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Special Purpose Materials For The Fusion Reactor Environment

A Technical Assessment

February 1978

U.S. DEPARTMENT OF ENERGY

Assistant Secretary for Energy Technology

✓ Office of Magnetic Fusion Energy

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Special Purpose Materials For The Fusion Reactor Environment

A Technical Assessment

Prepared by:
The Task Group on
Special Purpose Materials

February 1978

J. L. Scott, Chairman

U.S. DEPARTMENT OF ENERGY
Assistant Secretary for Energy Technology
Office of Magnetic Fusion Energy

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Preface for the Technology Assessment of the
Department of Energy
Division of Magnetic Fusion Energy
Development and Technology Program

During the past year, the Development and Technology Program has performed a series of assessments of the technologies required for a successful fusion power reactor. Technology Assessment is used here as the term to represent the evaluative process of determining (a) the requirements placed upon technology by fusion power reactor designs, (b) the fusion power reactor performance that might be achieved if future technology capabilities can be achieved, and (c) the state of the art in fusion technology compared to reactor requirements. Technology comprises that body of knowledge and experience which is needed to design, construct and operate practical fusion power reactor systems and to predict their performance and costs.

The usefulness of Technology Assessment to the Department of Energy (DOE), Division of Magnetic Fusion Energy (DMFE), lies in its assessment of the differences between ultimate technology requirements and current technology capabilities in relation to useful and practical fusion power reactor systems. Such an assessment can be used to set research and development objectives, to allocate resources, and to measure progress toward fusion power development.

The specific objectives of the series of energy Technology Assessments are as follows:

- To identify and define technology requirements associated with developing commercial fusion power.
- To assess the current state of these technologies.
- To identify extremely difficult problems which could preclude major near-term commitments to programs strongly dependent on that technology.
- For those problems which seem amenable to practical solution, to define the character, timing and effort needed to develop the technology for application to Experimental Power Reactors, Engineering Test Reactors, and full-scale power reactors.

Technology Assessments have been performed in the following six specific areas:

Blanket and Shield
Plasma Heating, Fueling and Maintenance
Materials for Intense Radiation Environments
Special Purpose Materials for Fusion Reactors
Energy Storage and Transfer
Superconducting Magnet Development

Although gaps in knowledge and experience in plasma physics and fusion reactor engineering subjects have been and will continue to be identified, it is not the intent of the present technology assessments to fill these gaps. Instead, the technology assessments have served their purpose when such gaps and problems are identified and quantitatively defined.

These technology assessments strive to point out known problems and highlight differences between technology requirements and technology capabilities. Since both requirements and capabilities are expected to change with time and further work, the differences should be projected only for a reasonable period into the future. In other words, a technology assessment is a snapshot in time. Further work, innovation and inventions can change the state of technology within a relatively short period (on the order of two years) and future assessments will be needed to re-evaluate and revise the conclusions presented here.

11/14/77
James M. Williams, Assistant Date
Director for Development
and Technology
Division of Magnetic Fusion
Energy

TABLE OF CONTENTS

	<u>Page</u>
1.0 SPECIAL PURPOSE MATERIALS_____	1-1
2.0 EXECUTIVE SUMMARY_____	2-1
2.1 BREEDING MATERIALS_____	2-1
2.2 COOLANTS_____	2-1
2.3 TRITIUM BARRIERS_____	2-2
2.4 GRAPHITE_____	2-2
2.5 CERAMICS_____	2-3
2.6 HEAT-SINK MATERIALS_____	2-3
2.7 MACNET MATERIALS_____	2-3
3.0 TECHNICAL ASSESSMENT_____	3-1
3.1 BREEDING MATERIALS_____	3-1
3.2 COOLANTS_____	3-3
3.3 TRITIUM BARRIERS_____	3-5
3.4 GRAPHITE AND SILICON CARBIDE_____	3-5
3.5 CERAMICS_____	3-7
3.6 HEAT-SINK MATERIALS_____	3-9
3.7 MAGNET MATERIALS_____	3-9
4.0 APPENDIX QUESTIONS AND ANALYSES FORMING THE FRAMEWORK FOR THE TECHNICAL ASSESSMENT_____	4-1
4.1 BREEDING MATERIALS_____	4-1
4.2 COOLANTS_____	4-14
4.3 MATERIALS A FOR TRITIUM CONTROL_____	4-29
4.4 ASSESSMENT OF GRAPHITE, BORATED GRAPHITE, AND SILICON CARBIDE FOR FUSION REACTOR BLANKETS_____	4-32
4.5 CERAMICS_____	4-51
SUMMARY_____	4-72
4.6 HEAT-SINK MATERIALS_____	4-74
4.7 HIGH FIELD SUPERCONDUCTION MAGNETS_____	4-77
ATTACHMENT_____	A-1

1.0 SPECIAL PURPOSE MATERIALS

The objective of the Task Group on Special Purpose Materials is to assess materials problems not addressed by the other task groups. Components that were considered include insulators, materials for high-field (>10 T) superconducting magnets, coolants, breeding materials, neutron multipliers, barriers for tritium control, materials for compression and OH coils and waveguides, graphite and SiC, heat-sink materials and current breakers for tokamaks. It is recognized that there will be many other materials problems that will arise during the design and construction of large magnetic-fusion energy devices such as TNS (The Next Step) and DEMO. Most of these problems will be specific to a particular design or project and are the responsibility of the project, not the Materials and Radiation Effects Branch. Consequently, the Task Group on Special Purpose Materials has limited its purview to *crucial and generic materials problems* that must be resolved if a given class of devices such as mirrors, pinches, or tokamaks is to succeed.

Within the scope defined above, the Task Group on Special Purpose Materials has conducted state-of-the-art assessments, and has identified the crucial problem areas which require attention. For these crucial problems, recommendations are made to the Materials and Radiation Effects Branch, DMFE (Division of Magnetic Fusion Energy), for necessary materials and processes development programs to be implemented.

Once these programs are underway at the various contractor organizations, the Task Group, through its subgroups, will conduct periodic reviews and evaluations of the progress being made on these programs relative to the original goals and the expected changes in goals; and recommend to DMFE any needed alterations to the conduct of these programs, to ensure continued correspondence of program activities to the needs of commercial fusion power.

The members of the Special Purpose Materials Task Group are indicated in Table 1.1. In addition, several subtask groups have been organized. The membership of these subtask groups is given in Table 1.2.

Table 1.1. Members of special purpose materials task group

J. L. Scott, ORNL, Chairman	Fred Fickett, NBS, Boulder
C. J. Long, ORNL, Secretary	W. J. Fretague, EBASCO
E. N. C. Dalder, ERDA	Robert Gold, Westinghouse
John Davis, McDonnell-Douglas	R. G. Micich, Grumman
James Dickinson, LASL	D. L. Smith, ANL
Norbert Elsner, GA	

Table 1.2. Members of special purpose materials subtask groups

A. Ceramics

F. W. Clinard, Jr., LASL, Chairman
J. L. Bates, BNW
M. M. Cohen, ERDA
R. H. Condit, LLL, Livermore
J. M. Dickinson, LASL
N. B. Elsner, GA

B. Coolants and Breeding Materials

D. L. Smith, ANL, Chairman
J. H. DeVan, ORNL
B. J. Fillo, BNL
R. E. Gold, Westinghouse
S. M. Rosenwasser, GA

C. Graphite

W. P. Eatherly, ORNL, Chairman
G. B. Engle, GA
W. J. Gray, BNW

D. Heat-Sink and Q-Enhancement Materials

J. M. Dickinson, LASL

E. Magnet Materials

F. Fickett, NBS, Boulder, Chairman
B. S. Brown, ANL
W. Chen, GA
D. Deis, LLL, Livermore
C. J. Long, ORNL

F. Tritium Handling

V. A. Maroni, ANL, Chairman
G. C. Abel, MLM
J. L. Anderson, LASL
J. T. Bell, ORNL
T. S. Elleman, N.C. State Univ.
S. Steward, LLL, Livermore
W. A. Swansiger, SLL

The methodology for developing this technical assessment was evolutionary. At the initial meeting of the Special Purpose Materials Task Group, November 11-12, 1976, at ERDA Headquarters, a comprehensive list of materials-problem areas was defined and task group members were asked to address them. It was concluded that the emphasis would be placed on the crucial, generic problems that must be addressed if commercial fusion power is to be achieved. Consequently, the scope of the SPM Task Group is limited to the problem areas outlined in Table 1.2.

Shortly after being commissioned to develop a program plan for special purpose materials, the task group was asked by Marvin Cohen of DMFE to conduct a technical assessment of the materials problems being addressed under the new scope. This assessment served to focus the attention of the SPM Task Group on the few crucial, generic problems. Each subtask group will develop detailed program plans, task by task, in their area of purview to address these problems.

2.0 EXECUTIVE SUMMARY

Even though to date most of the attention has been directed toward materials problems associated with the first-wall-blanket structural material and the plasma-materials interaction, there are a number of other crucial and generic materials problems that must be solved before commercial fusion power is addressed. These problems include:

- a. Selection and performance of the breeding material
- b. Selection of the coolant and its compatibility with the structure, breeding material, primary heat exchanger, and the like
- c. The development of a suitable tritium-barrier material
- d. The performance limits of graphite
- e. The definition of the operating requirements of ceramics and demonstration of their ability to meet the requirements
- f. The development of suitable heat-sink materials
- g. The development of magnet materials for use above 10 T if required

These problems were addressed by the Special Purpose Materials Task Group in their part of the technical assessment of materials.

2.1 BREEDING MATERIALS

The advantages and disadvantages of liquid lithium, solid-lithium compounds, and fused lithium-bearing salts were compared. Liquid lithium was selected as the most promising breeding material even though there are serious problems associated with its use, including:

- a. Startup and shutdown since it melts at 180°C
- b. Compatibility with the structural material
- c. MHD (magnetohydrodynamic) pumping problems if Li is also used as coolant
- d. Tritium inventory and recovery

These are both design and materials development problems.

2.2 COOLANTS

There is insufficient information at this time to select the best coolant. Liquid lithium would have the advantage of design simplicity if it is used both as a breeding material and coolant. There are the disadvantages of MHD pumping losses and associated high stresses in blanket structural components and compatibility with structural materials. Austenitic stainless steels in contact with lithium will be limited to a maximum temperature of 500°C by compatibility. Because of the high

solubility of Ni in liquid Li, nickel-base alloys are not recommended for use with Li. Vanadium and titanium alloys are compatible with Li provided that careful attention is paid to nitrogen, oxygen, and carbon impurities and to the potential for dissimilar-metal mass transfer, if the primary heat exchanger is a different material from the blanket structural material.

