels and corrosion rates and could change if either set of numbers changed. For maintenance exposure, the most significant case was Li/Li with Modified ("low-activation") HT-9 where the RMR at 1 day increased by 7% when corrosion products were included.

For helium, there are direct estimates of the contact dose at various parts of a helium circuit.[55] The dose rate was found to vary from 5 to 10,000 mrem/h depending on location with about 100 mrem/h at the steam generator.[55] Since the steam generator is a key maintenance issue in the area of coolant exposure, the helium cases were evaluated using an RMR of 100.

For water, the estimates of corrosion product mass deposited around the loop[5] were combined with the RMR for HT-9 and PCA to get an equivalent RMR for the water systems. The resulting RMR values for various materials are listed in Table 5.4.5-3.

The values deserve close inspection. Whereas it is true that low activation structural materials considered in the study do not make "hands-on" maintenance of the blanket possible, proper coolant choice does offer the possibility of "hands-on" maintenance of the coolant system! The same appears true for the breeder system. Hands-on maintenance of

TABLE 5.4.5-3. REMOTE MAINTENANCE RATINGS (mrem/h) FOR VARIOUS MATERIALS AT 1 DAY AFTER A SHUTDOWN

Coolants	Reference RMR	"Low Activation" Steel RMR
Lithium	5.9×10^6	same
17Li83Pb	8.9×10^7	same
Nitrate Salt	3.3×10^9	same
Water - PCA	5.1×10^5	∿same
- HT-9	3.2×10^5	∿same
Helium	1×10^2	∿same
Structures		
V15Cr5Ti	1.4×10^{10}	--
HT-9	4.2×10^9	∿same
PCA	1.2×10^{10}	1.3×10^{10}
Other		
Beryllium	1.7×10^8	--
Li_2O	5.5×10^6	--
$LiAlO_2$	2.8×10^{10}	--
FLIBE	1.7×10^6	--

blanket-related systems that are actually out of the blanket area appears a worthwhile goal. Selection of helium appears to be the way to achieve this goal. It should be noted that because an intermediate loop is used with the lithium-cooled designs, these have nil activation products in the steam generator. The lithium-cooled designs do have a highly activated intermediate heat exchanger.

The other parameters in the occupational exposure evaluation, blanket lifetime and total tritium inventory, were simply obtained from other efforts in the study.

The resulting figures-of-merit and Index 10 scores are given in Tables 5.4.5-4 and 5.4.5-5. In general, the helium-cooled concepts are the most attractive. The 17Li83Pb concepts are penalized both because of their high RMR value but also their low lifetime. The 17Li83Pb blankets have about a 20% higher dpa rate causing the lower lifetime. However, the 17Li83Pb blankets have a low tritium inventory, demonstrating one of the many trade-offs.

5.4.5.2 Waste Management, Index 11

Several factors relevant to the risks associated with waste management were identified early in the study and indicated in the interim report. The corresponding design philosophy was that "radioactive waste generated from normal operation, replacement of blanket components, and decommissioning should be minimized." Identified factors included the volume, volatility, biological hazard, radiation exposure from waste processing, waste class per 10CFR61, total tritium inventory, and total radioactivity produced over the reactor lifetime. As in the case of the occupational exposure evaluation, project resources necessitated the limitation of the evaluation to a reasonably small number of factors.

5.4.5.2.1 Types of Waste Management Risks

The number of factors relevant to the comparison was narrowed from those in the interim report by generally adopting the methodology of Maninger and Dorn for MARS,[105] see Table 5.4.5-6.

A very visible issue in the fusion community has been whether activated material would meet the requirements for near-surface burial, according to the spirit of existing legal requirements, 10CFR61. This is one of the parts of the BCSS comparison.

A second important issue is the exposure that might be caused by processing of the waste, either from processing it for disposal or for reprocessing it into a usable product. This is another part of the BCSS comparison.

TABLE 5.4.5-4. INDEX SCORES FOR OCCUPATIONAL EXPOSURE, INDEX 10, FOR MIRROR BLANKETS[a]

Blanket	F_{T1} Coolant Exposure	F_{T2} Blanket Changeouts[b]	F_{T3} Total Tritium (g)	Index Score
Li/Li/V15Cr5Ti	5.9×10^6	7.1	337	0.54
Li/Li/HT-9	5.9×10^6	8.9	340	0.49
17Li83Pb/17Li83Pb/V15Cr5Ti	8.9×10^7	12.1	6	0.55
17Li83Pb/17Li83Pb/HT-9	8.9×10^7	14.3	4	0.50
Li/He/V15Cr5Ti	1×10^2	7.3	236	0.77
Li/He/HT-9	1×10^2	8.7	237	0.72
17Li83Pb/He/HT-9	1×10^2	13.8	7	0.75
FLIBE/He/HT-9/Be	1×10^2	9.3	143	0.72
Li_2O/He/HT-9	1×10^2	8.9	137	0.75
$LiAlO_2$/He/HT-9/Be	1×10^2	9.1	30	0.82
$LiAlO_2$/He/PCA/Be	1×10^2	17.4	30	0.55
17Li83Pb/NS/HT-9	3.3×10^9	13.8	507	0.18
$LiAlO_2$/NS/HT-9/Be	3.3×10^9	8.9	788	0.32
$LiAlO_2$/NS/PCA/Be	3.3×10^9	16.7	788	0.06
$LiAlO_2$/H_2O/HT-9/Be	3.2×10^5	8.2	1571	0.48
$LiAlO_2$/H_2O/PCA/Be	5.1×10^5	15.4	1571	0.24

a. Scoring scheme and figures-of-merit detailed in Table 5.4.5-2.

b. Blanket lifetime only based on structure lifetime as used throughout the study.

TABLE 4.4.5-5. INDEX SCORES FOR OCCUPATIONAL EXPOSURE, INDEX 10, FOR TOKAMAK BLANKETS[a]

Blanket	F_{T1} Coolant Exposure	F_{T2} Blanket Changeouts[b]	F_{T3} Total Tritium (g)	Index Score
Li/Li/V15Cr5Ti	5.9×10^6	8.0	496	0.56
Li/Li/HT-9	5.9×10^6	9.8	509	0.51
17Li83Pb/17Li83Pb/V15Cr5Ti	8.9×10^7	13.3	11	0.57
17Li83Pb/17Li83Pb/HT-9	8.9×10^7	15.4	19	0.48
Li/He/V15Cr5Ti	1×10^2	8.3	338	0.78
Li/He/HT-9	1×10^2	10.0	349	0.73
17Li83Pb/He/HT-9	1×10^2	16.0	23	0.71
FLIBE/He/HT-9/Be	1×10^2	10.5	208	0.75
Li_2O/He/HT-9	1×10^2	10.0	155	0.78
$LiAlO_2$/He/HT-9/Be	1×10^2	9.5	59	0.85
$LiAlO_2$/He/PCA/Be	1×10^2	18.2	59	0.59
17Li83Pb/NS/HT-9	3.3×10^9	16.0	923	0.16
$LiAlO_2$/NS/HT-9/Be	3.3×10^9	10.0	2970	0.27
$LiAlO_2$/NS/PCA/Be	3.3×10^9	19.1	2970	0.00
$LiAlO_2$/H_2O/HT-9/Be	3.2×10^5	8.9	2378	0.50
$LiAlO_2$/H_2O/PCA/Be	5.1×10^5	16.7	2378	0.25

a. Scoring scheme and figures-of-merit detailed in Table 5.4.5-2.

b. Blanket lifetime only based on structure lifetime as used throughout the study.

TABLE 5.4.5-6. FACTORS POTENTIALLY RELEVANT TO WASTE MANAGEMENT COMPARISON

Factor[a]	Considered in BCSS Safety Evaluation?	Comment
Volume of waste	Yes	
Volatility of waste	No	Insufficient data
Biological hazard of waste	No	Indirectly accounted for in the 10CFR61 evaluation
Waste class per 10CFR61	Yes	In form of the dilution required to meet Class C requirements
Total tritium Inventory	No	Included in occupational exposure
Radiation exposure from waste processing	Yes	Judged via Remote Maintenance Rating after 10 years
Total radioactivity produced	Yes	Judged via volume generated and dilution required to meet Class C

a. listed in the interim report.

The third important issue being considered is the volume of generated waste. This influences the sheer magnitude of the waste management problem from processing requirements to land required for burial to the number and size of transport shipments.

The other factors mentioned in the interim report were not considered in the final evaluation. The total tritium inventory was not deemed directly relevant to waste management. Tritium is sufficiently valuable that all possible tritium would be recovered from used components before taking them from the reactor. The residual amounts of tritium are not known. Furthermore, the total tritium inventory is already considered under occupational exposure. The volatility of waste was not explicitly considered. Tritium is the obvious high volatility case, but for comparison purposes the unknown residual amounts of tritium are what is important. In general, there was not sufficient data to compare waste volatility. Finally, possible measures of strict biological hazard potential of the waste were not used, in favor of the more relevant 10CFR61-class and processing-dose figures-of-merit.

5.4.5.2.2 Evaluation Scheme

The basic approach in this study has been mentioned above and is listed in Table 5.4.5-7. The three parts are the dilution to meet near-surface burial requirements, the radiation field for waste processing, and the volume of generated wastes. These are now discussed in more detail.

A very worthwhile goal of the fusion program is to allow all fusion wastes to be considered as low-level wastes, available for near-surface burial. This would avoid the need for deep geological burial. The present legal requirements are set in 10CFR61.[27] However, 10CFR61 does not include limits on several potentially important fusion isotopes. Rather than assume that those isotopes will remain unlimited for waste disposal, the approach taken for MARS[105] and adopted here is that such isotopes will likely be bound by concentration limits based on 10CFR61 methodology. Thus, appropriate concentration limits were established, see Subsection 6.12. Given sufficient dilution, any of the fusion isotopes could meet 10CFR61, including tritium and ^{39}Ar, although the later would have to be limited to 100Ci per unpressurized container.[27]

A possible evaluation scheme would have been to simply state whether a blanket concept's materials meet or do not meet 10CFR61. However, this gets into some philosophical problems associated with the issue of dilution. Since 10CFR61 is based on concentration and not on the absolute amount, any waste could, in principle, be made to meet 10CFR61 by diluting the waste to the 10CFR61 concentration limit. How regulatory agencies will view this is not known, but at some unknown point such a dilution technique would run into severe practical limitations, e.g., land available for the near-surface burial. On the other hand, the regulatory limits could change, up or down, in the future. To avoid these issues, the comparison was based on the dilution that would be required to meet 10CFR61 rather

TABLE 5.4.5-7. SCORING SCHEME FOR WASTE MANAGEMENT, INDEX 11

Dilution Required to Meet 10CFR61-Type Limits for Near-Surface Disposal (one-third of the index)

Basis: F_{E1} $\equiv$ Waste Disposal Rating ($\overline{WDR}$) at 10 years after shutdown, where

$$F_{E1} = \overline{WDR} = \sum_c \left(\frac{V_c}{V_{TOTAL}}\right) WDR_c = \sum_c \frac{V_c}{V_{TOTAL}}\left(\sum_i \frac{A_c(i)}{L(i)}\right)$$

V_c = volume of material in blanket

V_{TOTAL} = total blanket volume

$A_c(i)$ = Specific activity of isotope i in material c

$L(i)$ = Concentration limit for near-surface burial for isotope i

Radiation Field for Waste Processing (one-third of the index)

Basis: F_{E2} $\equiv$ Remote Maintenance Rating ($\overline{RMR}$) at 10 years after shutdown

$$F_{E2} = \overline{RMR} = \sum_c \left(\frac{V_c}{V_{TOTAL}}\right) RMR_c \text{ (10 yr)}$$

RMR_c = RMR for blanket material c, evaluated 10 years after shutdown

Processing and Transportation Risks (one-third of the index)

Basis: F_{E3} $\equiv$ Total volume of waste material generated over reactor lifetime that must either be disposed of or reprocessed

$$F_{E3} = \sum_c V_c \left(\frac{L_{TOTAL}}{L_c}\right)$$

L_{TOTAL} = reactor lifetime = 40 yr.