A second candidate coolant is helium. Even though pure helium is inert, there will be a sufficient level of impurities in a commercial reactor system to preclude the use of vanadium alloys as structural material. There may also be a problem with the compatibility of titanium alloys. Either austenitic stainless steels or nickel-base alloys are compatible with commercial grade helium. Even though this helium is not completely compatible with liquid lithium, the consequences of a reaction are negligible. The primary problem will be the inadvertent venting of high-pressure helium into the low-pressure lithium circuit. Suitable rupture disks can probably be designed to circumvent this problem.

If helium is used in conjunction with nickel-base alloys, a solid breeding material such as LiAlO_2 might be utilized, but the compatibility of solid-lithium compounds with commercial helium is highly uncertain.

In the event that the pressure stress in the first wall is too high when Li or He is used as a coolant, taking into consideration the unavoidable high thermal stresses, a low-pressure coolant such as a fused salt may be required. This fused salt should meet the following criteria:

- a. Low melting point ($<250^\circ\text{C}$)
- b. Compatible with the structure
- c. Compatible with the breeding material
- d. Good heat-transfer characteristics
- e. Good radiation stability

A program to find a coolant that meets these criteria should be initiated.

2.3 TRITIUM BARRIERS

A tritium barrier material is required to prevent excessive tritium transport into the steam supply system. The primary material of choice is an in-situ-forming, self-healing oxide. This material should be achievable through the use of conventional steam generator materials possibly with deliberate minor alloy additions.

2.4 GRAPHITE

The primary application of graphite in fusion reactors is as a neutron reflector and moderator for shielding purposes. By its use, the lithium inventory of the system can be reduced. The use of graphite as a protection against first-wall damage is not recommended unless the

alloy development program fails to provide a first-wall structural material with a lifetime of 10-40 MW years/m². Silicon carbide should be considered as a low-Z first wall, but its use as an all-ceramic blanket structural material should be given a low priority.

2.5 CERAMICS

There are a number of places where ceramics are required in magnetic fusion energy devices. These include:

- a. Theta-pinch first-wall insulator
- b. Tokamak current breaker
- c. Neutral-beam injector insulators
- d. Mirror direct-convertor insulators
- e. Low-Z first wall and liners
- f. Insulators for tokamak rf heating systems
- g. Insulators for magnetic-field coils

Several of these applications require that the insulator survive the same irradiation environment as the structural first wall. There are several potential candidate ceramics such as spinels, yttrium aluminum garnet, Si₃N₄, etc., that may be able to meet these criteria; but an extensive development program will be required to find a suitable material for each application.

2.6 HEAT-SINK MATERIALS

Heat-sink materials, probably of several different types, will be required in magnetic fusion energy devices for beam stops, armour, limiters, divertor targets, calorimeters, direct convertors, and others. At present, there are insufficient design data available to make materials choices for these applications.

2.7 MAGNET MATERIALS

There are no major materials problems envisioned for conventional Nb-Ti superconducting magnet systems, except for the establishment of the radiation lifetime and the development of an adequate data base for design. There are a number of materials problems associated with the development of high-field (>10 T) superconducting magnet systems. Two especially important problems are the development of a strain-tolerant Nb₃Sn superconducting composite and the establishment of radiation limits for high-field superconducting magnet systems. There is also a need for nonconducting composite structural materials if the magnetic-field rise times turn out to be quite short.

3.0 TECHNICAL ASSESSMENT

The technical assessment of SPECIAL PURPOSE MATERIALS is based on the same set of considerations that were used for the first wall/blanket structural materials. The considerations are:

- Primary and backup materials of choice
- Testing and manpower requirements
- Major milestones
- Natural resources and industrial capability
- Minority conclusions

Each will be individually discussed. In view of the broad range of materials and components covered by SPECIAL PURPOSE MATERIALS, each of the following components will be assessed separately:

- a. Breeding Materials
- b. Coolants
- c. Tritium Barriers
- d. Graphite and SiC
- e. Ceramics
- f. Heat Sink Materials
- g. Magnet Materials

3.1 BREEDING MATERIALS

3.1.1 Primary and Backup Materials of Choice

Liquid lithium is presently considered a primary choice for breeding material. It has the following advantages:

- a. Good breeding potential without enrichment in ^6Li or neutron multipliers
- b. Excellent heat-transfer fluid
- c. No radiation damage
- d. Use of Li as both breeding material and coolant simplifies the design

There are also problems with liquid lithium that must be addressed in the development program:

- a. Tritium removal and inventory
- b. MHD pumping power and stressing in the first wall
- c. Compatibility with container and coolant
- d. Startup and shutdown procedures

A number of other breeding materials such as FLIBE and solid-lithium compounds — Li_2O , LiAlO_2 , and Li_7Pb_2 — have been proposed as alternatives because of these problems. Each system proposed introduces its own serious problems, which appear to be more difficult to resolve than those associated with liquid Li.

The backup breeding material chosen is the best solid-lithium compound to be determined on the basis of a development program. It would be used in conjunction with a nickel-base alloy structure and helium as coolant.

Problems that must be resolved with solid breeding materials are:

- a. Irradiation stability at high fluences ($\sim 10^{23}$ n/cm²)
- b. Compatibility with helium and the structure
- c. Tritium recovery after high burnups
- d. Methods of cooling the system, especially when enriched with ^6Li
- e. Chemical and structural stability at high temperatures
- f. Breeding potential
- g. Recycling and reuse of breeding material

The use of beryllium as a neutron multiplier is strongly discouraged because of its unique role in other applications such as windows for x-ray devices. It would be immoral to base the fusion economy on the scarce and unique element.

3.1.2 Testing and Manpower Requirements

The major testing required for liquid Li involves liquid-metal compatibility with candidate structural materials, the resolution of the MHD pumping-power problem by clever design plus materials development (e.g., insulated pipe), and work on the engineering of a tritium-removal system.

Relative to the backup material, a solid-lithium compound, a series of survey irradiations to high burnup should be conducted under conditions simulating as closely as possible a fusion-reactor blanket environment. These tests should demonstrate good radiation stability to high burnups in a helium-purged system at temperatures and heat-generation rates equal to those in the intended application. The quantitative release of tritium should also be demonstrated. At the conclusion of the tests, a decision to pursue further the best or drop these materials would be made.

3.1.3 Major Milestones

The major milestones are listed in approximate chronological order:

- a. Determine the temperature limits for lithium compatibility with candidate structural materials for system operating goals of 10-40 MW years/m²
- b. Demonstrate the extraction of T from Li on an engineering scale
- c. Develop blanket designs which minimize MHD problems and demonstrate the manufacture of insulated pipe where required
- d. Demonstrate the radiation stability and in-situ tritium recovery from solid breeding materials to operating goals of 10-40 years/m²

3.1.4 Natural Resources and Industrial Capability

There are ample resources and industrial capability for lithium. There appear to be insufficient resources of Be, especially considering that it is a unique and valuable element for other uses. The use of bismuth as a neutron multiplier also appears to be resource limited. There is no resource limitation associated with the use of Pb as a neutron multiplier.

3.2 COOLANTS

3.2.1 Primary and Backup Materials of Choice

The most viable coolant/structural/breeding materials combinations are shown in Table 3.2.1.

Table 3.2.1. Most viable coolant/structural/breeding materials combinations

Coolant	Breeding material	Structure
Lithium	Lithium	Austenitic stainless steels (<500°C)
Lithium	Lithium	Titanium or vanadium alloy
Sodium	Lithium	Austenitic stainless steel
Helium	Sodium-lithium compound	Nickel-base alloy

The major problems associated with the use of lithium as coolant are (a) compatibility with the containment material and (b) MHD (magneto-hydrodynamic) effects created by the strong magnetic fields. The use of lithium as coolant and breeding materials with austenitic stainless steels simplifies the design problems, but the corrosion due to flowing lithium may be severe enough that sodium would be a preferred coolant choice.

The use of lithium coolant with titanium- or vanadium-base alloys involves the presumption that interstitial impurities such as C, O, or N can be carefully controlled and that the blanket circuit is compatible with primary heat exchanger circuit so that interstitial impurity or dissimilar-metal mass transfer will not be excessive.

There are a number of problems associated with solid-lithium compounds such as radiation stability and compatibility with a helium purge stream that must be resolved before the helium/solid-lithium-compound/nickel-base-alloy option can be given the blessing of the materials community.

3.2.2 Testing and Manpower Requirements

Both the MHD pumping problem and materials compatibility problems with lithium should be resolved in a timely manner. These issues should be resolved in an engineering-scale test facility, involving realistic magnetic fields, temperatures, heat fluxes, and flow rates. It is unlikely that radiation damage will seriously alter the corrosion kinetics.

The issues associated with the use of titanium- or vanadium-base alloys or structural materials should be addressed before extensive work on their reactions with coolants is warranted.

Extensive work done for the LMFBR (Liquid-Metal Fast Breeder Reactor) should be used as a basis for the evaluation of sodium as a coolant provided the corrosion problems with lithium are serious at realistic operating temperatures.

The validity of the assumptions associated with the use of solid breeding materials such as good radiation stability, good tritium release, adequate thermal conductivity, and compatibility with commercial helium should be established at an early date.

3.2.3 Major Milestones

- a. Resolve the compatibility and MHD problems associated with the use of lithium-austenitic stainless steel systems in engineering-scale loops.
- b. Establish that the solutions to the MHD problems such as insulated piping are viable for titanium- or vanadium-base alloys.
- c. Establish the compatibility relationship between the blanket and the primary heat exchanger circuits.
- d. Establish the compatibility between commercially pure helium and solid breeding materials.

3.2.4 Natural Resources and Industrial Capability

The use of lithium as a coolant may involve the use of insulated piping. This is an extension of existing technology. No natural-resource limitations are involved with the use of lithium, sodium, or helium as coolants provided that the present helium supply is stored instead of being squandered.

3.3 TRITIUM BARRIERS

3.3.1 Primary and Backup Materials of Choice

A tritium barrier is required to prevent excessive tritium transport into the steam supply system. The primary material of choice is an in-situ-forming, self-healing oxide. Insufficient work has been done to identify the primary system of choice or the backup.

3.3.2 Testing and Manpower Requirements

An extensive test program is required on tritium migration through commercial and developmental steam generator alloys under realistic reactor conditions of temperature and steam pressure. There is also a need for an expanded permeation data base, including the effects of bulk and surface impurities, influence of alloying elements, and studies of low driving pressures ($P \geq 10^{-6}$ torr) using tritium. The effects of helium formation from tritium decay on permeation rates through metals and on the integrity of permeation barriers should also be determined.