L_c = material replacement lifetime = blanket lifetime for solids = 40 yr. for fluids.

Index Score

$$I_{11}(TMR) = 1.658 - (0.0517) \ln(F_{E1}) - (0.0472) \ln(F_{E2}) - (7.81\times10^{-5})\, F_{E3}$$

$$I_{11}(TOK) = 1.635 - (0.0535) \ln(F_{E1}) - (0.0482) \ln(F_{E2}) - (5.75\times10^{-5})\, F_{E3}$$

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than a yes/no evaluation for meeting 10CFR61. This Waste Disposal Rating (WDR) was established by Maninger and Dorn[105] and is defined by:

$$WDR_c = \sum_i A_c(i)/L(i)$$

where

WDR_c = waste disposal rating for material c

$A(i)$ = specific activity of isotope i for material c

$L(i)$ = 10CFR61 type concentration limit of isotope i.

For BCSS purposes, the specific activity is volumed averaged over the blanket. The figure-of-merit is therefore given by

$$F_{E1} = \overline{WDR} = \sum_i \frac{\overline{A(i)}}{L(i)} = \sum_c \left(\sum_i \frac{A(i)}{L(i)}\right)$$

$$= \sum_c \left(\frac{V_c}{V_{TOTAL}}\right)\left(\sum_i \frac{A(i)}{L(i)}\right) = \sum_c \left(\frac{V_c}{V_{TOTAL}}\right) WDR_c$$

where

$\overline{WDR}$, $\overline{A(i)}$ are blanket averaged values

V_c = volume of material c in blanket

V_{TOTAL} = total volume of blanket, not including gases.

Beryllium chemical toxicity is not included in this comparison. For blanket concepts using beryllium, resource limits will require recycle and not burial. Some small amounts of beryllium will be lost from the processing system as wastage. Such wastage and beryllium-contaminated equipment will ultimately be disposed of. The future requirements for beryllium disposal and the amounts to be disposed are unknown. However, it is unlikely that beryllium would require geological burial on the basis of chemical toxicity, and thus its use should not influence the WDR.

The second basis for comparison in the relative exposure involved in processing waste, either for burial or for reprocessing. Maninger and Dorn[105] established the Remote Maintenance Rating, defined in Subsection 5.4.5.1, for that purpose. For BCSS purposes, the RMR for waste considerations is evaluated at 10 years after shutdown. Ten years after shutdown is a typical amount of time one might wait before commencing processing.

For BCSS purposes, the RMR is averaged over the blanket. The figure-of-merit is therefore given by

$$F_{E2} = \overline{RMR} = \sum_c \left(\frac{V_c}{V_{TOTAL}}\right) RMR_c \ (10 \text{ years})$$

where

$\overline{RMR}$ = RMR averaged over the blanket

RMR_c = RMR for material c

Beryllium will be sufficiently radioactive from impurities that workers could not come into direct contact with the beryllium[7], hence its chemical toxicity would not be an issue. Processing and fabrication of beryllium before its first use in a reactor would entail some chemical toxicity exposure. This issue was deemed beyond the scope of the safety evaluation. A cost penalty is included in beryllium costs in the economic evaluation.

The third basis for comparison is the volume of generated waste. This measure influences the land necessary for disposal, the size of processing facilities, and the number and size of transportation shipments, hence transportation risk. The waste volume should be integrated over the reactor lifetime. The volume of solids that must be processed is the blanket volume times the number of blanket changeouts over the reactor lifetime, assumed to be 40 years. Liquids, however, do not have to be processed just because a blanket sector is changed. In fact, one would probably add a small amount of new liquid as needed and continuously cleanup the liquid in service. Thus, the liquid volume to be processed for waste management is to first order given by the total liquid volume in service at any single time. The figure-of-merit is given by

$$F_{E3} = \sum_c V_c \left(\frac{L_{TOTAL}}{L_c}\right)$$

where

L_{TOTAL}	=	reactor lifetime = 40 years
L_c	=	material replacement lifetime, blanket lifetime for solids, 40 yrs for fluids

Late in the study, an alternative scheme to the preceding three figures of merit was identified that directly integrates waste volume with WDR and RMR. As defined above, F_{E1} $(\overline{WDR})$ is the average dilution required for 10CFR61 burial and F_{E3} is the total volume of blanket material to be disposed of, before dilution. The actual amount of waste volume buried or processed (after dilution) is a possible replacement figure-of-merit and is given by

$$F_E \text{ (alt 1)} = \sum_c V_c \, WDR_c \left(\frac{L_{TOTAL}}{L_c}\right)$$

In similar fashion, the $\overline{RMR}$ (F_{E2}) can be integrated with the waste volume (F_{E3}) to be processed. This possible replacement figure-of-merit is given by

$$F_E \text{ (alt 2)} = \sum_c \left(\frac{V_c}{V_{TOTAL}}\right) (RMR_c) \left(\frac{L_{TOTAL}}{L_c}\right)$$

This has units of mrem/h for undiluted material times the number of replacement operations for that material over the reactor lifetime. The impact of using these alternative figures-of-merit is mentioned below.

5.4.5.2.3 Index 11 Results

The first stage in the waste disposal evaluation is the Waste Disposal Ratings (WDR) and Remote Maintenance Ratings (RMR) at 10 years after shutdown. These are listed in Table 5.4.5-8. The "low-activation steels" are seen to be successful in dramatically improving the WDR versus the reference steels to the point of being near V15Cr5Ti. Although they improve the RMR, they do not achieve the RMR level of V15Cr5Ti.

Several of the other materials appear to be waste disposal problems, 17Li83Pb, $LiAlO_2$, and Nitrate Salt. $LiAlO_2$ was a surprise. 99% of the WDR for $LiAlO_2$ is from the presence of ^{26}Al, which poses two issues. First, the activation calculations are for first wall exposure. This is thought to work to the detriment of $LiAlO_2$ because the ^{26}Al production is expected to fall off dramatically further into the blanket. Second, the high WDR for ^{26}Al depends on a fairly restrictive 10CRF61-type concentration limit that was calculated for this study, see Section 6.12. The finding that 17Li83Pb (Pb) and $LiAlO_2$ (Al) do not meet 10CFR61 has been confirmed by another recent study.[106]

The other inputs to the waste management evaluation are the volumes of the various materials and the blanket lifetimes, both are detailed with the Economic Evaluation. Given these data, the calculation of the three figures-of-merit for the various blankets is straight forward. These and the corresponding Index 11 scores are listed in Tables 5.4.5-9 and 5.4.5-10.

As expected the top blankets are the two V15Cr5Ti concepts, Li/Li/V15Cr5Ti and Li/He/V15Cr5Ti, each scoring above 0.90. The LiPb/LiPb/V concept did much less well, about 0.40, for three reasons connected to the difference between lithium and 17Li83Pb. First, although the Li-V combinations have a volume-averaged $\overline{WDR}$ under 1.0, the only concepts to do so, the LiPb/LiPb/V blanket has a $\overline{WDR}$ of about 13,

TABLE 5.4.5-8. WASTE DISPOSAL RATINGS AND REMOTE MAINTENANCE RATINGS FOR BLANKET MATERIALS AT 10 YEARS AFTER SHUTDOWN

Material	WDR	RMR
V15Cr5Ti	1.7	6.5×10^{3}
HT-9	484.7[a]	4.4×10^{7}
Modified HT-9	1.4	1.1×10^{7}
PCA	197.2	2.3×10^{8}
Tenelon	1.5	1.4×10^{7}
Lithium	$\sim$0	9.1×10^{4}
17Li83Pb	24.4	4.2×10^{6}
FLIBE	0.1	2.8×10^{4}
Li_2O	0.3	4.4×10^{4}
$LiAlO_2$	150.5	5.4×10^{6}
Beryllium	0.4	1.3×10^{6}
Nitrate Salt	71.0	4.6×10^{7}
Water	$\sim$0	$\sim$0
Helium	$\sim$0	$\sim$0

a. Niobium content in HT-9 designs is taken to be 0.11 weight percent.

TABLE 5.4.5-9. INDEX SCORES FOR WASTE MANAGEMENT, INDEX 11, FOR MIRROR BLANKETS

Blanket	F_{E1} DWR[a]	F_{E2} RWR[b]	F_{E3} Waste Volume, m^3	Index Score
Li/Li/V15Cr5Ti	0.87	4.8×10^4	2950	0.92
Li/Li/HT-9[c]	248.2	2.3×10^7	3500	0.30
17Li83Pb/17Li83Pb/V15Cr5Ti	12.6	2.0×10^6	5720	0.39
17Li83Pb/17Li83Pb/HT-9	262.4	2.5×10^7	6540	0.06
Li/He/V15Cr5Ti	0.58	6.2×10^4	2270	0.98
Li/He/HT-9	165.8	1.5×10^7	2620	0.41
17Li83Pb/He/HT-9	200.7	2.0×10^7	4540	0.24
FLIBE/He/HT-9/Be	367.4	3.4×10^7	3880	0.23
Li_2O/He/HT-9	100.6	9.2×10^6	3440	0.40
$LiAlO_2$/He/HT-9/Be	208.9	1.4×10^7	2600	0.41
$LiAlO_2$/He/PCA/Be	143.9	5.6×10^7	4970	0.17
17Li83Pb/NS/HT-9	266.2	3.5×10^7	2280	0.37
$LiAlO_2$/NS/HT-9/Be	148.8	1.4×10^7	3280	0.37
$LiAlO_2$/NS/PCA/Be	120.4	3.2×10^7	5770	0.15
$LiAlO_2$/H_2O/HT-9/Be	156.5	1.1×10^7	3490	0.35
$LiAlO_2$/H_2O/PCA/Be	95.8	5.1×10^7	6080	0.11

a. Waste Disposal Rating, averaged over blanket materials, see Table 5.4.5-7

b. Remote Maintenance Rating, averaged over blanket material volumes, see Table 5.4.5-7