3.3.3 Major Milestones

- a. Complete assessment of commercial steam generator materials
- b. Determine effects of candidate alloying additions on the stability of oxide films
- c. Develop specifications for optimum steam generator alloy for tritium service
- d. Fabricate and test steam generator under service conditions

3.3.4 Natural Resources and Industrial Capability

The optimum alloy is envisioned to be a minor modification of a currently commercial alloy. No natural resource limitations are envisioned.

3.4 GRAPHITE AND SILICON CARBIDE

3.4.1 Primary and Backup Materials of Choice

Graphite: Commercially available graphites have been developed for large-scale usage in the High-Temperature Gas-Cooled Reactor, specifically aimed to minimize neutron damage in the temperature range 600-1200°C.

Manufacturers can be expected to produce similar materials if a commercial market develops.

Silicon Carbide: Bulk silicon carbide (β -phase) is commercially available from a number of vendors in plate-like shapes.

Boron Carbide: This material is also produced in large scale by several vendors and is commercially manufactured as dispersed in a graphite matrix.

3.4.2 Testing and Manpower Requirements

Two major testing programs are recommended to establish feasibility only. Further work would be required to establish permissibles in the post-1980 period. The irradiation tests proposed are in the High-Flux Isotope-Reactor target positions, for which 12-months' reactor time is required to attain fluences of $\sim 3 \times 10^{22}$.

Materials	— Bulk graphite, bulk SiC, fibrous graphite
Temperatures	— 1400, 1600, 1800°C for graphite; 1500°C for bulk SiC
Fluences	— 3×10^{22} n/cm ² in three successive capsules
Measurements	— Dimensions; mechanical properties, including thermal shock for SiC

A lower-priority effort is suggested in exploratory graphite development, leading toward a doubled lifetime ($4-5 \times 10^{22}$ n/cm²). This program should be initiated early because of the long lead times required, including demonstration by irradiation testing.

3.4.3 Major Milestones

Irradiation Testing:

Facility design	— Initiate, beginning FY 1978 Complete July 1978
Begin irradiations	— October 1978
Complete irradiations	— (a) January 1979 (b) October 1979 (c) June 1980
Final reports	— September 1980
Manpower required	— 3 Professional man-years/year

Graphite Development:

Initiation of program	— Beginning FY 1978
First samples available for test	— March 1978
Second-generation material available	— FY 1979
Manpower required	— 1.5 Professional man-years/year

3.4.4 Natural Resources and Industrial Capability

Neither SiC nor graphite present a significant problem in terms of industrial capability. In the event of large-scale fusion energy generation, existing production facilities would require expansion. Graphite is currently manufactured from petroleum coke and is subject to the problems presented by that energy source. There is no expected raw-material (petroleum) problem facing the industry until the 1990s. There is little question that nuclear graphites could be developed from coal cokes and/or pitches, although a commercially attractive route remains to be developed and demonstrated.

3.5 CERAMICS

3.5.1 Primary and Backup Materials of Choice

There are seven major applications for ceramics that were identified in this assessment. Each application has its own unique materials requirements. At present, there is insufficient information on the radiation response of ceramics to identify a primary material of choice. Instead, we have chosen to list several candidate materials for each application. (See Table 3.5.1.)

3.5.2 Testing and Manpower Requirements

In three of the applications of ceramics listed in Table 3.5.1 (theta-pinch first-wall insulation, low-Z first walls, and liners and insulators for tokamak rf heating systems), the ceramics are exposed to the same peak fluences and the blanket structural material. A major developmental program will be required for these applications, including: (a) scoping of problem areas, (b) screening of materials, (c) definition of the nature of degradation mechanisms, and (d) optimization. Implicit in all phases are extensive radiation testing, property determination, and fabrication development. The total level of effort should be about 20% of that directed toward alloy development.

3.5.3 Major Milestones

The major milestones are listed in approximate chronological order:

- a. Determine the temperature and fluence limits for structural integrity of candidate ceramics for system operating goals of 10-40 MW years/m²
- b. Survey electrical, chemical, and mechanical properties of "best" ceramics after irradiation
- c. Fabricate mockup ceramic parts to designer's specifications
- d. Evaluate degradation mechanisms
- e. Determine optimum materials for all applications with respect to fabrication, design, radiation stability, electrical properties, chemical stability, mechanical strength, and thermal stability.

Table 3.5.1. Candidate materials of choice
for ceramic applications

Application	Candidate materials
Theta-pinch first-wall insulator	Spinel, YAG, ^a Si ₃ N ₄ , Al ₂ O ₃ single crystals plus glass
Tokamak current breaker	Powdered Al ₂ O ₃ in metal tubes, enamels, anodized coatings
Neutral-beam injector insulators	Glass, glass-ceramics, Al ₂ O ₃ , BeO
Mirror direct-convertor insulators	Al ₂ O ₃ , Si ₃ N ₄ , AlN
Low-Z first walls and liners	SiC, ¹¹ B ₄ C, Al ₂ O ₃ single crystals, spinel, Si ₃ N ₄
Insulators for tokamak rf heating systems	Spinel, YAG, Si ₃ N ₄
Insulators for magnetic-field coils	Castable aluminum silicates, spinel, YAG, Si ₃ N ₄

^aYttrium aluminum garnet.

3.5.4 Natural Resources and Industrial Capability

Since Be has been excluded from emphasis in this assessment, there are not materials resource limitations. There are size limits with respect to current industrial capability. These limits must be considered by designers. Some advances in ceramics fabrication technology regarding size, purity, or costs are envisioned.

3.6 HEAT-SINK MATERIALS

It appears that heat-sink materials, probably of several different types, will be required in magnetic fusion energy devices for several applications. The uses will include beam stops, armour, limiters, divertor targets, calorimeters, and others. Direct convertors for mirror machines are also likely to use heat-sink materials.

3.6.1 Primary and Backup Materials of Choice

There are insufficient design data available to make materials choices for the applications.

3.6.2 Testing and Manpower Requirements

There is insufficient information to define the requirements. Once the requirements have been established, a fairly extensive development program is envisaged.

3.6.3 Major Milestones

The major milestones are listed as follows:

- a. Develop a conceptual design of one or more components
- b. Evaluate the capability of existing materials to meet the application
- c. Identify candidate materials and the development program required
- d. Develop optimum materials for the application

3.6.4 Natural Resources and Industrial Capability

This is unknown now. Since the amount of material consumed in the application should be small, no material resource limitations should arise.

3.7 MAGNET MATERIALS

This assessment considers only high-field superconducting magnets (>10 T). Conventional Nb Ti superconducting magnet systems were not assessed. Even though no Nb Ti-type superconducting coils have been built of a size required for commercial tokamak or mirror reactors, the problems are considered to be design instead of materials problems with

the exception of radiation damage limits, which are set by the insulation. This problem also arises in high-field conductor systems and was addressed in this assessment.

3.7.1 Primary and Backup Materials of Choice

<u>Primary</u>	<u>Backup</u>
Conductor — Nb ₃ Sn	?
Stabilizer — Cu	Al
Structural material — austenitic stainless steel	Austenitic stainless steel
Insulation — epoxy-fiberglass	Inorganics

3.7.2 Testing and Manpower Requirements

More strain-tolerant Nb₃Sn superconductors are needed for high-field (>10 T) applications in large magnets. Cabling techniques for the conductor or methods of modifying the internal Nb₃Sn layer (waves, twists, or coils) need to be developed. Quality control and inspection techniques for these wires are also needed.

The low-temperature data base on all pertinent existing materials must be expanded to provide a range of choices to magnet designers. In general, new materials will not be necessary, at least through EPR (Experimental Power Reactor) or DEMO (Demonstration Reactor) time development and construction.

If the rise times of pulsed tokamak fields are quite short, the potential for application of composite materials as structure should be evaluated. Experimental programs to evaluate eddy-current losses in various materials are also desirable.

Neutron and gamma-ray damage studies in insulators at low temperatures are greatly needed. The first phase of the program will be screening to evaluate polymer films, bulk organic insulators, and organic matrix composite materials. A careful evaluation of the *total* radiation environment by each of the superconducting coil types is also needed. The effects of penetrations through the blanket and shield are especially troublesome. The degradation of the superconducting composites, especially with Nb₃Sn conductors, requires further assessment experimentally.

Since it is possible that organic insulators may limit the lifetime of superconducting coils to an unacceptably low value, it seems prudent to commence a relatively small screening program to evaluate inorganic insulators in the correct configuration at low temperatures and under thermal cycling.

3.7.3 Major Milestones

- a. Assess the factors which set radiation limits on high-field superconducting magnets
- b. Establish the radiation lifetimes of organic insulation
- c. Evaluate the need for a composite structure
- d. Develop a complete data base for design
- e. Develop methods of fabricating coils with inorganic insulators
- f. Develop strain-tolerant high-field superconducting composites

3.7.4 Natural Resources and Industrial Capability

Resource availability does not appear to be a serious problem in magnet systems. The industrial capability for producing strain-tolerant high-field conductors can readily be established once the material is developed. In a fusion economy, economics may force certain trade-offs (aluminum instead of copper, for example); but most materials required will be available or will have acceptable substitutes. Niobium metal may be the one exception; but even there, it is the quality of the ore rather than its abundance that may create a problem. We believe that techniques can be developed to use the lower grade ores if necessary.

APPENDIX

4.0 QUESTIONS AND ANALYSES FORMING THE FRAMEWORK
FOR THE TECHNICAL ASSESSMENT

Rather than address these questions individually, it was found to be more convenient to address the overall subject.

4.1 BREEDING MATERIALS

4.1.1 Key Questions Concerning Breeding Materials

- a. What are the advantages and disadvantages of lithium in comparison with solid breeding materials?
- b. What solid breeding materials are proposed?
- c. How are they fabricated and handled prior to use?
- d. How long do they operate? What limits their life?
- e. What chemical control is required of the He purge to maintain compatibility?
- f. Will the tritium be released rapidly enough? What is the expected inventory?
- g. Is irradiation-induced sintering a problem?

4.1.2 Response to Questions Concerning Breeding Materials

4.1.2.1 Problem Definition

In presently conceived D-T fusion power reactors, the breeding of tritium seems inevitable and lithium in some form appears to be the only viable candidate for the breeding material. Therefore, establishing the viability of the various breeding blanket concepts should be a major focal point and goal of the fusion materials development program. The forms of lithium that have been proposed are (a) liquid lithium, (b) solid-lithium compounds such as Li_2O , LiAlO_2 , and Li_7Pb_2 , and (c) molten-lithium salts such as LiF-BeF_2 . The considerations and trade-offs regarding the merits of these breeding alternatives are quite complex since the breeding material must not only interface closely with both coolant and structure, but must also be analyzed in terms of radiation, thermal, neutronic, and tritium-processing characteristics. The considerations regarding the viability of the three breeding options are greatly influenced by the coolant considerations, since certain reactor concepts utilize the same material, e.g., liquid lithium and molten-lithium salts, for both the breeding medium and the coolant. Table 4.1.1 summarizes the various blanket/coolant concepts that have been proposed or are considered potentially viable.

Table 4.1.1. Potentially viable blanket/coolant concepts

Coolant/Breeding material	Liquid lithium	Solid-lithium compounds	Molten-lithium salts
Liquid lithium	Yes	No	No
Liquid sodium	Yes	Yes	Yes
Helium	Yes	Yes	Yes
Molten salt	?	?	Yes
Water	?	?	Yes

A close interface between the breeding and structural materials also exists and requires coordination of blanket development with the alloy development programs, particularly with regard to compatibility. Since the proposed structural alloys are generally not compatible with all of the blanket/coolant options listed in Table 4.1.1, the viability and requirements of compatible blanket/coolant options should be a major consideration in the establishment of the alloy development programs. The following sections assess the technology relevant to the three breeding concepts and define the critical areas in which further development is required.

4.1.2.2 Problem Analysis

4.1.2.2.1 Liquid Lithium. In many respects, liquid lithium is considered to be the ideal tritium-breeding material. Acceptable breeding ratios are readily attainable even without enrichment of ^6Li or use of neutron multipliers. Since liquid lithium is an excellent heat transfer fluid, in theory it can also be used for the coolant as well as the breeding medium. Use of the same material for both purposes greatly simplifies the blanket design. Tritium processing and replenishment of the breeding material is readily facilitated in a recirculating system; however, further development is required to demonstrate that adequate tritium-removal efficiencies are attainable by the proposed methods. Other major concerns for the concepts that utilize liquid lithium in this dual capacity relate primarily to effects on pumping and heat transfer in strong magnetic fields and compatibility with the containment material. Attempts to mitigate these problems have resulted in alternate concepts that utilize liquid lithium as a semistagnant breeding medium with a separate coolant, e.g., helium or sodium.

4.1.2.2.1.1 Compatibility. Compatibility with the structural or containment material will have a strong influence on the viability of the liquid-lithium blanket concept. The two major compatibility problems encountered related to: (a) effects on the mechanical integrity of the structural containment and (b) mass transfer of corrosion product that may lead to plugging of circulating systems or excessive deposition of radioactive material in unshielded regions. For the various structural materials considered under the alloy development program, viz., Path A — austenitic stainless steels, Path B — higher nickel Fe-Ni-Cr alloys,

Path C — reactive/refractory metal alloys, and Path D — innovative materials, the operating limitations and compatibility problems are quite different.

For the Path A and B alloys, metallic mass transfer in recirculating lithium systems will limit operating temperatures. The more important aspect of this problem relates to the deposition of the corrosion product rather than the removal of material from the high-temperature, i.e., high corrosion-rate, regions. In addition to the concern for plugging of valves and other restricted regions that results from excessive mass transfer, radioactive mass transfer of Path A and B alloys will lead to difficult shielding problems external to the blanket. Based on an allowable corrosion rate of $15 \mu\text{m/y}$ (a factor of 10 higher than the $1.5 \mu\text{m/y}$ corrosion rate of Type 316 stainless steel at 600°C in high-velocity reactor-grade sodium), the maximum operating temperature is $\sim 500^\circ\text{C}$ for Path A alloys in a recirculating lithium system. Although no data are available with which to assess the extent, slight reduction in the corrosion rates may result from more laminar flow characteristics of lithium in high magnetic fields. Higher temperatures, $50\text{--}100^\circ\text{C}$ higher, are possible for Path A alloys in reactor concepts that utilize a semistagnant lithium blanket with a different coolant. At these higher operating temperatures, intergranular attack will become more important with respect to effects on the mechanical integrity of the Path A alloys. Although not well defined at the present time, compatibility will limit the use of Path B alloys to temperatures below those for Path A alloys. This conclusion is based primarily on the high solubility of nickel in lithium and comparison of limited corrosion data under similar conditions. The better high-temperature properties of the Path B alloys cannot be fully exploited in a liquid-lithium blanket. Since compatibility with lithium will be an important factor in determining the allowable operating temperature for Path A and B alloy-lithium concepts, this problem is critical to the viability of certain fusion reactor design concepts. Therefore, compatibility studies should be given a high priority in the alloy development program for these materials systems in which liquid lithium is used.

It has been demonstrated that several of the Path C metals, e.g., V, Nb, Mo, Ti, and selected alloys, have good corrosion resistance to lithium under certain conditions. It has also been demonstrated that the compatibility of certain of these materials can be severely degraded by the presence of nonmetallic impurities. Excessive oxygen in niobium leads to rapid penetration by lithium. Also, transfer of oxygen to lithium from niobium- or vanadium-base alloys can lead to reduction of the mechanical strength of the structure. Nitrogen and carbon migration in lithium-Path C alloy systems is a major concern. Although nitrogen and carbon transport in selected refractory metal systems has been observed, e.g., in bimetallic systems, the importance of these impurities on the corrosion and mechanical integrity of the Path C alloys have not been investigated in detail. Molybdenum alloys are least affected by these two impurities followed in order by vanadium, niobium, and titanium alloys. In general, it is believed that acceptable corrosion rates of selected Path C alloys in lithium are attainable to temperatures

exceeding 800°C; however, the nonmetallic impurity concentration limits have not been well established.

Compatibility of liquid lithium in Path D materials cannot be generally classified at the present time. However, aluminum is considered unacceptable for use in lithium at reasonable operating temperatures.

A liquid-lithium blanket could conceivably be used with all of the coolants listed in Table 4.1.1. Without question, the combined use of liquid lithium as a coolant is by far the most desirable from a designer's viewpoint. As discussed in more detail below, pumping a highly conducting (electrical) medium in a strong magnetic field produces additional problems. Liquid sodium has been proposed as an alternate coolant for a liquid-lithium blanket. This concept attempts to mitigate the mass-transfer problems associated with circulating liquid lithium in Path A or B alloys. Liquid sodium poses no serious compatibility problems when used in conjunction with a lithium blanket.

Helium is a potential coolant for a liquid-lithium blanket. It poses no serious compatibility problems with lithium; however, transfer of gaseous impurities, e.g., oxygen and nitrogen, to the lithium must be considered. The more difficult problem relating to this concept is selection of the optimum structural material that is compatible with both lithium and helium.

Use of molten salts or water as a coolant for a liquid-lithium blanket is not likely. Although both are excellent heat-transfer media, water and some of the salts considered, e.g., nitrates, hydroxides, and carbonates, are highly reactive with lithium at elevated temperatures. As a result, the combination of lithium with these coolants inside a reactor is considered undesirable.

Methods of purity control in large lithium systems have received little attention. Impurities, particularly nonmetallic elements, are important with respect to compatibility with certain containment materials. However, the permissible level of impurities and the difficulty of purity control have not been well established. Other impurities that may lead to unnecessary parasitic neutron absorption or activation have not been systematically investigated. Although this is considered to be a secondary problem, further work will be required if liquid lithium is seriously considered for the blanket.

Lithium melts at 180°C, therefore, problems associated with maintaining lithium in the liquid phase will be encountered. Approximately 5% expansion occurs during melting. The relatively high boiling temperature, 1330°C, and relatively low vapor pressure, 1 torr at 750°C, are attractive for fusion reactor applications. The high thermal conductivity of lithium facilitates heat-removal processes and the relatively high heat capacity provides substantial thermal inertia in the blanket, which tends to reduce thermal transients that result from the cyclic plasma burn. Since lithium reacts readily with oxygen,

nitrogen, and water, it must be isolated from air and water, particularly at elevated temperatures. Atmospheric contamination is a major concern in large lithium systems. Neutron radiation does not significantly affect lithium in the liquid form. Tritium and helium are the predominant transmutation products. Tritium is the desired product from breeding and helium presents no significant problem.

Tritium breeding is the primary purpose of the lithium blanket. The recirculating liquid metal facilitates the tritium removal processes. Proposed methods of tritium removal include molten-salt extraction and transport through a tritium window to a processing fluid such as sodium or NaK. The major problem is to maintain tritium concentrations in the blanket at acceptably low levels, hopefully <1 ppm. Further development work is required to assure that either or both of these methods are adequate. Concern has been expressed over effects of impurities from the salt on lithium compatibility with the containment materials.

A major advantage of liquid lithium over the solid-lithium compound and molten-salt concepts is that breeding ratios substantially greater than unity are relatively easily attainable even without a neutron multiplier. In fact, this may be the dominant consideration in the final selection of the optimum breeding material. Because of the high breeding ratios, reactor designs that breed only in the outer blanket have been proposed. The inner blanket region between the plasma and the major axis serves as a high-efficiency magnet shield, which maximizes the field at the plasma. Enrichment of ^6Li can also be used to further enhance the breeding capability if desired. Since only minor amounts of other elements are present in lithium, parasitic absorption is minimized.

One of the major concerns regarding the use of a recirculating liquid-lithium blanket relates to MHD (magnetohydrodynamic) effects arising from a moving conductor (lithium) in a strong magnetic field. Although there is a high degree of uncertainty as to their severity, two problems have been identified: (a) the high pumping power requirements and (b) containment pressures created by the moving fluid. Design options proposed to mitigate the pumping problem include flowing lithium parallel to magnetic field lines and locating supply headers and manifolds in outer blanket regions where the magnetic field is lower. Modifications to duct surfaces (lowering electrical conductivity) may also reduce MHD (magnetohydrodynamic) pumping problems. Designs that utilize an alternate coolant, e.g., helium, have been proposed to circumvent the MHD pumping problem. Only low lithium recirculation rates sufficient for tritium processing are required in these designs. The effects of high magnetic fields on the containment pressures have not been well established; however, the presence of high MHD pressures will strongly influence the containment design.

Recirculation of the lithium facilitates both the tritium processing and the makeup of lithium in the blanket.

4.1.2.2.2 Solid-Lithium Compounds. As indicated in Table 4.1.1, blanket concepts which use solid-lithium compounds for tritium breeding and liquid sodium or gaseous helium as the coolant are considered technically viable. While it is unclear whether similarly viable designs can be defined for molten-salt or water-cooled blankets, it is probably true that systems employing either of these coolants would use either liquid lithium or a molten salt as the tritium-breeding medium.