c. Niobium content of 0.11 weight percent assumed for HT-9 designs

TABLE 5.4.5-10. INDEX SCORES FOR WASTE MANAGEMENT, INDEX 11, FOR TOKAMAK BLANKETS

Blanket	F_{E1} DWR[a]	F_{E2} RWR[b]	F_{E3} Waste Volume, m^3	Index Score
Li/Li/V15Cr5Ti	0.78	5.3×10^4	3130	0.95
Li/Li/HT-9[c]	221.5	2.0×10^7	3630	0.33
17Li83Pb/17Li83Pb/V15Cr5Ti	14.0	2.3×10^6	5910	0.45
17Li83Pb/17Li83Pb/HT-9	234.8	2.2×10^7	6620	0.15
Li/He/V15Cr5Ti	0.67	5.8×10^4	2310	0.99
Li/He/HT-9	191.0	1.7×10^7	2700	0.39
17Li83Pb/He/HT-9	226.0	2.2×10^7	4850	0.25
FLIBE/He/HT-9/Be	339.9	3.1×10^7	4040	0.26
Li_2O/He/HT-9	170.3	1.6×10^7	2940	0.40
$LiAlO_2$/He/HT-9/Be	205.0	1.3×10^7	4250	0.32
$LiAlO_2$/He/PCA/Be	148.0	5.0×10^7	8100	0.04
17Li83Pb/NS/HT-9	266.2	3.5×10^7	2600	0.35
$LiAlO_2$/NS/HT-9/Be	149.0	1.4×10^7	4440	0.31
$LiAlO_2$/NS/PCA/Be	117.7	3.5×10^7	7880	0.09
$LiAlO_2$/H_2O/HT-9/Be	157.1	1.2×10^7	3640	0.37
$LiAlO_2$/H_2O/PCA/Be	93.6	5.3×10^7	6390	0.17

a. Waste Disposal Rating, averaged over blanket materials, see Table 5.4.5-7

b. Remote Maintenance Rating, averaged over blanket material volumes, see Table 5.4.5-7

c. Niobium content of 0.11 weight percent assumed for HT-9 designs dominated by 17Li83Pb, a nonlow-activation material. Second, the $\overline{RMR}$ for LiPb/V is two orders of magnitude higher than for Li-V concepts, almost as bad as the various steel concepts. Third, the greater density of 17Li83Pb requires increased structure, and the higher dpa rate in a 17Li83Pb blanket causes a lower blanket lifetime. The combination causes almost a doubling of the waste volume for LiPb/V versus Li/V.

After the V-concepts, all other blankets score a maximum of about 0.40, Li/He/HT-9, Li_2O/He/HT-9, and $LiAlO_2$/He/HT-9/Be. Among HT-9 concepts the first two score best because the breeder does not add much to the structural activity WDR and RMR. $LiAlO_2$/He/HT-9/Be demonstrates a higher WDR because of ^{26}Al in $LiAlO_2$, but this is somewhat compensated by the lower blanket thickness, i.e. less waste, than for Li/He/HT-9 and Li_2O/He/HT-9.

The lowest scoring blankets are the PCA versions but not because of the WDR. In fact, as seen in Table 5.4.5-8, the WDR for HT-9, 485, is higher than for PCA, 197, caused by the assumed niobium content in HT-9. Niobium (^{94}Nb) is also the limiting element in V15Cr5Ti. The HT-9 advantage over PCA comes instead from its lower RMR by a factor of 5 and its longer lifetime by a factor of 2. The latter leads to less waste generation.

It should be noted that without the waste-volume third of the waste management evaluation the steel concepts would score even lower, increasing the differential between V15Cr5Ti and the steels. This is also the net result of integrating the impact of waste volume generation directly with WDR and RMR as detailed in Subsection 5.4.5.2.2. The use of the two replacement figures-of-merit, where WDR is combined with the waste volume to obtain the total amount of waste buried or processed (after dilution) and the RMR is similarly integrated with waste volume, gives an alternative waste management score, listed in Table 5.4.5-11 along with the reference-type scores for "low-activation" steel blankets.

Use of the alternative scoring scheme increases the V15Cr5Ti to steel advantage, largely because the basic V-alloy advantage in WDR and RMR is

TABLE 5.4.5-11. IMPACT OF USING LOW-ACTIVATION STEELS OR THE ALTERNATE SCORING SCHEME ON THE WASTE MANAGEMENT EVALUATION FOR MIRROR BLANKETS

	Reference Materials	Low Activation Steels	Reference Materials
Blanket	Reference Score	Reference Score	Alternative Score
Li/Li/V	0.92	--	0.98
Li/Li/HT-9	0.30	0.67	0.13
17Li83Pb/17Li83Pb/V15Cr5Ti	0.39	--	0.70
17Li83Pb/17Li83Pb/HT-9	0.06	0.26	0.06
Li/He/V15Cr5Ti	0.98	--	0.99
Li/He/HT-9	0.41	0.77	0.17
17Li83Pb/He/HT-9	0.24	0.42	0.09
FLIBE/He/HT-9/Be	0.23	0.60	0.10
Li_2O/He/HT-9	0.40	0.72	0.28
$LiAlO_2$/He/HT-9/Be	0.41	0.48	0.23
$LiAlO_2$/He/PCA/Be	0.17	0.28	0.10
17Li83Pb/NS/HT-9	0.37	0.52	0.16
$LiAlO_2$/NS/HT-9/Be	0.37	0.40	0.26
$LiAlO_2$/NS/PCA/Be	0.15	0.21	0.14
$LiAlO_2$/H_2O/HT-9/Be	0.35	0.46	0.22
$LiAlO_2$/H_2O/PCA/Be	0.11	0.25	0.10

more heavily weighted. Use of "low-activation" steels decreases the V15Cr5Ti to steel advantage by raising the steel scores. The steel concept improvement is a strong function of breeder and coolant. The best improvement, about 0.37 (worth 3.7 total SFM points), is for the Li/HT-9 combinations because the breeder does not hurt the improved WDR. The Li_2O/He improvement, 0.32, is slightly lower because of the slightly higher WDR for Li_2O. The lowest improvement when "low-activation" steels are used is 0.03 (worth 0.3 SFM points) for the $LiAlO_2$/NS/HT-9/Be cases because both the Nitrate Salt and $LiAlO_2$ have high WDRs themselves. This illustrates the fact that blankets with 17Li83Pb, Nitrate Salt, and $LiAlO_2$ (assuming ^{26}Al) are not low-activation waste, even if the structural material did not activate at all. For those concepts, then, it is not surprising that the improvement from reference steels to low-activation steels is dampened.

5.4.5.3 Summary

The results for the reference materials have already been given in Tables 5.4.5-4, 5.4.5-5, 5.4.5-9, and 5.4.5-10. Overall, the V15Cr5Ti concepts have an advantage over the steels, particularly in the area of waste management. Also the helium-cooled concepts have advantages over others in the area of occupational exposure. For these issues of occupational exposure and waste management, a Li/He/V15Cr5Ti concept would be best.

The impact of using "low-activation" steels has been examined earlier in the structure source term characterization, Index 1, and fault tolerance to cooling transients, Index 5. In both cases the use of "low-activation" steels was seen to be either very small or negative. The net impact of "low-activation" steels on occupational exposure also appears small, as was anticipated. The Panel on Low-Activation Materials[10] indicated that hands-on maintenance of the blankets did not appear to be achievable and the current results agree. Although the "low-activation" steels have not helped in the preceding list of activation issues, accidents and maintenance, their intent was to help achieve low-activation status for waste disposal. This has been achieved, although it is tempered in some cases by breeder or coolant materials that are not low-activation, especially 17Li83Pb and Nitrate Salt.

Overall, the uncertainties are too large to say that "low-activation" steels are a net improvement over reference steels in the safety evaluation. However, it appears that Modified HT-9 would be a net improvement over HT-9. It also appears that Tenelon would not be a net improvement over PCA. These matters will be further discussed in Subsection 5.4.6. A summary of figures-of-merit relevant to low activation steels in the areas of maintenance and waste management is listed in Table 5.4.5-12.

Although hands-on maintenance of the blanket itself does not appear likely, limited hands-on maintenance of other systems might be. The tritium purge/processing system will be contaminated by either the breeder or breeder purge stream. Selection of a proper fluid could lead to hands-on maintenance. A helium purge stream probably qualifies. 17Li83Pb would certainly not. The steam generator is likely to require substantial maintenance, but will be contaminated by the coolant. Selection of a helium coolant could allow limited hands-on maintenance. Because the lithium self-cooled designs have an intermediate loop, the steam generator in those concepts is unactivated. However, the primary-to-intermediate heat exchanger would be activated.

5.4.6 Results and Conclusions

The primary purpose of the Safety Evaluation has been to compare the safety and environmental attractiveness of various blanket designs given the best available information and technical judgment. Subsection 5.4.1 is a brief description of the basic methodology. Subsections 5.4.2 to 5.4.5 contain technical discussions of how the blankets compare in four major areas: accident source term characterization, accident fault tolerance, effluents, and maintenance and waste management. This Subsection (5.4.6) is a compilation of the overall results (5.4.6.1) with discussion of results with respect to individual materials (5.4.6.2) and with respect to

TABLE 5.4.5-12. IMPACT OF USING LOW ACTIVATION STEELS ON OCCUPATIONAL EXPOSURE AND WASTE MANAGEMENT FIGURES-OF-MERIT

Blanket Changeout, RMR (structure, 1 day)	
V15Cr5Ti	1.36×10^{10} mrem/h
HT-9	4.17×10^{9}
Modified HT-9	4.24×10^{9}
Change from HT-9	2% worse
PCA	1.16×10^{10}
Tenelon	1.33×10^{10}
Change from PCA	15% worse
Dilution Required to meet 10CFR61, WDR for Structure Only	
V15Cr5Ti	1.7
HT-9	484.7
Modified HT-9	1.4
Change from HT-9	Factor of 346 better
PCA	197.2
Tenelon	1.5
Change from PCA	Factor of 131 better
Radiation Field for Waste Processing, RMR (10 years), for Structure Only	
V15Cr5Ti	6.53×10^{3}
HT-9	4.41×10^{7}
Modified HT-9	1.11×10^{7}
Change from HT-9	Factor of 4 better
PCA	2.30×10^{8}
Tenelon	1.39×10^{7}
Change from PCA	Factor of 17 better

specific blankets (5.4.6.3). Overall conclusions are discussed in 5.4.6.4. Readers who desire a briefer description of results and conclusions are referred to Subsection 3.3.

5.4.6.1 Safety Rankings and Key Sensitivities.

Each blanket has been given 11 individual index scores (I_i) by virtue of quantified figures-of-merit (76% of SFM) and/or engineering judgment (24% of SFM). The individual index score can range from 0 to 1. These have been discussed and listed in previous subsections. The total Safety Figure-of-merit (SFM) is a weighted (W_i) sum of these indices:

$$SFM = \sum_{i=1}^{11} I_i \, W_i$$

SFM ranges from 0 to 100. The weights have been assigned as follows: accident source term characterization--30%, accident fault tolerance--30%, effluent control--20%, and maintenance and waste management--20%.

5.4.6.1.1 Safety Scores and Rankings

The total SFM for the 16 blankets for both mirror versions and Tokamak versions are listed in Tables 5.4.6-1 through 5.4.6-4 along with the resulting rank ordering. It is seen that the rank ordering among mirror concepts is very similar to that among Tokamak concepts. Nine of these 16 blankets (seven for Tokamak) were mirror blanket reference cases for detailed evaluation for safety, economics, engineering, and R&D. Safety evaluation of the remaining blankets was performed to allow for more direct comparison of materials (see Subsection 5.4.6.2). It is emphasized that scores for nonreference blankets are more uncertain because the relevant values of such values as for tritium inventory were simply scaled from the reference blankets.

In brief, the most attractive blankets are a mixture of those that are most attractive from the tritium control standpoint (Li/Li/V15Cr5Ti, Li/He/HT-9) with those most attractive from the chemical reaction control

TABLE 5.4.6-1. SAFETY FIGURE-OF-MERIT (SFM) AND RANKINGS

Blanket	TMR SFM	TMR Rank	Tokamak SFM	Tokamak Rank
Li/Li/V15Cr5Ti[a,b]	63.0	3	59.7	3
Li/Li/HT-9[a]	53.8	9	49.2	9
17Li83Pb/17Li83Pb/V15Cr5Ti[a]	64.7	1	62.6	1
17Li83Pb/17Li-83Pb/HT-9	60.0	5	55.6	5
Li/He/V	57.4	7	56.4	4
Li/He/HT-9[a,b]	60.5	4	55.3	6
17Li83Pb/He/HT-9	56.7	8	53.6	8
FLIBE/He/HT-9/Be[a,b]	50.6	11	48.3	10
Li_2O/He/HT-9[a,b]	63.8	2	59.8	2
$LiAlO_2$/He/HT-9/Be[a,b]	58.5	6	54.1	7
$LiAlO_2$/He/PCA/Be	51.1	10	45.4	11
17Li83Pb/NS/HT-9	38.8	12	35.2	13
$LiAlO_2$/NS/HT-9/Be[a,b]	35.7	14	30.8	14
$LiAlO_2$/NS/PCA/Be	29.5	16	23.5	16
$LiAlO_2$/H_2O/HT-9/Be[a,b]	38.5	13	25.7	12
$LiAlO_2$/H_2O/PCA/Be	31.9	15	28.3	15

a. Reference blanket for detailed evaluation of mirror concepts.

b. Reference blanket for detailed evaluation of Tokamak concepts.

standpoint (Li_2O/He/HT-9) with a blanket doing generally well in both areas (17Li83Pb/17Li83Pb/V15Cr5Ti). The other liquid-breeder blankets are generally inferior to Li/Li/V15Cr5Ti, Li/He/HT-9. The other solid breeder blankets are generally inferior to Li_2O/He/HT-9 because of less favorable tritium control, higher tritium inventory, higher accident activation inventory, higher maintenance exposure, and higher waste management problems.

The comparison of the reference mirror blankets is more directly seen in Figure 5.4.6-1. The reference Tokamak blankets are compared in Figure 5.4.6-2. In both cases, the variance of effluent control, mainly tritium, and accident source term appear to be the major factors. Accident fault tolerance is not a major discriminator. Although significant differences among designs were found for specific fault tolerance issues, overall they tended to balance out. Blankets that do poorly in one of the four major safety evaluation areas do not do well overall. Specifically, the FLIBE/He, $LiAlO_2$/NS, and $LiAlO_2$/H_2O blankets do particularly poorly in effluent control and are the lowest overall blankets. The figures also help to explain various sensitivity cases.

5.4.6.1.2 Sensitivity Cases

A large number of sensitivity cases were examined. The most interesting ones are briefly discussed here. The resulting impacts on SFM (Tables 5.4.6-2, 5.4.6-3) and safety rank ordering (Tables 5.4.6-4 and 5.4.6-5) for the reference blankets have been calculated. The corresponding sensitivity cases are as follows:

Base Case: Reference index scores and index weights were used, accident source term--30%, accident fault tolerance--30%, effluent control--20%, maintenance and waste--20%.

Equal Weights, SFM1: Reference index scores were used but with equal weighting per index. By major area the weights are as follows: accident source term--27%, accident fault tolerance--45%, effluent control--9%, maintenance and waste--18%. The major difference is a large, 20% to 9%, decrease in the effluent importance. As seen in Tables 5.4.6-4 and 5.4.6-5

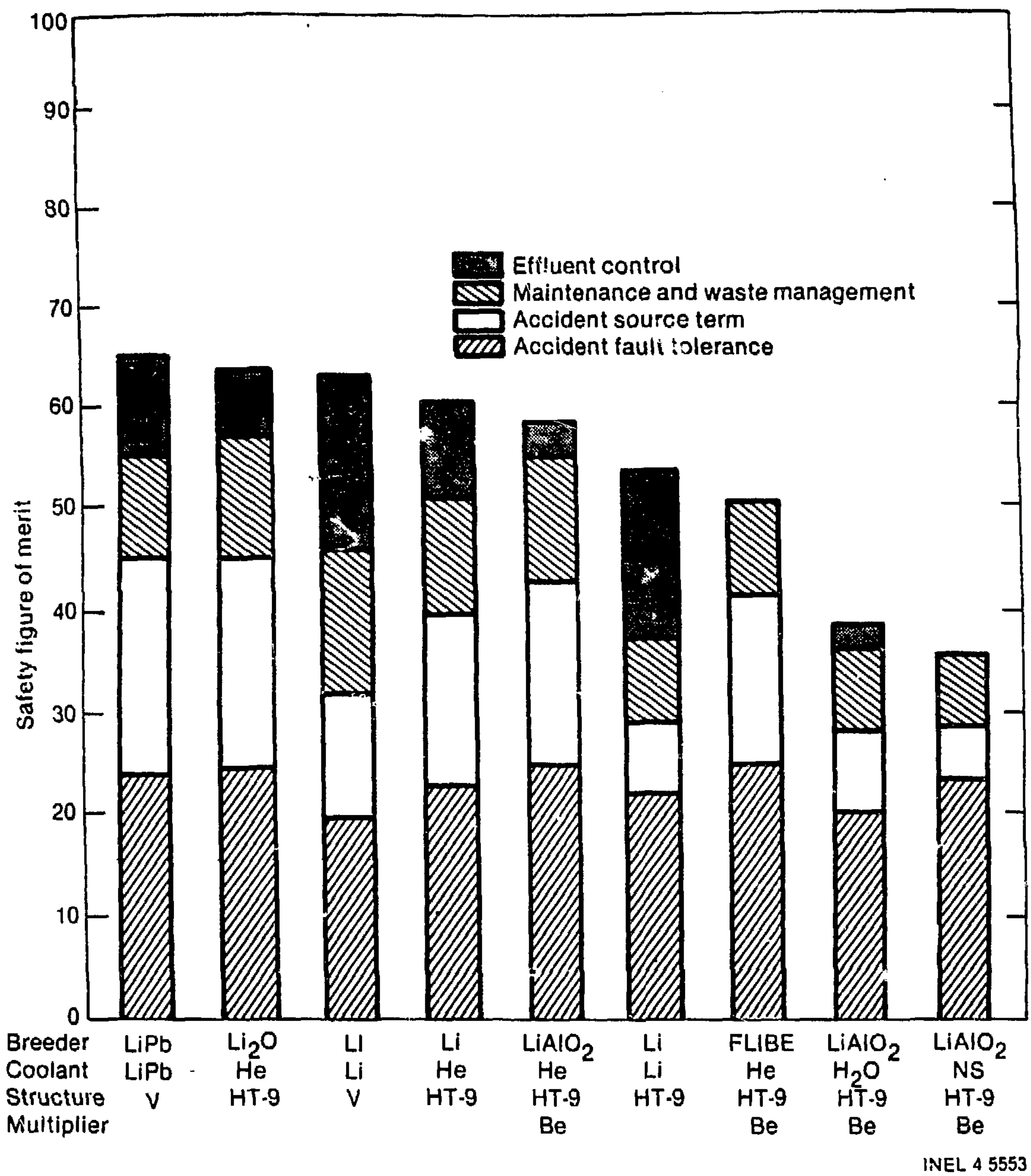

Figure 5.4.6-1 Safety Evaluation Results for Mirror Blankets.

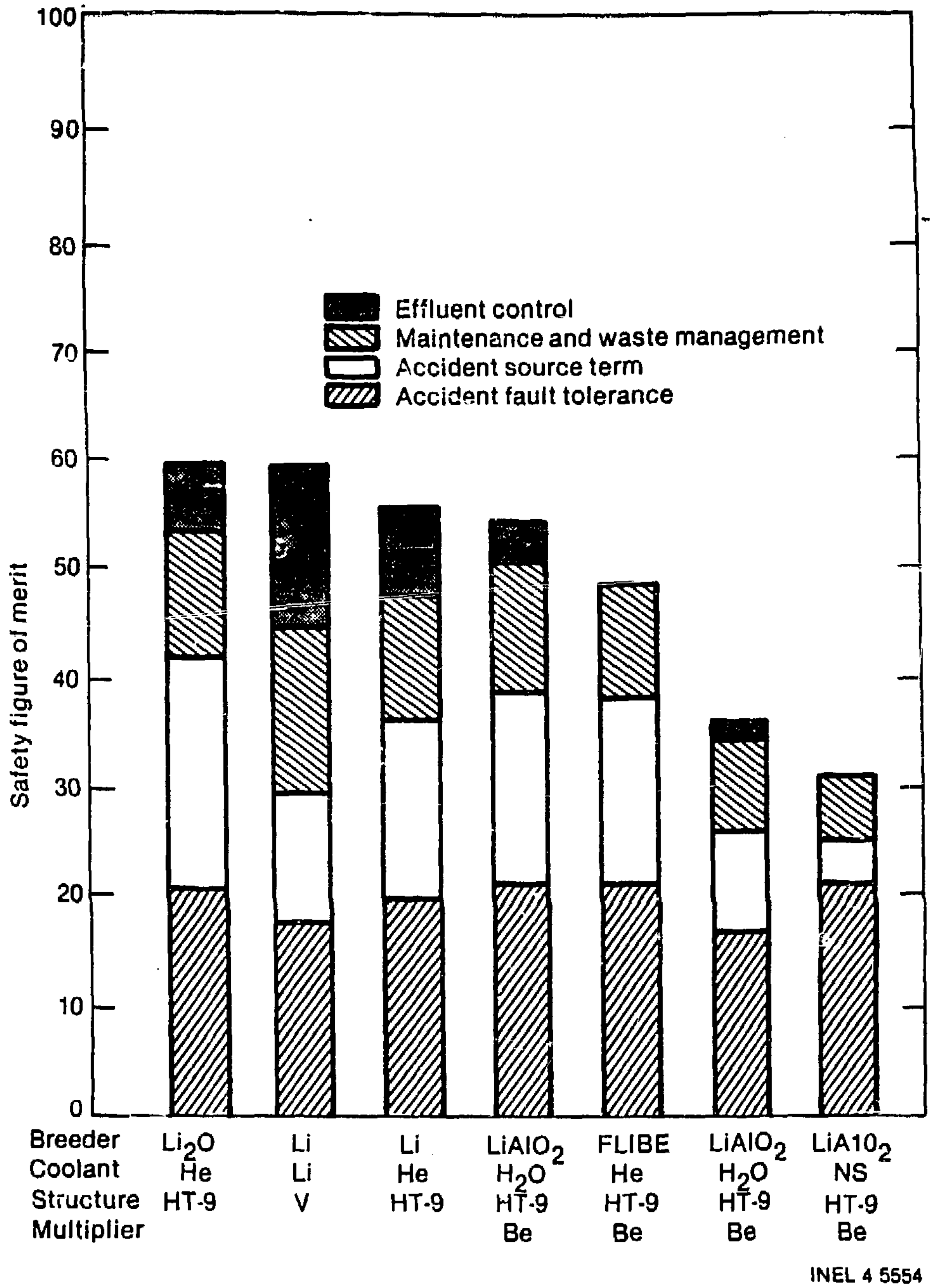

Figure 5.4.6-2 Safety Evaluation Results for Tokamak Blankets.