A reasonably wide range of lithium compounds and "alloys" have been suggested for this purpose. An incomplete listing of these would include: Li_7Pb_2 , Li_2O , $\text{Li}_{15}\text{Si}_4$, LiAl , LiAlO_2 , and Li_2SiO_3 . Blanket designs proposed to date generally incorporate these materials in the blanket in the form of very fine powders to facilitate tritium diffusion from the breeding compound. This, in turn, imposes the requirement for canning, usually in a porous can disposed in a manner which permits tritium removal via a helium purge stream. Designs using other than helium as a coolant would require a separate, and isolated, helium purge system.

Initial blanket designs using solid-lithium breeding compounds were motivated by one or more of the following advantages:

- a. Solid breeder designs would be free of the problems of MHD pumping losses associated with blanket designs using liquid lithium as the coolant (for those solid breeder designs using gas coolants)
- b. Reduced tritium inventory in the blanket
- c. Reduced coolant/compatibility limitations

Some of the initial optimism regarding the virtues of solid-lithium-compound blankets was diminished when the tritium-breeding potential was examined more closely. However, the merits for these designs are such that they should remain as tentative blanket breeding-design options, at least until the uncertainties regarding their overall engineering credibility can be answered.

Since the solid breeders are present as powders and do not, as a rule, circulate throughout the blanket, compatibility problems are likely to be insignificant. The exceptions to this would be those designs in which chemically reactive coolant and breeding materials are combined. Since the basic chemical reactions are known, these difficulties should be easily avoided by proper systems design and materials selection. A more subtle aspect of the compatibility question might be related to the possibility of contamination of the breeder by the oxygen, water vapor, or other active impurities which might be present as impurities or intentional additions to the helium-gas purge stream.

For those designs where very high temperatures are required to effect high tritium-release rates, compatibility of the structural containment materials with the coolant might prove more limiting than compatibility with the breeding material itself.

The importance of purity control in the solid-lithium compounds is primarily related to the question of neutronics. Since several of the candidate breeding materials are already borderline in their ability to furnish the lithium atom density required to obtain a satisfactory tritium-breeding ratio, unwanted impurities should be strictly controlled. This should pose no significant difficulty since many of the materials considered for these applications are available in reasonably high purities on a commercial basis at the present time.

The thermophysical and chemical/structural properties of most of the candidate solid-lithium compounds are known to the extent necessary for engineering design. Those which are most relevant to fusion reactor design include:

- a. Thermal conductivity
- b. Melting point/boiling point
- c. Heat capacity
- d. Density and molecular weight
- e. Crystal structure
- f. Phase stability and relationships

These data are available in standard handbooks of chemistry and physics. In addition, data on the (measured or estimated) retentivity of tritium and residual radioactivity associated with irradiation in fusion reactor blankets can be found in the recent literature.

The thermal stability of the candidate solid-lithium compounds is important in that fusion reactor blanket designs employing these materials will typically operate in the 450-650°C temperature range; reasonable safety margin should exist so that afterheat during a reactor shutdown or temporary loss-of-coolant event does not cause a meltdown or extreme sintering problem. This presents no obvious difficulty for the ceramic compounds, but should be carefully examined for designs using lower melting materials such as Li_7Pb_2 or $\text{Li}_{15}\text{Si}_4$ where thermal performance might be marginal.

Chemical stability concerns are closely allied with those of compatibility discussed above. Since these are inseparably related to choice of coolant and specific blanket design, individual design evaluations are required.

The radiation stability of the solid-lithium tritium-breeding compounds represents an area of particular importance to the successful implementation of these materials in fusion reactor blankets. At least three separate questions can be posed:

- a. What are the effects of high-fluence neutron radiation on the sintering characteristics of solid-lithium-compound powders?

- b. Will swelling and/or particle attrition due to high helium-gas generation rates be significant?
- c. Will chemical composition changes occurring as a result of neutron-induced reactions alter the stability or tritium-release properties?

With the possible exception of the ceramics such as LiAlO_2 used in the Hanford Coproduct Program, the answers to these questions are largely unknown.

The ability of solid-lithium compounds to breed large quantities of tritium has been rather clearly established in programs at both the Savannah River Laboratory and at Hanford. Whether or not these materials can be incorporated into the power-producing blankets of fusion reactors in a manner which would permit a *net*-plant tritium-breeding ratio greater than unity has not been established. Neither of the previous programs has attempted to continuously remove tritium as it is bred; it appears certain that this would have to be done for fusion reactors.

Tritium extraction and recovery techniques which have been suggested generally rely on the trapping of tritium from a helium stream which has been "doped" with a few ppm of oxygen to facilitate tritium gettering. The first step in each of these recovery schemes is the diffusion of tritium from the solid-lithium compound. For materials which would oxidize in the doped helium, reduced tritium-release kinetics at the gas-solid interface could pose a difficulty. The final judgment of the feasibility of these types of recovery schemes will have to await in-reactor testing at appropriate temperatures and recovery gas pressures/flow rates.

The solid-lithium compounds most frequently suggested for fusion reactor tritium breeding are those combining high lithium-atom density with good thermal properties. Even with these materials, the large amount of blanket volume which must be dedicated to breeder-containment hardware and coolant ducting, particularly for helium gas-cooled designs, makes it extremely difficult to obtain acceptable tritium-breeding ratios (the compounds Li_7Pb_2 and $\text{Li}_{15}\text{Si}_4$ have the highest breeding efficiencies). This deficiency is usually handled by enriching the breeding material in ^6Li , which has a large (n,α) cross section for thermal neutrons, and using a neutron multiplier such as Be to increase the local neutron activity and enhance tritium production. Such measures have to be weighed carefully, however, since enrichment with ^6Li can lead to unacceptably high volumetric power densities which can pose cooling difficulties. In addition, there has been some concern over the resource availability of beryllium as a neutron multiplier.

The possible presence of impurities having large cross sections for parasitic neutron captures also has to be closely evaluated. Such impurities could be present from either the power production processes or as the products of neutron reactions in the breeding material itself.

Since virtually all present schemes which use solid-lithium compounds to breed tritium assume in-situ (i.e., in-the-blanket) removal of tritium, the breeding materials are not recycled in the normal sense that, for example, liquid lithium would be continuously recycled. The residence time of the solid breeding compounds would normally be dictated only by considerations of burnup. Estimates by Brookhaven indicate this could be deferred until as much as 30% of the initial ^6Li was reacted. For most reactor designs, this represents nearly the design lifetime of the blanket; if more frequent blanket-management operations were desired in order to optimize breeding or extraction efficiency, these changeovers could be timed to coincide with other scheduled maintenance or repair operations.

Following their useful irradiation lifetime in the reactor, the lithium compounds would be removed and chemically reprocessed, or, if too radioactive for reprocessing, would have to be stored as wastes. While not trivial, none of these operations would be likely to require the development of new technologies.

4.1.2.2.3 Molten-Lithium Salts (LiF-BeF_2). Among the molten salts of potential interest as lithium carriers, only the fluorides offer the high chemical-binding energies necessary for radiation stability along with reasonable breeding ratios and the potential for tritium recovery. Extensive testing in support of fission reactor programs has provided a detailed data base on the chemical and physical properties of molten Li_2BeF_4 . The chemical stability of the salt makes it particularly attractive for blanket applications, where chemical compatibility with structural and moderator materials is essential. Also, the limited solubility of tritium in the salt greatly facilitates the process of tritium removal. The primary disadvantages of the salt are (a) its relatively high melting point (364°C) and (b) its neutronic properties, which require the use of a neutron multiplier in concert with the salt blanket.

The corrosion properties of LiF-BeF_2 mixtures have been studied over a wide range of temperature and flow conditions. The mixtures are relatively nonreactive toward such metals as nickel, molybdenum, and iron. However, more reactive alloying elements such as chromium, aluminum, titanium, niobium, and vanadium are oxidized by LiF-BeF_2 melts unless highly reducing conditions can be maintained in the melt. Thus, alloys such as Hastelloy N and molybdenum-TZM, based on nickel and/or molybdenum, resist corrosion in these salt mixtures up to temperatures where the strength of the alloy becomes limiting (700°C or above). Path A and B alloys containing chromium as a major addition (greater than 10%) will require some form of chemical control to limit the oxidation potential of the fluoride mixture. Acceptable corrosion rates of Type 316 stainless steel at 650°C have been attained in LiF-BeF_2 systems where the redox potential of the salt was buffered by a beryllium reductant. The redox potentials required to accommodate Path C alloys based on niobium and vanadium appear impractical, given the unique operating features of fusion reactors discussed below.

There are important unknowns in assessing the corrosion behavior of fluoride salts relative to fusion reactor blanket applications. The fraction of tritium carried as TF compared to that carried as T_2 will establish the redox potential of the salt; and, as discussed above, only very low TF: T_2 ratios are permissible with container metals containing chromium as a major alloying addition. Transmutations of beryllium and fluorine in the blanket will also yield oxidants capable of corroding structural metals. Thus, techniques for buffering the redox potential of the salt will be required both to maintain a desired TF: T_2 ratio and to control the corrosion effects of transmutation reaction products.

Molten LiF-BeF₂ salts do not wet or chemically react with graphite. Reactions of the salts with steam or air produce accelerated corrosion of structural materials but do not induce safety-related problems.

As discussed above, some form of redox buffer will be required to control the TF: T_2 ratio and transmutation product buildup in a LiF-BeF₂ blanket system. This is achievable through a coupling reaction such as $Ce^{3+} \rightleftharpoons Ce^{4+}$ or by the presence of reducing metals such as beryllium and/or chromium. Techniques for the initial purification of fluoride salts are well demonstrated.

Molten LiF-BeF₂ salts are characterized by relatively high heat capacities (2.4 J/g·K at 540°C) but relatively low thermal conductivities (1.0 W/m·K at 540°C). This latter property works to the detriment of a semistagnant blanket concept. Molten fluoride salts characteristically undergo a relatively large expansion in transforming from the liquid to solid states. However, in the case of the Li₂BeF₄ composition, the densities of the solid and liquid states are nearly equivalent at the melting point.