TABLE 5.4.6-2. SENSITIVITY CASES FOR REFERENCE MIRROR BLANKETS FOR SAFETY FIGURE-OF-MERIT[a]

Blanket	Base Case SFM	Equal Weights SFM1	Risk-Based SFM2	Optimistic Effluent Control	Pessimistic Effluent Control	Optimistic Chemical Reaction Control	Pessimistic Chemical Reaction Control
Li/Li/ V15Cr5Ti	63.0	61.4	47.1	66.0	66.0	69.0	--
Li/Li/ HT-9	53.8	54.2	35.6	56.8	56.8	56.8	--
17Li83Pb/ 17Li83Pb/ V15Cr5Ti	64.7	68.5	53.5	74.7	--	67.7	--
Li/He/ HT-9	60.5	64.2	46.3	70.5	50.5	63.5	--
FLIBE/He/ HT-9/Be	50.6	60.9	36.5	70.6	--	50.6	50.6
Li_2O/He/ HT-9	63.8	69.2	52.4	76.4	--	63.8	63.8
$LiAlO_2$/ He/HT-9/ Be	58.5	66.0	45.5	74.7	--	58.5	58.5
$LiAlO_2$/ NS/HT-9/ Be	35.7	47.0	14.6	55.7	--	35.7	35.7
$LiAlO_2$/ H_2O/HT-9/ Be	38.5	46.0	21.2	56.3	--	38.5	38.5

a. The numerical values are only relevant within a given sensitivity case.

TABLE 5.4.6-3. SENSITIVITY CASES FOR REFERENCE TOKAMAK BLANKETS FOR FIGURE-OF-MERIT[a]

Blanket	Base Case SFM	Equal Weights SFM1	Risk-Based SFM2	Optimistic Effluent Control	Pessimistic Effluent Control	Optimistic Chemical Reaction Control	Pessimistic Chemical Reaction Control
Li/Li/ V15Cr5Ti	59.7	58.0	44.2	64.5	64.5	65.7	--
Li/He/ HT-9	55.3	58.6	40.8	67.3	47.3	58.3	--
FLIBE/He/ HT-9/Be	48.3	57.1	33.9	68.3	--	48.3	48.3
Li_2O/He/ HT-9	59.8	64.3	47.2	73.6	--	59.8	59.8
$LiAlO_2$/ He/HT-9/ Be	54.1	60.3	40.2	70.5	--	54.1	54.1
$LiAlO_2$/ NS/HT-9/ Be	30.8	40.7	11.4	50.8	--	30.8	30.8
$LiAlO_2$/ H_2O/HT-9/ Be	35.7	42.2	19.9	54.5	--	35.7	35.7

a. The numerical values are only relevant within a given sensitivity case.

TABLE 5.4.6-4. SENSITIVITY CASES FOR REFERENCE MIRROR BLANKETS FOR SAFETY RANKINGS[a]

Blanket	Base Case SFM	Equal Weights SFM1	Risk-Based SFM2	Optimistic Effluent Control	Pessimistic Effluent Control	Optimistic Chemical Reaction Control	Pessimistic Chemical Reaction Control
Li/Li/ V15Cr5Ti	3	5	3	6	1	1	--
Li/Li/ HT-9	6	7	7	7	2	6	--
17Li83Pb/ 17Li83Pb/ V15Cr5Ti	1	2	1	2	--	2	--
Li/He/ HT-9	4	4	4	5	3	4	--
FLIBE/He/ HT-9/Be	7	6	6	4	--	7	3
Li$_2$O/He/ HT-9/Be	2	1	2	1	--	3	1
LiAl[illegible]/ He/HT-9/ Be	5	3	5	2	--	5	2
LiAlO$_2$/ NS/HT-9/ Be	9	8	9	9	--	9	5
LiAlO$_2$/ H$_2$O/HT-9/ Be	8	9	8	8	--	8	4

a. Circles denote the number of the top group of blankets for a given sensitivity case.

TABLE 5.4.6-5. SENSITIVITY CASES FOR REFERENCE TOKAMAK BLANKETS FOR SAFETY RANKINGS[a]

Blanket	Base Case SFM	Equal Weights SFM1	Risk-Based SFM2	Optimistic Effluent Control	Pessimistic Effluent Control	Optimistic Chemical Reaction Control	Pessimistic Chemical Reaction Control
Li/Li/V15Cr5Ti	2	4	2	5	1	1	--
Li/He/HT-9	3	3	3	4	2	3	--
FLIBE/He/HT-9/Be	5	5	5	3	--	5	3
Li_2O/He/HT-9	1	1	1	1	--	2	1
$LiAlO_2$/He/HT-9/Be	4	2	4	2	--	4	2
$LiAlO_2$/NS/HT-9/Be	7	7	7	7	--	7	5
$LiAlO_2$/H_2O/HT-9/Be	6	6	6	6	--	6	4

a. Circles denote the members of the top group of blankets for a given sensitivity case.

the major impact is a slight shift from liquid metal blankets, which have better tritium control, to solid-breeder blankets. Thus, Li_2O/He/HT-9 looks even better, while the Salt and water-cooled blankets still rank at the bottom.

Risk-Based, SFM2: Reference index scores were used. The 60% weighting to accidents was not divided equally between source term and fault tolerance but instead these two parts were added together, forming alternate SFM No. 2. Somewhat surprisingly, this case did not significantly affect the rank ordering of the reference blankets. Overall, the base, SFM1, and SFM2 cases do not show a large difference on rank ordering. All use the reference index scores but with different weighting. Thus, it appears that the final rank ordering is not highly sensitive to the exact judged index weighting.

Optimistic Effluent Control: Here all blankets are given the best score for the effluent index, 20 points. This would require (a) various tritium control, also ^{39}Ar for Nitrate Salt, techniques to be developed and proven, (b) the effluent release standards to be looser by about two orders of magnitude (starting at 1ι 100 Ci/day, ∿10 mrem/yr), or (c) the tritium movement assumptions based in this study to be found to be too conservative by two orders of magnitude. The impact on the rank ordering is dramatic. The Li/Li/V15Cr5Ti design drops from 3rd to 6th (mirror) and 2nd to 5th (Tokamak). Figures 5.4.6-1 and 5.4.6-2 help to demonstrate why the change occurs. The self-cooled lithium cases score the best in the area of tritium control. Thus, when it is assumed that all blankets do well in this area, it is not surprising that the lithium self-cooled blankets are the most affected. The most likely way to achieve dramatically improve tritium control is develop tritium control barriers and/or on-line tritium oxidizers for helium streams.

Pessimistic Effluents Control: Here it is assumed that either (a) a major tritium control assumption is found to be too optimistic or (b) effluent standards are 1-2 orders of magnitude stricter than assumed here. In this case, the lithium-self-cooled designs were scored 20 pts in the tritium effluent area because they appear able to still achieve low tritium release. The Li/He/HT-9 design was given 0 pts for tritium control but may still be acceptable. All other blankets would appear unacceptable. The key uncertainties are (a) tritium coming through first wall increases, which impacts all helium, salt, or water-cooled blankets, (b) hydrogen dilution of tritium fails to lower tritium permeation, which impacts all 17Li83Pb, helium, salt, or water-cooled blankets, (c) beryllium releases significant amount of tritium, which impacts $LiAlO_2$/He and FLIBE/He designs, and (d) the deviation of tritium permeation through oxidized HT-9 is much less than a factor of 100, which impacts all blankets.

Optimistic Chemical Reaction Control: Here it is assumed that chemical reaction control is even better than that assumed in the reference case, which allows water in shields, choke coils, direct converters, and halo scrapers but assumes adequate isolation between water and reactive metals. If the water were replaced in these components then the remaining lithium and vanadium fault tolerance penalties for water chemical reactions (Index 7) would be removed. In this case the reference top blankets would not change, but the Li/Li/V15Cr5Ti and 17Li83Pb/17Li83Pb/V15Cr5Ti cases would rise to the top of the list.

Actually, the full impact of the optimistic chemical reaction control case is even greater since the high volatility assigned to lithium-related elements, because of the possibility of combustion, would be lowered. That is, the volatility of the activity in liquid lithium would be lowered. In this instance, Li/Li/HT-9 might replace Li_2O/He/HT-9 in the top group blankets.

To insure that water chemical reactions with Li, V15Cr5Ti, and 17Li83Pb are adequately controlled, it is strongly recommended that future designers using these materials do not use water anywhere in the nuclear island, see Subsection 5.4.3.5.

Pessimistic Chemical Reactions: Here it is assumed that the favorable assumptions relating to control chemical reactions are proven false. Then those blankets with either lithium or vanadium may not be acceptable. The key assumptions for the reference case are (a) nonwater-cooled limiters are used for lithium or vanadium blankets, (b) adequate separation between lithium, 17Li83Pb, and vanadium versus water is provided by design so that the risk of metal-water reactions are small, (c) a nitrogen cover gas is used for lithium and 17Li83Pb designs, and (d) nitrogen-lithium reactions are mild in the 500-700°C range.

It should be noted that if both the pessimistic tritium control and pessimistic chemical reaction control cases prove true, it is not clear that there is an acceptable blanket option. For these reasons the research items relating to tritium and chemical reactions are of highest priority, as seen in the R&D assessment.

Impact of Deviation from 5 MW/m^2 Neutron Wall Loading: The exact value of the wall-loading value will change design details. In general, the relative ranking and attractiveness of concepts in the Safety Evaluation would not change if the 5 MW/m^2 value were lowered or raised. However, two specific issues might change, which might provide an impact on the overall Safety Evaluation. The first issue is the relative ability of concepts to survive cooling transients. Afterheat levels will vary up or down with the wall loading. An increased wall loading will harm the liquid breeder concepts since some may no longer be able to passively deal with cooling transients. (It is already assumed that solid breeder issues have troubles in this regard.) A decreased wall loading would probably have little effect since it is not likely that the afterheat level could fall sufficiently to allow solid breeder concepts to passively handle these transients. The second issue is whether a change in neutron wall loading makes it harder or easier to use non-water-cooling for near-plasma components, e.g., limiters. Assuming that surface heat flux would scale roughly as the neutron wall loading, it seems likely that increased wall loading would make it more difficult to replace water cooling, hence significantly lowering the safety attractiveness of lithium and/or vanadium concepts. Similarly, reduced wall loading seems likely to improve chances of replacing water cooling, hence raising the attractiveness of reactive metal blankets.

In summary, raising the 5 MW/m^2 value would tend to decrease the safety attractiveness of liquid-metal and/or vanadium blankets because of increased difficulty in passively handling cooling transients and in replacing water cooling in near-plasma components. Decreasing the 5 MW/m^2 value would tend to increase the safety attractiveness of lithium and/or vanadium blankets because of increased ability to replace water cooling.