LiF and BeF₂ are among the most thermodynamically stable fluoride compounds. The salt with the most favorable breeding ratio is LiF. However, this salt melts at 848°C and could not be used with Path A or B alloys. Of known solutes which lower the melting point of LiF, BeF₂ is the best in terms of neutronic properties. A 50/50 mixture of LiF-BeF₂ melts below 380°C, but is relatively viscous (more than 50 cP at 450°C). The optimum mixture in terms of viscosity and breeding ratio is nearer the composition Li₂BeF₄, which melts at 460°C. Mixtures of LiF and BeF₂ afford excellent thermal and radiation stability in the liquid state but are subject to radiolytic decomposition in the solid state. Vapor pressures of the mixture are relatively low even at 1000°C.

Tritium will exist in molten fluorides as either molecular tritium (T_2) or TF, depending on the oxidation state of the salt. The solubility of T_2 in Li₂BeF₄ has not been measured, but the solubility of H_2 is on the order of 7×10^{-5} moles H_2 /l of salt-atm of H_2 at 1000 K. Assuming the solubility of T_2 to be of this low order, either permeation through a metal membrane or gas sparging would be an effective means for T_2 removal. Although the limited solubility of T_2 in Li₂BeF₄ greatly

facilitates the removal of T_2 , it also requires that the blanket be processed at a relatively high rate to minimize diffusion losses of T_2 through the blanket wall. While TF is more difficult to strip than T_2 , the TF should not diffuse through the blanket walls, and, on this basis, could be allowed to build up to higher pressures than T_2 . Under such conditions, gas sparging should be effective for TF removal, and the rate of processing of the blanket fluid would be reduced from that required where T_2 is the predominant form. However, as discussed previously, significant TF pressures would be permissible only with relatively noble structural materials such as nickel-molybdenum alloys. Also, with larger tritium inventories, the radioactive decay of tritium to ^3He adversely affects the doubling time for breeding.

Neutronic studies of LiF-BeF_2 systems indicate that the tritium breeding ratios attainable with this salt are marginal without some form of neutron multiplication. The breeding ratio can be augmented by neutron multipliers such as lithium, lead, or beryllium or by enrichment of the fluoride salt in ^6Li .

The major parasitic capture reactions in LiF-BeF_2 systems are $\text{Be}(n,\alpha)\text{Li}$ and $^{19}\text{F}(n,\alpha)^{16}\text{N}$. However, beryllium also provides neutron multiplication through the reaction $\text{Be}(n,2n)\text{He}$.

Because of a low thermal conductivity, molten LiF-BeF_2 salts must develop turbulent flow to cool (or be cooled) efficiently. In the absence of a magnetic field, the laminar-turbulent transition occurs at a Reynolds number of about 2200. Calculations of the Hartman number for Li_2BeF_4 indicate that the imposition of a magnetic field, even up to 8 webers/m², will not increase this transition Reynolds number above that normally specified for molten-salt heat transfer systems. This implies that molten Li_2BeF_4 can be made to flow turbulently within the blanket region with state-of-the-art flow circuitry and pump designs. Of greater concern is the electrical potential induced in these salt mixtures by the magnetic fields at the required flow velocities (see 4.1.2.2.3).

Tritium breeding and parasitic capture reactions in LiF-BeF_2 will affect slight compositional shifts, but the reaction cross sections are sufficiently small that the salt make-up requirements are nominal.

4.1.2.3 Summary of Critical Problem Areas

Establishing the viability of the various breeding blanket concepts should be a major objective of the fusion materials program. The importance of this question is far greater than indicated by the scheduled need for breeding blankets, since the direction of other programs, e.g., alloy development, will be strongly influenced by the type of breeding material. From the preceding assessment, the structure and coolant combinations considered most compatible with each of the three blanket types are given in Table 4.1.2. Summarized below are the critical areas where additional information is required before a final assessment of the viability of each blanket concept can be made.

Table 4.1.2. Most viable breeder/structure/coolant combinations

Breeding material	Structure	Coolant
Liquid lithium	Path C (reactive/refractory metal alloys)	Lithium
Liquid lithium	Path A (stainless steel)	Sodium
Solid lithium compounds Li ₇ Pb ₂ , Li ₂ SiO ₃	Path B (Fe-Ni-Cr alloys)	Helium
Molten lithium salt LiF-BeF ₂	Path B (Fe-Ni-Cr alloys)	Molten salt

4.1.2.3.1 Liquid Lithium. Based on the preceding assessment, three key areas are identified where additional analytical and experimental information is required before a final assessment can be made on the feasibility of a liquid-lithium breeding blanket. These include:

a. Electromagnetic Effects. Information regarding the behavior of a highly conductive recirculating liquid metal in a strong magnetic field is essential. What are the pumping power requirements? How important are induced hydraulic pressures? What are the magnetic field effects on heat and mass transfer?

b. Compatibility with Structure: Compatibility limitations are important in establishing the temperature limitations and viability of blanket concepts that utilize liquid lithium with Path A or B structural alloys, particularly in circulating systems. Determination of the rate-controlling mechanisms and temperature limitations should be a major factor influencing alloy development programs. Lithium purity requirements and purification methods required for lithium-Path C alloy concepts should be established, particularly with respect to dissimilar metal mass transfer.

c. Tritium Processing. Capabilities of proposed tritium processing methods for liquid lithium must be established. The acceptability of these processes will influence other materials considerations and operating parameters.

4.1.2.3.2 Solid-Lithium Compounds. Based on the preceding discussions, several areas can be identified where additional analytical or experimental information is required before a final assessment of the feasibility of tritium breeding with solid-lithium compounds can be made. These are summarized as follows.

a. Radiation Stability. A clear answer to the issues of radiation-enhanced sintering, swelling, and compositional changes must be obtained. Data from previous reactor irradiation programs is not appropriate to currently conceived fusion reactor designs.

b. Neutronic Performance. Are neutron multipliers required? Does this present a resource limitation? Blanket designs should emphasize approaches which minimize the need for ^6Li enrichment and neutron multiplication, if possible.

c. In-Situ Tritium Recovery. The practicality of performing efficient in-situ recovery of bred tritium by doped helium purge streams should be examined more closely. In particular, methods for doing this at temperature (and times) consistent with reasonable blanket design should be emphasized.

Only after these problem areas have been more closely examined can the broader issues regarding the utilization of solid breeders be answered. Interpretation of the results of these evaluations should include consideration of how important any remaining uncertainties might be to the overall concept viability.

4.1.2.3.3 Molten-Lithium Salts. The technology surrounding the use of fluoride salts as coolants is relatively well developed. However, there are special considerations in the use of these salts as breeding blankets that cannot be quantitatively addressed from information now at hand. These include the following.

a. Compatibility. The magnitude of the corrosion problem arising from electrical potentials induced by motion of the salts through strong magnetic fields must be determined.

b. Tritium Recovery. Although tritium recovery appears to be a simpler problem for molten fluorides than for molten lithium, there is uncertainty as to what fraction of tritium can be allowed to exist as TF given the constraint of oxidation reactions with the blanket wall. If the TF fraction must be subordinated to that of T_2 , can a stripping technique be developed to hold the T_2 pressure low enough to avoid diffusion losses of this species through the blanket wall?

c. Tritium Breeding. The breeding ratios calculated for Li_2BeF_4 in most reactor design studies indicate that neutron multipliers will be required if this salt is to be used exclusively as the breeding medium. Thus, the design strategies and attendant resource requirements engendered in gaining the requisite breeding ratios from fluoride salts need better definition.

Since extensive background information already exists on LiF-BeF_2 mixtures, the research programs needed to resolve these problem areas appear less formidable than do those for most other blanket candidates. Questions concerning the permissible redox potentials in molten fluorides, how they can be controlled, and the effects of magnetic fields on corrosion kinetics can be studied in related loop experiments. The approaches proposed for T_2 and TF removal derive from well-established chemical engineering technologies and can be proof-tested using hydrogen as a stand-in for tritium.

4.2 COOLANTS

4.2.1 Key Questions Concerning Coolants

- a. What coolants have been proposed?
- b. What are the operating-temperature limits for each?
- c. What materials and under what conditions are compatible and/or incompatible with each coolant?
- d. What purity limits must be maintained in the coolant?
- e. What design pressures are required to remove the heat?
- f. For Li and fused salts, how does one start up the system? shut it down? restart?
- g. How difficult will T removal be from the coolant?
- h. Is an insulator/composite tube required to pump liquid metals?
- i. Are there adequate resources for all coolants?
- j. Do the magnetic fields accelerate the corrosion rate of metals containing fused salts?
- k. Are engineering-scale loops in intense magnetic fields required?

4.2.2 Response to Questions Concerning Coolants

4.2.2.1 Problem Definition

Removal of sensible heat from the first-wall/blanket region of a fusion power reactor poses an important and difficult problem. In addition to being an acceptable heat-transfer medium, the selection of a suitable fusion reactor coolant is based on the following considerations: (a) compatibility with the containment or structural materials, (b) compatibility with the breeding material, (c) susceptibility to radiation damage, (d) behavior in a strong magnetic field, and (e) interface with energy conversion systems. Four types of coolants that have been considered for fusion reactor applications include: (a) liquid metals — lithium and sodium, (b) gas — helium, (c) molten salts, and (d) water-pressurized or steam. As discussed in the previous chapter on Breeding Materials, there is a very close interface between the breeding, coolant, and structural materials. The various coolant types listed above are considered only for certain combinations of breeding materials (liquid lithium, solid lithium compounds, and molten lithium salts) and structural alloys (Path A — stainless steel, Path B — high nickel Fe-Ni-Cr alloys, Path C — reactive/refractory metal alloys, and Path D — innovative materials). Table 4.2.2.1 summarizes the coolant/structure/breeding combinations that have been proposed or are considered potentially viable. The viability of the various breeding material options will strongly influence the selection of the most appropriate coolant. Likewise, the most suitable coolant will depend on the choice of structural alloy and vice versa. The technologies relevant to the

four coolant types are assessed in the following section and the final section summarizes the critical areas in which further development is required.