Other Physics Confinement Concepts: The Safety Evaluation found new-identical relative rankings among concepts independent of whether the physics concept was TMR or tokamak. Although the rank ordering would have to be examined for other physics concepts on a case-by-case basis, it does not appear that the safety rank ordering would necessarily change. The most likely issues that could lead to changes are (a) differences in the chemical reaction risk of water and (b) presence of significant amounts of copper. If a given physics concept had special requirements for water-cooled components, it could severely harm the vanadium and/or lithium concepts. On the other hand, if a given physics concept had no requirements for water-cooled components, the blankets with reactive metals would be helped. If significant amounts of copper were present, the Li/Li/V concept relative ranking would decrease. This is because copper grossly fails 10CFR61 and its presence would mean no blanket, even Li/Li/V, would score well in the Waste Management Index. If a physics concept required copper coils imbedded in the blanket, for example, one could forget about fusion meeting the near-surface burial goal.

5.4.6.1.3 Low-Activation Steels

One purpose of the BCSS was to examine the value of replacing the reference steels, HT-9 and PCA, with two "low-activation" alternate steels, modified HT-9 and Tenelon. These were designed to meet the goal of near surface-burial for the structural material itself. In the area of waste management, it was found in this study that they do come close to satisfying 10CFR61 as does V15Cr5Ti, whereas the reference steels are two orders of magnitude off. Basically, the "low-activation" steels solve the waste disposal problem by eliminating elements that give rise to long-term

(>10 year) isotopes and replacing them with elements that give rise to shorter-term isotopes. However, in all other areas where activation is important - accident source term, afterheat, maintenance of blanket structure, and maintenance of cooling systems - use of these proposed "low-activation" steels has either an insignificant impact or a negative one. In Modified HT-9, the key problem is tungsten, which replaces molybdenum. In Tenelon, the key problem is manganese, which replaces nickel. Specific numerical comparisons are listed in Table 5.4.6-6 and 7.

On balance, it cannot be said that the proposed "low-activation" steels are necessarily an improvement over the reference steels. It does appear that Modified HT-9 would be a net improvement over HT-9 but that Tenelon would not be a net improvement over PCA. One key aspect limiting the value of "low-activation" steels is that most blankets do not achieve near-surface burial even with use of "low-activation" steels because of the breeder or coolant. For the reference-composition structural materials and activation areas studied, impurities were generally not found to be important. The main exception was that niobium content causes waste management problems.

If one wants a low activation structural material, the proposed steels do not appear to the answer. V15Cr5Ti is one possibility. Adoption of a long time frame opens up the possibility for truly low activation materials like SiC.

Structural materials are only one part of the activation picture. Many of the other blanket materials, breeders and coolants, have significant radioactivity. In some cases the major activating elements are prime constituents, e.g. lead in 17Li83Pb and sodium and potassium in Nitrate Salt. Blankets with these materials would not be low activation even if the structural material were SiC. In other cases the main activating elements are impurities, especially sodium and potassium in all lithium-bearing materials. A worthwhile goal seems to be economical reduction of impurities in such materials as lithium and Li_2O. Breeder materials using enriched lithium may automatically benefit from lower levels of lithium-related impurities, e.g. 17Li83Pb.

TABLE 5.4.6-6. COMPARISON OF LOW-ACTIVATION STEELS WITH REFERENCE STEELS AND V15Cr5Ti[a]

	V15Cr5Ti	HT-9	Modified HT-9	PCA	Tenelon
Relative Biological Health Potential	1.0	4.3	5.0 (15% worse than HT-9)	8.5	13.5 (59% worse than PCA)
Relative Public Health Effects	1.0	7.7	7.1 (9% better)	23.4	20.4 (15% better)
Relative BHP with Volatility Effects	1.0	5.9	7.3 (26% worse)	4.7	12.7 (174% worse)
Relative Health Effects with Volatility Effects	1.0	9.0	10.0 (11% worse)	7.7	18.6 (141% worse)
Remote Maintenance Rating (mR/h) at 1 day	1.4×10^{10}	4.2×10^{9}	4.2×10^{9} (2% worse)	1.2×10^{10}	1.3×10^{10} (15% worse)
Remote Maintenance Rating (mR/h) at 10 yr	6.5×10^{3}	4.4×10^{7}	1.1×10^{7} (factor of 4 better)	2.3×10^{8}	1.4×10^{7} (factor of 16 better)
Waste Disposal Rating at 10 yr	1.7	484.7	1.4 (factor of 346 better)	197.2	1.5 (factor of 131 better)

a. There is also an impact on afterheat levels. In particular Tenelon would have afterheat levels that are approximately three times that of PCA.

TABLE 5.4.6-7. COMPARISON OF LOW-ACTIVATION STEELS WITH REFERENCE STEELS BY INFLUENCE ON SAFETY FIGURE-OF-MERIT

Issue	Modified HT-9 vs. HT-9	Tenelon vs PCA
Accident Source Ter.. (Index 1)	0-1.2 points better	0-0.4 points worse
Afterheat, Cooling Fault Tolerance (Index 5)	0-3 points worse	0-6 points worse
Occupational Exposure (Index 10)	nil	nil
Waste Management (Index 11)	0.7-4.3 better	0.6-1.4 better
Overall	2.3 worse to 5.5 better	5.8 worse to 1.4 better

Breeder and coolant-related radioactivity may prove more important in some cases than structural materials. For example, whereas it is agreed that hands-on maintenance of blanket structure is highly unlikely, limited hands-on maintenance of components such as tritium purge systems and steam generators appears achievable and a very worthwhile goal. Use of helium appears the way to achieve this goal.

5.4.6.2 Discussion of Results by Material

The discussion of overall results will be done two ways. First, the evaluation of 16 blankets allows direct pairwise comparison of a specific material choice, all other materials kept constant. For example, the advantage of HT-9 versus PCA is examined for the pairs of blankets, $LiAlO_2$/He/X/Be, $LiAlO_2$/NS/X/Be, and $LiAlO_2$/H_2O/X/Be, where X=HT-9 or PCA. Second, (Subsection 5.4.6.3) the results are discussed by blanket.

5.4.6.2.1 Structural Materials

The Table 5.4.6-8 shows the pairwise SFM comparison of structural materials, V15Cr5Ti versus HT-9 and HT-9 versus PCA. The differences that

TABLE 5.4.6-8. PAIRWISE SAFETY-FIGURE-OF-MERIT COMPARISON OF STRUCTURAL MATERIALS

Comparison: V15Cr5Ti Advantage Over HT-9

Relevant Blankets: Li, Li/X, 17Li83Pb/17Li83Pb/X, Li/He/X

Issue	Type Blanket	Difference
Source Term	Lithium	+4.3 ± 0.7
	17Li83Pb	+3.4 ± 0.5
Fault Tolerance	Self-cooled	-1.5 ± 0.6
	Helium-cooled	-2.4 ± 0.5
Tritium Control	Self-cooled	0
	Helium-cooled	-9.0 ± 1.0
Maintenance and Waste	Lithium	+6.5 ± 0.3
	17Li83Pb	+3.9 ± 0.1
Overall	Li/Li/X	+9.9 ± 0.7
	17Li83Pb/17Li83Pb/X	+5.9 ± 1.2
	Li/He/X	-2.0 ± 1.2

Comparison: HT-9 Advantage Over PCA

Relevant Blankets: $LiAlO_2$/He/X, $LiAlO_2$/NS/X, $LiAlO_2/H_2O$/X

Issue	Type Blanket	Difference
Source Term	Helium-cooled	+2.3 ± 0.0
	Salt-cooled	+1.4 ± 0.0
	Water-cooled	+1.9 ± 0.1
Fault Tolerance	all	+0.5 ± 0.5
Tritium Control	all	0
Maintenance and Waste	all	+4.9 ± 0.5
Overall	Helium-cooled	+8.1 ± 0.7
	Salt-cooled	+6.8 ± 0.6
	Water-cooled	+7.0 ± 0.4

are quoted refer to the difference in total SFM for the stated blankets for both mirror and Tokamak versions.

For V15Cr5Ti versus HT-9, the key vanadium advantages are activation related (accident source term and maintenance and waste). The advantage is stronger when a lower activation breeder, lithium, is used rather than 17Li83Pb. There is a vanadium disadvantage in fault tolerance because of chemical reactivity. The key swing issue is tritium control. The V15Cr5Ti versus HT-9 difference in tritium control is zero for the self-cooled designs because the permeation through the first wall is not relevant and because a HT-9 heat exchanger is used in all cases. The V15Cr5Ti to HT-9 tritium control difference for helium cooled designs appears high, although the Li/He/V15Cr5Ti case received little attention. Overall, V15Cr5Ti is preferred unless its use adversely impacts tritium control. Inability to reduce the risk of vanadium-oxidation could also impose severe penalties for V15Cr5Ti use.

The HT-9 to PCA comparison is simpler. HT-9 has significant advantages in the activation areas and no identified disadvantage. However, the tritium control for HT-9 was assumed the same as PCA with the higher permeability of HT-9 offset by a higher "oxide barrier factor." If this assumption changes, PCA could have an advantage over HT-9. In terms of the BCSS, about a factor of 10 lower net permeability for oxidized PCA versus oxidized HT-9 could be needed to offset the HT-9 activation advantages. No examination of crack propagation was included in the BCSS and the Ductile-Brittle Transition Temperature problem for HT-9 was assumed to not be serious.

5.4.6.2.2 Breeder Materials

Table 5.4.6-9 shows pairwise breeder material comparisons. For Li_2O versus $LiAlO_2$/Be the key areas are source term (activation and tritium inventory) and tritium control, all favoring Li_2O. Li_2O is slightly worse than $LiAlO_2$ in the waste area because of somewhat more structure. The long-term waste disposal advantage of Li_2O is not relevant because use of HT-9 penalizes both concepts waste disposal rating (WDR). The use

TABLE 5.4.6-9. PAIRWISE SAFETY-FIGURE-OF-MERIT COMPARISON OF BREEDER MATERIALS

Comparison: Li_2O Advantage Over $LiAlO_2$/Be

Relevant Blankets: X/He/HT-9

Issue	Difference
Source Term	+2.9 ± 0.3
Fault Tolerance	-0.1 ± 0.0
Tritium Control	+3.1 ± 0.5
Maintenance and Waste	-0.4 ± 0.5
Overall	+5.5 ± 0.2

Comparison: Li_2O Advantage Over FLIBE/Be

Relevant Blankets: X/He/HT-9

Issue	Difference
Beryllium Toxicity	+3.2 ± 0.1
Structure Tritium Inventory	+1.4 ± 0.1
Fault Tolerance	-0.9 ± 0.1
Tritium Control	+6.8 ± 0.6
Maintenance and Waste	+1.9 ± 0.2
Overall	+12.4 ± 0.9

Comparison: Li Advantage Over 17Li83Pb

Relevant Blankets: X/X/V, X/X/HT-9, X/He/HT-9

Issue	Type Blanket	Difference
Tritium Inventory	Self-cooled	-9.2 ± 1.0
	Helium-cooled	-4.4 ± 0.2
Source Term Activation	Self-cooled	-1.7 ± 0.0
	Helium-cooled	-1.0 ± 0.1
Fault Tolerance	Self-cooled	-3.8 ± 0.2
	Helium-cooled	-2.5 ± 0.1
Tritium Control	Self-cooled	+6.7 ± 0.3
	Helium-cooled	+9.0 ± 1.0

TABLE 5.4.6-9 (continued)

Issue	Type Blanket	Difference
Maintenance and Waste	X/X/V	+5.1 ± 0.2
	X/X/HT-9	+2.2 ± 0.1
	X/He/HT-9	+1.5 ± 0.1
Overall	X/X/V	-2.3 ± 0.6
	X/X/HT-9	-6.3 ± 0.1
	X/He/HT-9	+2.8 ± 1.0

Comparison: Li_2O Advantage Over Li

Relevant Blankets: X/He/HT-9

Issue	Difference
Tritium Inventory	+0.4 ± 0.2
Source Term Activation	+4.1 ± 0.1
Cooling Transients	-4.0 ± 0.1
Chemical Reactivity	+5.0 ± 0.1
Tritium Control	-2.2 ± 0.4
Maintenance and Waste	+0.4 ± 0.4
Overall	+3.9 ± 0.6

of beryllium with $LiAlO_2$ does not play a major role because $LiAlO_2$ already has far higher BHP than Li_2O without the addition of beryllium toxicity. Beryllium would make a difference (several points) if the breeder were lower BHP, e.g. lithium, Li_2O, or FLIBE. Overall Li_2O is strongly preferred over $LiAlO_2$/Be.

$LiAlO_2$ has served as a representative ternary ceramic in this study. The key issues among ternary ceramic options are tritium inventory, tritium release form, short-term radioactivity, and long-term radio activity. From the activity standpoint, zirconium-based ceramics would be greatly inferior to $LiAlO_2$ and silicon-based ceramics might be better. Specifically, if a silicon-based ceramic could have a low enough impurity level, it would meet near-surface burial requirements (10CRF61), unlike $LiAlO_2$.

Li_2O has several advantages over FLIBE/Be, beryllium toxicity, structure tritium inventory, tritium control, and maintenance and waste.

The last is caused by the higher structural fraction in the FLIBE blanket. FLIBE has a slight advantage in fault tolerance, mainly because of perceived greater ease in surviving cooling transients. Overall, Li_2O is strongly preferred over FLIBE/Be.

The lithium-17Li83Pb comparison is complex. 17Li83Pb has advantages in the areas of tritium inventory, source term activation, and fault tolerance. The accident source term activation advantage of 17Li83Pb is caused by the higher assigned volatility for elemental lithium radioactivity caused by the possibility of chemical combustion. Without volatility effects, lithium is better than 17Li83Pb in the area of accident source term activation. The fault tolerance advantage of 17Li83Pb is caused by the higher chemical reactivity of lithium. The 17Li83Pb tritium inventory advantage (9.2 pts for self-cooled, 4.4 pts for helium) balances against a lithium tritium control advantage (6.2 pts for self-cooled, 9.0 pts for helium-cooled), so that 17Li83Pb is preferred for self-cooled concepts and lithium is preferred for helium-cooled concepts. Lithium has an advantage in the maintenance and waste management area because of its lower activation. This advantage is highest with the low activation V15Cr5Ti structure and lower with the HT-9 cases.

The value of the Li_2O versus Li comparison is limited because only the X/He/HT-9 cases provide a direct comparison. Li_2O has advantages in tritium inventory, tritium control, source term activation, maintenance and waste, and chemical reactivity (fault tolerance). The lower tritium effluent for Li_2O/He/HT-9 versus Li/He/HT-9 is a result of designs being frozen at a specific time. Given more optimized designs, Li/He/HT-9 should demonstrate better tritium control than Li_2O/He/HT-9. The source term activation advantage is largely caused by the high volatility assigned to lithium species because of the possibility of chemical combustion. Lithium has an advantage in the area of fault tolerance to cooling transients. Overall, Li_2O is preferred over lithium for X/He/HT-9 blankets.

5.4.6.2.3 Coolant Materials

The final set of pairwise comparisons is shown in Table 5.4.6-10. The lithium-17Li83Pb comparison shown with the breeders is not repeated.

The helium-water comparison shows several helium advantages and no water advantages. The lower activation and tritium inventory of helium lead to source term and maintenance and waste management advantages. The lower pressure leads to fault tolerance and tritium control advantages. The helium advantage in tritium control is partially caused by the fact that the water pressure is high enough so that steam generator leaks go from primary to steam, releasing tritium.

The helium-salt comparison shows several helium advantages and no salt advantages. All are related to the high tritium and high radioactivity in the salt. No net fault tolerance difference is seen. The relative attractiveness of the Nitrate Salt would improve dramatically if it were elementally-tailored to replace sodium and potassium. Perhaps a lithium-based nitrate/nitrite salt would be a good starting point. Such a tailored salt would avoid much of the Nitrate Salt's disadvantages due to high radioactivity.

The helium to lithium comparison is more complex, but only the Li/X/HT-9 and Li/X/V15Cr5Ti pairs provide a direct comparison. Helium has advantages in source term and maintenance and waste management because of its nil tritium inventory and lower activity. There is a slight fault tolerance advantage for helium, but limited because both Li/X/HT-9 and Li/X/V15Cr5Ti have some lithium chemical reaction concerns. Lithium has tritium control advantages, more for Li/X/V15Cr5Ti than for Li/X/HT-9. The Li/He/V15Cr5Ti blanket is apparently more penalized versus Li/Li/V15Cr5Ti because only the helium cases tritium control is influenced by the high vanadium permeability. Overall, the strength of the tritium control advantage for lithium determines which, helium or lithium, is preferred.

TABLE 5.4.6-10. PAIRWISE SAFETY-FIGURE-OF-MERIT COMPARISON OF COOLANT MATERIALS

Comparison: Helium Advantage Over H_2O

Relevant Blankets: $LiAlO_2$/X/HT-9/Be, $LiAlO_2$/X/PCA/Be

Issue	Difference
Source Term	+9.4 ± 0.9
Fault Tolerance	+4.1 ± 0.1
Tritium Control	+2.0 ± 0.4
Maintenance and Waste	+3.2 ± 1.1
Overall	+18.7 ± 1.6

Comparison: Helium Advantage Over Nitrate Salt

Relevant Blankets: $LiAlO_2$/X/HT-9/Be, $LiAlO_2$/X/PCA/Be

Issue	Difference
Source Term	+13.3 ± 0.6
Fault Tolerance	0.0 ± 0.0
Effluent Control	+3.7 ± 0.1
Maintenance and Waste	+5.5 ± 0.4
Overall	+22.4 ± 0.9

Comparison: Helium Advantage Over Lithium

Relevant Blankets: Li/X/HT-9, Li/X/V15Cr5Ti

Issue	Type Blanket	Difference
Source Term	either	+8.9 ± 0.6
Fault Tolerance	either	+0.7 ± 0.5
Tritium Control	HT-9	-7.1 ± 0.1
	V15Cr5Ti	-16.1 ± 0.9
Maintenance and Waste	either	+2.9 ± 0.5
Overall	HT-9	+6.4 ± 0.3
	V15Cr5Ti	-4.5 ± 1.2

5.4.6.3 Discussion of Results by Blanket

The proceeding discussion broke results down to the individual material level. Ultimately the total blanket, comprised of various materials, is what is relevant. Not all combinations make sense or are practical. The following discussion is intended to give a brief description of why each of the reference blankets ranks where it does. The order is best to worst, recognizing that only the final "top" group of blanket concepts were evaluated. Figures 5.4.6-1 and 5.4.6-2 are particularly relevant.

The Li_2O/He/HT-9 blanket generally does very well in all areas. The largest uncertainty is tritium control: if tritium control becomes significantly easier, this blanket becomes the unquestioned best choice, if tritium control is significantly less favorable, the blanket is far less attractive and may become unacceptable.

The 17Li83PB/17Li83Pb/V15Cr5Ti blanket (TMR only) scores about the same as Li_2O/He/HT-9. This blanket also does well across the board, generally a little better than Li_2O/He/HT-9 in tritium control and worse in occupational exposure and waste management. The latter is caused by the high activity of 17Li83Pb. This blanket has two key uncertainty areas, how well will the tritium control scheme work and how well will 17Li83Pb and V15Cr5Ti be protected from water and air? Highly unfavorable outcomes in either area might make the blanket unacceptable. No obvious area of significant safety improvement appears.

The Li/Li/V design scores about the same as 17Li83PB/17Li83Pb/V15Cr5Ti. The lithium advantages in radioactivity and better tritium control are offset by chemical reaction concerns and higher tritium inventory in lithium. This blanket could be significantly hurt, to the point of being unacceptable, if air and water reactions were not controlled to the extent assumed in this study. On the other hand, if air and water reaction risk is further lowered (neither water or air in the reactor building?), the blanket would be even more attractive and would be the best overall choice.

Next comes the Li/He/HT-9 blanket. The use of HT-9 (radioactivity) and helium (tritium control) outweighs its advantages over Li/Li/V15Cr5Ti in the area of helium's low radioactivity. Tritium control is the major uncertainty, which could raise or lower the attractiveness of this blanket. The overall safety attractiveness has significantly fallen from the cases of Li_2O/He/HT-9 and 17Li83PB/17Li83Pb/V15Cr5Ti.

The fifth blanket is $LiAlO_2$/He/HT-9/Be. All of the higher rated blankets have significantly lower activation: Li_2O/He/HT-9 is lower because of Li_2O versus $LiAlO_2$/Be, 17Li83PB/17Li83Pb/V15Cr5Ti is lower because of V15Cr5Ti versus HT-9, Li/Li/V15Cr5Ti is lower because of V15Cr5Ti and Li versus $LiAlO_2$/Be and HT-9, and Li/He/HT-9 is lower because of Li versus $LiAlO_2$/Be. All of the higher blankets also do better in the area of tritium control. The $LiAlO_2$/He/HT-9/Be blanket is as close to the others as it is because of the lack of chemical reaction concerns considered in the evaluation. The blanket would be significantly more attractive if tritium control were 1-2 orders of magnitude better, but not better than Li_2O/He/HT-9 because of the activation and chemical toxicity difference between Li_2O and $LiAlO_2$/Be.

The sixth blanket, Li/Li/HT-9 (TMR only), is very similar to Li/Li/V except for higher activation from HT-9. The V15Cr5Ti to HT-9 activation difference is the largest for this pair of blankets because lithium is fairly low activation (does not mask the structure) and because there is no tritium control difference. The latter is predicated on using steel for the Li/Li/V15Cr5Ti loop for the nonblanket parts. Overall, the activation disadvantage for HT-9 versus Li/Li/V15Cr5Ti is sufficient to drop Li/Li/HT-9 into the middle of the pack. Better chemical reaction control would help this blanket, as for Li/Li/V15Cr5Ti, and raise its attractiveness, but never to more than Li/Li/V15Cr5Ti.

The seventh blanket, FLIBE/He/HT-9/Be, does very poorly in tritium control and is heavily penalized as a result. The chemical toxicity of beryllium is a distinct disadvantage because otherwise FLIBE would compare favorably with Li_2O in terms of BHP. The blanket would be helped if tritium control were improved by two orders of magnitude or if it were found that beryllium toxicity is not as bad as assumed here (see Subsection 5.4.2.3). However, it does not appear that this blanket could be more attractive than Li_2O/He/HT-9 or $LiAlO_2$/He/HT-9/Be.