Table 4.2.2.1. Proposed or potentially viable coolant/structure/breeding concepts

Coolant	Breeding material	Structure
Lithium	Lithium	Path A,C
Sodium	Lithium	Path A,C
	Solid-Lithium Compound	Path A,B,C
Helium	Lithium	Path A,C (Mo Alloy)
	Solid-Lithium Compound	Path A,B,C (Mo Alloy)
	Molten-Lithium Salt	Path A,B,C (Mo Alloy)
Molten Salt	Lithium	Path A,C (Mo Alloy)
	Solid-Lithium Compounds	Path A,B,C (Mo Alloy)
	Molten-Lithium Salt	Path A,B,C (Mo Alloy)
Water	?	Path A,B

4.2.2.2 Technical Assessment

4.2.2.2.1 Liquid Metals. In general, the liquid-alkali metals are excellent heat-transfer fluids. They typically possess high heat capacities and thermal conductivities and produce high heat-transfer coefficients. Considerable experience and technology related to liquid metals as heat-transfer fluids have been developed in the Liquid-Metal Fast Breeder Reactor (LMFBR) program. These liquid metals are particularly effective for heat removal in high-energy density systems where minimal coolant volume and small coolant channel sizes are essential. This is an important consideration in a fusion reactor because of shielding requirements. The two most important problems involving the use of liquid-metal coolants relate to (a) compatibility with the containment material and (b) MHD effects created by the strong magnetic fields. Secondary problems of considerable importance relate to tritium purification and reactivity with air and water. Lithium and sodium are the two prime candidates for a fusion reactor coolant. As discussed in the previous chapter, liquid lithium is also a candidate breeding material. Combined use of lithium for both purposes greatly simplifies the blanket design. The primary interest in sodium as a coolant relates to reduced containment compatibility problems. Sodium can be used in conventional Path A and B alloys at higher temperatures than are feasible with lithium.

4.2.2.2.1.1 Lithium. The primary incentive for use of lithium as a coolant in commercial fusion reactors is the fact that it is also a viable candidate for the tritium breeding medium. Therefore, lithium coolants with the solid-compound or molten-salt breeding concepts are

not considered to be viable options. Since many of the aspects relative to the use of lithium in a fusion reactor blanket have been discussed in the previous chapter on Breeding Materials, only those aspects that are particularly relevant to the coolant application are discussed here.

4.2.2.2.1.1.1 Compatibility. Compatibility limitations of the structural materials will be a major factor in the determination of the viability of blanket designs that utilize liquid lithium as the coolant. The two major compatibility problems relate to (a) deterioration of the mechanical integrity of the structural containment and (b) mass transfer of corrosion product that leads to plugging of circulating systems or excessive deposition of radioactive material in unshielded regions. The latter problem, i.e., the mass transfer, becomes more critical for the coolant application than for the breeding application because of the higher flow rates required. It is particularly critical for the Path A and B alloys, since considerable metallic mass transfer has been observed at temperatures as low as 500°C and the corrosion product is highly radioactive. Based on an allowable corrosion rate of 15 $\mu\text{m}/\text{y}$ (a factor of 10 higher than the 1.5 $\mu\text{m}/\text{y}$ corrosion rate of Type 316 stainless steel at 600°C in high-velocity reactor-grade sodium), the maximum operating temperature for a circulating lithium coolant is 500°C for Path A alloys. The compatibility limit for Path B alloys in circulating lithium systems is lower than that for Path A alloys. This conclusion is based primarily on the high solubility of nickel in lithium and reported corrosion data under limited conditions. Although modifications to the containment surface, e.g., coating, cannot be eliminated as a possibility for reducing the compatibility problem of Path A and B alloys, this type of solution is generally not recommended if preferential or enhanced attack can occur at defects in the surface modification. Substantial research would be required to develop surface modifications that would reliably withstand the severe conditions imposed by an operating fusion reactor.

It has been demonstrated that several Path C metals, e.g., V, Nb, Mo, Ti, and selected alloys, have good corrosion resistance to lithium at temperatures in excess of 800°C under certain conditions. However, the compatibility of some of these materials is severely degraded by the presence of nonmetallic impurities. Excessive oxygen in niobium and high carbon concentrations in some ferrous alloys are prime examples where rapid penetration by lithium has been observed. Also, transfer of oxygen to lithium from niobium- and vanadium-base alloys can lead to a reduction in the mechanical strength of the structure. In general, carbon and nitrogen migration and resultant effects in lithium - Path C alloy systems, are the major concerns. These problems are important primarily in blanket designs in which lithium may contact external sources of nitrogen or carbon, e.g., bimetallic systems (ferrous alloys external to the blanket), graphite moderators, or atmospheric contamination. It is believed that acceptable corrosion rates of selected Path C alloys are attainable; however, the nonmetallic impurity concentration limits have not been well established. Also, methods of purity control in large lithium systems have received little attention.

4.2.2.2.1.1.2 Thermal, Chemical, and Radiation Properties. As discussed more fully in the Breeding Materials section, the thermophysical properties of lithium are well documented and, for the most part, lithium is an excellent coolant for fusion reactor applications. Probably the major concerns in these areas relate to atmospheric contamination in large lithium systems and the fact that lithium reacts readily with oxygen, nitrogen, and water. These effects are not prohibitive, but must be considered in subsequent reactor designs.

4.2.2.2.1.1.3 Tritium and Neutronics. If liquid lithium is used as the coolant, it undoubtedly would also be used as for tritium breeding. The considerations with regard to tritium processing and the neutronic aspects have been dealt with in the Breeding Materials chapter. However, it is important to reemphasize the fact that combined use of lithium as both breeding material and coolant permits a compact blanket design, which is critical with respect to overall reactor cost effectiveness.

4.2.2.2.1.1.4 Recirculating Characteristics. A major concern regarding the use of liquid lithium as a coolant relates to MHD effects arising from a moving conductor in a strong magnetic field. This problem is particularly critical for the coolant application since relatively high flow rates are essential. Although there is a high degree of uncertainty as to their severity, three problems caused by the high magnetic fields have been identified: (1) high pumping power requirements, (2) high hydraulic pressures created by the moving fluid, and (3) reduced heat transfer efficiency. Design options proposed to mitigate the pumping problem include alignment of the coolant channels parallel to the magnetic field lines whenever possible and locating supply headers and manifolds in the outer blanket regions (farthest from the major axis) where the magnetic field is lower. The magnitude of MHD pumping effects for various configurations and directions with respect to the field must be determined to assess overall pumping requirements of proposed blanket designs. The effects of high magnetic fields on the hydraulic containment pressures have not been well established. This may be a major factor in establishing the size and design limitations of lithium blanket modules. The more laminar flow characteristics of an electrical conductor in a magnetic field will also affect the heat transfer properties. Knowledge of this effect is necessary to determine mass flow rate requirements for the coolant. All of these MHD effects are further complicated by the pulsed magnetic fields that are superimposed on the toroidal field. Surface modifications that decrease the electrical conductivity of the channel walls may provide a means of reducing the undesirable MHD effect. However, the magnitude and importance of these MHD effects must be known before the viability of blanket concepts that utilize lithium as a coolant can be determined, and before priorities for development work designed to mitigate the MHD problems can be established.

4.2.2.2.1.2 Sodium. Sodium is known to be a good heat-transfer fluid, and a large technology base for this application has been developed under the LMFBR program. Sodium possesses many of the same favorable and unfavorable characteristics that were discussed above for

lithium. The two major differences, which relate to tritium breeding and compatibility with the structure, are discussed in detail below. The MHD problem for sodium will be much the same as for lithium; however, an extensive technology base for sodium-steam generator energy conversion systems has been developed. In principle, sodium can be used as a coolant for all three of the breeding concepts, viz., liquid lithium, solid lithium compounds, and molten lithium salts.

4.2.2.2.1.2.1 Tritium. Sodium is useful only as the coolant and cannot serve the dual breeding/coolant capacity as can lithium. However, since hydrogen-isotope solubilities in sodium are much less than those in lithium, direct tritium removal to concentrations below 1 ppm in sodium by tritide precipitation processes is possible. This provides a simple and effective method of tritium processing and control. Additional development work is required to further explore these possibilities.

4.2.2.2.1.2.2 Compatibility. The major incentive for use of sodium as a coolant derives from the compatibility considerations with the conventional Path A and B alloys. As stated earlier, compatibility problems will limit the temperature at which lithium can be used as a coolant. Sodium can be contained in Path A and B alloys at substantially higher temperatures than can lithium. For example, the corrosion rate of Type 316 stainless steel in high-velocity reactor-grade sodium at 700°C is about the same as the corrosion rate of stainless steel in flowing lithium at 500°C, viz., $\sim 15 \mu\text{m/y}$. Higher thermal efficiencies can be expected in blanket designs that utilize sodium rather than lithium for the coolant with stainless steel or nickel alloy structures. Sodium would be the more appropriate liquid metal coolant for the solid or molten salt breeding concepts. The extensive technology developed for sodium in the LMFBF programs provides an excellent basis for the present application.

4.2.2.2.2 Helium. Pressurized helium offers several advantages for use as a fusion reactor coolant. Helium is compatible with several commercial structural alloys, does not produce significant radioactive products, is not subject to MHD effects, and possesses favorable characteristics for tritium containment and processing. In addition, considerable technology related to the use of helium as a heat transfer medium has been developed in gas-cooled fission reactor (HTGR) programs. Helium should be adequate for heat removal requirement of $5\text{--}10 \text{ MW/m}^3$ in fusion reactor blankets based on comparative HTGR core designs. As for all gaseous coolants, pumping power losses are high and it is anticipated that a helium pressure of 50-70 atm will be required to minimize this problem. Other major problem areas include containment and leakage of the high pressure coolant and shielding difficulties related to the relatively large channel size requirements. A minimum helium outlet temperature above 400°C for a practical energy conversion system combined with limited heat transfer coefficients results in relatively high temperature requirements for the structure. Helium cooling is considered to be most appropriate for the solid lithium compound breeding concepts with Path A and B, or possibly Path D, structural materials.

4.2.2.2.2.1 Compatibility and Purity Control

4.2.2.2.2.1.1 Impurity Levels in Helium. Although pure high pressure helium is not chemically reactive with blanket and power conversion loop components, it is expected that trace impurities present in the helium might be reactive with high temperature metal surfaces. Reactive impurities in the helium coolant will most likely include H_2 , deuterium, tritium, H_2O , CO , CO_2 , CH_4 , N_2 , or O_2 .

The impurity content of the helium coolant will be determined by ingress through leaks, the structural and nonstructural materials present in the loop, desorption from loop components, component maintenance and replacement schedules, diffusion into the loop from the plasma side or from the power conversion side, transmutations, chemical reactions of impurities between themselves or other materials in the loop, and the characteristics of the coolant purification system.

The principal potential source of water in a fusion reactor will be possible leaks in the steam generator (which would be at pressure on the order of 250 atm) if steam conversion is used and to a lesser extent back diffusion from the lower pressure precoolers heat exchanger if direct cycle turbine conversion is employed. Outgassing of adsorbed water from blanket components such as solid breeder material, porous graphite distributors, and other solid ceramic structural or fibrous insulator materials will be a major but continually decreasing source of water. The reduction of preexisting rust on ferritic heat exchanger surfaces will produce water at a relatively slow rate.