The eighth blanket is $LiAlO_2$/H_2O/HT-9/Be. The blanket does poorly in all major safety areas. Its biggest problems are tritium control and pressure. The pressure is inherently high enough so that tritium may leak from the primary to steam side. The high pressure makes entrained activation products and tritium very mobile. The two-phase high-pressure nature allows for significant pressurization of whatever chamber the water would leak into. The blanket does avoid chemical reaction problems.

The ninth blanket is $LiAlO_2$/NS/HT-9/Be. This blanket does poorly in all safety areas except fault tolerance, where the low operating pressure is an advantage. This advantage is based on the questionable assumption that salt decomposition is not a problem. If that assumption is not made, the blanket appears potentially unacceptable. The very high activity and high tritium inventory in the salt outweigh its low pressure advantages. ^{39}Ar, rather than tritium, would be a major effluent control problem. The major way to improve nitrate salt would be replacement of its sodium and potassium with something else, a form of elemental tailoring. A low-activation nitrate salt with good thermal stability would rank much better in the safety evaluation.

5.4.6.4 Conclusions

Given the reference assumptions that

(a) some tritium control ideas will work,

(b) air chemical reaction problems are largely solved by nitrogen or carbon dioxide cover gases, and

(c) water chemical reaction problems are largely solved by elimination of water-cooled limiters and adequate separation of water and reactive metals (lithium, vanadium),

then the top blankets are

17Li83Pb/17Li83Pb/V15Cr5Ti,

Li_2O/He/HT-9,

Li/Li/V15Cr5Ti, and

Li/He/HT-9.

These top choices are a mixture of blankets that are especially attractive in terms of tritium control, i.e. elemental lithium-bearing, and those most attractive in terms of chemical reaction control, i.e. Li_2O/He/HT-9.

This is most easily seen by looking at two alternative sets of assumptions.

First, if one believes that

(a) adequate tritium control is economically credible for all designs,

(b) cooling and pressure transients can be passively protected against, and

(c) chemical reaction problems (lithium, 17Li83Pb, NS, vanadium, Be-powder) are not solved,

then the preferences are

Li_2/He/HT-9 (appears on reference top list),

$LiAlO_2$/He/HT-9/Be, and

FLIBE/He/HT-9/Be,

and the blankets with either lithium or vanadium may not be acceptable. In other words, if passive control of lithium or vanadium reactions is not sufficient to effectively eliminate these accident concerns, then He/solid breeders are the most attractive.

Second, if one believes that

(a) nonair building atmosphere or protection schemes are economically credible and solve chemical reaction problems,

(b) nonwater-cooled components are technically credible and are used to reduce water-metal problems, and

(c) tritium control is extremely difficult,

then the preferences are

Li/Li/V15Cr5Ti (appears on reference top list),

Li/Li/HT-9, and

Li/He/HT-9 (appears on reference top list).

and most other designs may not be acceptable. In other words, if tritium control of 17Li83Pb and solid breeder designs is not adequate to meet social health and safety standards, the elemental lithium-bearing designs are the most attractive.

Therefore, it is seen that the top blanket preferences depend on some optimism in tritium control and chemical reaction control. Pessimism in both areas produces an empty set of acceptable blanket choices. That is, the combination of tritium/effluent control and chemical reaction control concerns is a fusion feasibility issue. None of the blankets studied avoids both major problem areas. The nitrate salt blanket appears to avoid tritium and lithium chemical reaction problems but has ^{39}Ar effluent and chemical decomposition problems.

The key parts of these major issues and corresponding needed research are as follows:

- better define vanadium-air reactions
- verify that nitrogen and carbon dioxide are fairly inert toward V15Cr5Ti and steels
- perform studies to determine the viability and cost of using an inert cover gas/building atmosphere
- better define lithium and 17Li83Pb reactions with air and water in geometries of interest
- verify that nitrogen is sufficiently inert toward lithium and 17Li83Pb at temperatures of 500-700°C
- perform studies to determine the viability and cost of using nonwater-coolants for limiters, resistive choke coils, shields, direct converters, and halo scrapers
- better understand tritium migration through first walls
- understand deviation from classical tritium permeation ("oxide barrier factor") including impact of hydrogen addition
- better define tritium release form from solid breeders
- better define tritium oxidation kinetics in helium streams, especially the impact of surface reactions

- examine the behavior of tritium in beryllium.

A list of safety-related potential critical flaws is given in Table 5.4.3-25. Future designers are urged to verify that their designs do not fail any of the issues posed in that Table. Specifically, designers using Li, V15Cr5Ti, and to a lesser extent 17Li83Pb, are strongly urged to not use water anywhere in the nuclear island. That is, replace water in limiters, shields, resistive choke coils, halo scrapers, and direct converters. Designers using 17Li83Pb, FLIBE, or any of the solid breeders are strongly urged to pay special attention to limiting tritium effluent.

In conclusion, given the current understanding and analysis of the impact of various uncertain issues, the most attractive blankets from the safety standpoint are as follows:

o Li_2O/He/HT-9	(<u>best</u> if tritium control better and chemical control worse than the reference case)
o Li/He/HT-9	(similar to the above, except more difficult chemical reaction control and easier tritium control; more of a compromise)
o Li/Li/V15Cr5Ti	(<u>best</u> if chemical control better and tritium control worse than the reference case)
o 17Li83Pb/ 17Li83Pb/V15Cr5Ti	(similar to the above, except more difficult tritium control and easier chemical reaction control; more of a compromise)

Two of these are helium-cooled HT-9 concepts. Two are liquid-metal, self-cooled V15Cr5Ti concepts. The preferred breeder for helium-cooled HT-9 is Li_2O if tritium effluent control is favorable; a lithium-breeder would be the backup. The preferred liquid-metal for the liquid-metal-cooled V15Cr5Ti concept is lithium if air and water reactions are adequately controlled; 17Li83Pb would be the backup. The Safety Evaluation results show that water and Nitrate Salt coolant and PCA structure score poorly relative to the other concepts studied. These findings are similar to those in References 5-6. In that study the best near-term option was identified to be (Li or 17Li83Pb)/He/HT-9 versus (Li_2O or Li)/He/HT-9 here. The best advanced option in that study was identified to be Li or 17Li83Pb-cooled V15Cr5Ti, the same as in the current study. Finally, it should be mentioned that some blankets that could be extremely attractive from the safety standpoint, e.g. helium-cooled, Li_2O or Li_2SiO_3/Be-breeder, SiC, were not examined in the BCSS.

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5.5 R&D Evaluation Methodology

This category of evaluation of blanket options in the BCSS is somewhat different in purpose and approach than the three areas discussed in sections 5.2 to 5.4. Blanket options employing more advanced materials and design features will likely score higher in the performance related evaluation categories of engineering feasibility, economics, and safety and environmental effects. On the other hand, the more advanced blankets will likely require more research and development resources, will take longer to develop, and may entail a higher risk in terms of favorably resolving the major R&D issues. Thus, this evaluation category attempts to provide the BCSS evaluation process with a balanced view with respect to better performance and the associated larger R&D resource requirements and risks.

The basic approach in the R&D evaluation is to use two figures of merit: one which measures the overall R&D investment "cost" in terms of time, funding and new facilities, and one which measures the "risk" of R&D in terms of the number of key issues to be resolved for a particular blanket option (i.e., the larger the number and importance of the issues, the greater the risk).

5.5.1 R&D Investment Cost

The R&D investment cost (RDI) score is a composite of three ratings (X_1, X_2, and X_3) shown in Table 5.5-1. A judgment is made of the total time (X_1), average annual R&D operating cost (X_2), and required facilities cost (X_3) to develop a particular blanket option to the point where sufficient information would be available to select that blanket with reasonable confidence for inclusion in a demonstration reactor. (For purposes of the study, a demonstration reactor [1] first wall/blanket is assumed to operate at a neutron wall loading of ~ 2 MW/m^2. The fluence on the first wall/blanket before changeout would be ~ 10 MW-yr/m^2.) The numerical values for X_i range from one to three points.

TABLE 5.5-1. RDI CATEGORIES

Time Scale	Score X_1	Average Annual Operating Cost	Score X_2	Required Facilities	Score X_3
< 10 years	(1)	< \$5M	(1)	No New Facilities > \$10M	(1)
10 years → 20 years	(2)	\$5-20M	(2)	New Facilities \$10-\$50M	(2)
> 20 years	(3)	> \$20M	(3)	New Facilities > \$50M	(3)

The composite score for RDI is then given by

$$RDI = \frac{X_1 + X_2 + X_3}{3} \quad , \qquad [5.5-1]$$

where the lowest score is one. A low rating is desirable in the sense that it indicates that the blanket can be developed for the DEMO for less R&D resources than a blanket with a higher score.

In assigning the values for X_1, X_2, and X_3 in Table 5.5-1, a judgment is made of the time, annual cost, and facility needs for all aspects of the blanket concept including structural material, breeder, coolant, and neutron multiplier as well as additional component and engineering development needs. As an example, let us take the Li/He/FS blanket option. First, estimates are made with respect to time scale, annual cost, and facilities costs for ferritic steel for unirradiated materials properties, welding/fabrication, corrosion/compatibility, and irradiated properties. Then, a similar judgment is made for the lithium breeder issues such as basic properties, corrosion/compatibility, magnetic effects and tritium recovery for the three areas of time, cost and facility needs. A similar judgment is also made for helium coolant issues. Finally, an overall judgment is made for the blanket as follows:

- The time scale score (X_1) is determined by the longest estimate for any element in the blanket, i.e. research in structural materials, coolant and breeder is assumed to be carried out in parallel.

- The annual operating cost (X_2) is a summation of the estimates for all elements of the blanket (structure, . . .).

- The facility cost (X_3) is also a summation of the estimates for all elements of the blanket.

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5.5.2 R&D Risk

It is desirable to assess the relative risk in carrying out research and development efforts for a particular blanket option. The greater the likelihood and potential consequence of a number of key issues or "potential flaws" of a blanket option, the more likely one will invest R&D resources which may turn out not to result in a workable fusion blanket; hence, this represents an R&D risk.

The basic figure of merit for R&D risk (RDR) for each blanket option is as follows:

$$RDR = \sum_{i=1}^{n} C_i \times P_i \quad , \qquad [5.5-2]$$

where N is the number of key issues for a particular blanket. C_i is the consequence of the key issue on blanket performance and is rated as follows:

Relative Consequence	C_i Point Value
• low impact	1
• moderate impact	2
• severe impact	3

P_i is the relative probability that the key issue will be a problem and is rated as follows:

Relative Probability	P_i Point Value
• unlikely	1
• about even (~ 50/50)	2
• likely	3

In general, the blankets will score better in this category if they rate low in the RDR score.

5.5.3 R&D Evaluation Composite Rating

An overall R&D evaluation figure of merit (RDFM) for each blanket has been developed as follows:

$$RDFM = \frac{30}{RDR} + \frac{1}{RDI} \quad .$$

This formulation acknowledges that it is desirable for each blanket to show a low R&D investment cost (RDI) and a low value of the R&D risk factor (RDR); thus, blankets with those features will rate higher in the R&D figure of merit (RDFM). The factor 30 is included to give approximately equal weight between the investment cost and risk factors.

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