Besides the continuing ingress of deuterium and tritium from diffusion through the first wall, hydrogen production from n,p reactions and tritium from the solid breeder, other sources of hydrogen will be proton diffusion into the coolant from water side corrosion of ferritic steam generator tubes, desorption from graphite components, and reaction of H_2O with graphite in the loop.

Methane might be produced by low temperature ($<600^\circ C$) reaction of H_2 with graphite components and the decomposition of hydrocarbons (oil vapor) from oil lubricated components such as turbine, bearings, pumps, or circulators that operate in the primary coolant.

The reaction of water with graphite components (if employed) will produce some CO , and CO_2 may be present through outgassing of graphite components.

Oxygen and nitrogen can enter the system through system leaks, ingress during maintenance or outgassing of ceramic structural or insulating materials in contact with the helium.

The partial pressures of impurities in fusion reactor helium coolant are design dependent as discussed previously, and cannot be accurately predicted at this time. However, based on operating experience with

steam generating HTGRs and analysis of projected gas turbine and process heat HTGRs, the following broad ranges might be assumed: 10^2 to 10^3 μatm H_2 , <1 to 10^2 μatm H_2O , 10^1 to 10^2 μatm CH_4 , 10^1 to 10^2 μatm CO , 10^1 to 10^2 μatm N_2 , 10^{-1} to 1 μatm CO_2 . The highest values of oxidants shown ($\text{H}_2\text{O} + \text{CO} + \text{CO}_2$) are equivalent to a steady state water supply from steam generator leaks of about 0.1 lb/h (0.013 g/s). Based on the equilibrium between H_2 , O_2 , and H_2O in the possible operating temperature range, the pressure of oxygen could be 10^{-24} to 10^{-32} μatm at 400°C to 10^{-7} to 10^{-16} μatm at 1000°C . However, one of the tritium removal schemes being considered is to maintain a sufficient oxygen partial pressure in the coolant (i.e., ~ 10 μatm O_2 at 650°C) to form water from all hydrogen isotopes entering the coolant. This scheme would significantly change the oxidation and carburization potential of the coolant as discussed below.

4.2.2.2.2.1.2 Interaction of Helium Coolant with System Components.

The classes of materials being considered for fusion reactor blankets and power conversion components include austenitic stainless steels, austenitic nickel-base alloys and refractory-metal alloys of Mo, V, and Nb. The first two classes of materials depend on the formation of stable protective oxide films to prevent high temperature internal corrosion. The refractory metals do not form protective oxides and are susceptible to interstitial contamination and oxide sublimation at very low partial pressures of active impurities.

Considerable data have been accumulated during the last few years concerning the interaction of helium impurities typical of current and advanced gas-cooled fission reactors with austenitic steels and nickel-base alloys. The data indicate that at temperatures up to about 700°C , the oxide scale effectively prevents internal oxidation or carburization of austenitic stainless steels (i.e., Type 316) and nickel-base alloys at oxygen potentials as low as 10^{-20} μatm . At higher temperatures (up to 1000°C) at the same oxygen potentials, many of these alloys exhibit increasingly nonprotective scales, with the resultant subscale (internal) oxidation of reactive elements such as Al, Ti, Mn, and Si, and in-depth carburization. These effects could produce substantial reduction of tensile, creep, and fatigue properties. Specific alloying elements such as Ti in conjunction with Al_2O_3 scales and perhaps Si in conjunction with Cr_2O_3 scales appear to enhance the protective nature of the scale even at these high temperatures (up to 900°C) and low oxidation (high carburization) potentials. Several nickel-base superalloys have performed satisfactorily up to 12,000 h.

The implications from the current data with respect to austenitic steels or nickel alloys for fusion reactor blankets or power conversion components, are that there will not be compatibility problems with helium coolant impurities at any oxygen potential below 700°C operating temperatures. If these alloys are used in the 700 to 1000°C range (which is doubtful), then reaction with coolant impurities could affect structural integrity at low oxygen potentials (below about 10^{-12} μatm) and additional data, and perhaps alloy modification will be required.

Much less data are available on the reactions of Mo, Nb, and V alloys with impurities in helium. It is known that active gases in inert gas environments can result in interstitial contamination (O, N, and H) and embrittlement, degassing, decarburization, and sublimation of the alloys as volatile oxides. These reactions can occur from 600°C at pressures above 10^{-7} μatm without the formation of a surface scale.

Vanadium produces an oxide V_2O_5 which melts at 670°C, and Mo produces an oxide MoO_3 which melts at 795°C. These temperatures are well within the expected working range of these materials in fusion reactor blankets or power conversion loop. MoO_3 will form at an oxygen partial pressure as low as 2.5×10^{-10} μatm at 795°C and V_2O_5 will form at 2.3×10^{-20} μatm oxygen at 670°C. Thus, continuous melting and vaporization of these oxides in the dynamic (~ 150 – 300 meters/s surface velocity) helium system might result, unless extremely low oxygen partial pressures ($< 10^{-21}$ μatm) are maintained. Protective coatings would probably be required at higher oxidizing potentials.

Recent data on MoTZM in a helium environment containing 50 μatm CH_4 and a $\text{H}_2/\text{H}_2\text{O}$ ratio of 10^3 indicate the formation of an embrittling Mo_3C coating after 3000 h at 800 to 1000°C. The compatibility of candidate refractory alloys even in helium with extremely low oxygen potentials must be determined experimentally in representative simulated impure helium environments. It is also clear that schemes to remove tritium from the coolant by oxidizing all hydrogen species to H_2O (i.e., 10 μatm O_2) might not be feasible with refractory alloy blankets.

4.2.2.2.2 Tritium. The helium coolant can also be used for tritium processing. Methods for separation of tritium from helium by oxidation to T_2O have been fairly well developed. The major concern related to this process involves increased compatibility problems with Path C type structural alloys caused by the higher oxygen pressures in the helium. At the oxygen levels anticipated, this does not appear to be critical for the Path A and B alloys. It appears that tritium concentrations in a helium coolant can be maintained at acceptably low levels; however, as discussed in the chapter on Breeding Materials, satisfactory rates of tritium transport from the breeding materials to helium have not been demonstrated.

4.2.2.2.3 Neutronics. Because of its relatively low density, helium requires rather large coolant channels for adequate heat removal. High pressures are used to minimize this difficulty. Even so, the lower effective blanket densities associated with the large channels required for a helium coolant increase the magnet shielding problems. The most important aspect of this problem relates to neutron streaming down radial supply manifolds and headers. This problem is further enhanced by the fact that helium is neutronically inert. Most other coolants provide a higher degree of neutron stopping power. A favorable aspect of the neutronic properties of helium is the fact that it is not a source of significant radioactivity.

4.2.2.2.2.4 Recirculating Characteristics. A major concern regarding the use of helium as a coolant relates to the high pumping power requirements. Estimates of 3 to 7% of the reactor thermal power have been predicted for the pumping power in helium-cooled commercial fusion reactors. These values are substantially higher than those predicted for other coolants and constitute a significant burden on overall power performance.

4.2.2.2.3 Molten Salts. Molten salts as a class afford relatively high boiling points and, hence, can be contained at relatively low static pressures. Their electrical resistivities are higher than those of liquid-metal coolants, so that they can be moved across strong magnetic fields without incurring the strong braking action induced in liquid metals. However, those molten salt systems which are known to provide acceptable chemical and radiation stability have characteristically high melting points (more than or equal to 300°C). Also, the electrical potential induced by the movement of molten salts through magnetic fields poses an important unknown in assessing their chemical stability. Table 4.2.2.2 is a listing of molten-salt systems which are under consideration as potential coolants for fusion reactors.

4.2.2.2.3.1 Compatibility. In electrolytes flowing across strong magnetic fields, the Lorentz forces acting on the ionic components cause charge separation. The effect is to induce an emf in the system which will alter both the thermodynamic driving force and kinetics of chemical reactions occurring between the salt and its container wall. The magnitude of the emf will depend on the ionic character of the salts, physical dimensions of piping cross sections and on flow velocities. Despite the expected importance of magnetic fields on the chemical compatibility of molten salt coolants, the effect cannot be quantitatively assessed without more definitive reactor designs and electrochemical data on salt-metal systems.

2LiF-BeF₂ (FLIBE). Section 4.2.2.3 discusses the compatibility of LiF-BeF₂ under conditions where the salt is used as a blanket or as blanket and coolant combined. The discussion under 4.2.2.2.3 is also relevant to coolant applications except that transmutation effects become less important. Without the latter effect, the redox potential of the salt can be held to extremely reducing conditions, as required for container materials with relatively high concentrations of reactive alloying additions, such as chromium, titanium, and aluminum.

Intermixing of BeF₂ with lithium produces a mildly exothermic reaction which will precipitate beryllium, thereby raising the melting temperature of the coolant. This will result in a postleak clean-up problem, but would not present a safety problem.

LiF-BeF₂ does not wet or chemically react with graphite. Reaction of the salt with steam or air produces accelerated corrosion but does not constitute a safety problem.

Table 4.2.2.2. Potential molten-salt coolants

Type of salt	Melting temperature (°C)	Advantages	Disadvantages
$\text{KNO}_3\text{-NaNO}_3\text{-NaNO}_2$	140	Low melting point; inexpensive; compatible w/ss	Radiolytic decomposition; reaction w/Li
NaOH-KOH-LiOH	160	Low melting point; inexpensive; radiation stability	Corrosive; reaction w/Li
Carbonates	400	Inexpensive; compatible w/ss	Radiolytic decomposition; reaction w/Li; high melting point
LiF-BeF_2	460	Compatible w/ss; radiation stability; limited reaction w/Li	High melting point; limited beryllium availability
Chlorides	100-450	Low melting point; radiation stability; limited reaction w/Li; possibility of nonionic bonding	Corrosive

KNO₃-NaNO₃-NaNO₂-(HTS). Nitrate-based fused-salt mixtures have been widely used as heat transport fluids and for metallurgical heat treating. One of the better known mixtures (Heat Transfer Salt or HTS) is composed of 40% NaNO₂, 7% NaNO₃, and 53% KNO₃. Although detailed corrosion data are not available, the salt has been suitably contained within its useful temperature range (143-540°C) by both nickel- and iron-based alloys with corrosion rates less than or equal to 4.8×10^{-5} $\mu\text{m/s}$ (1 mpy). Since the oxygen and nitrogen activities of this salt are relatively high, it is unlikely that it would be compatible with reactive metals such as niobium or titanium.

When lithium and HTS are mixed, vigorous reactions are expected in which the nitrate-nitrites would be reduced to Li₂O and Li₃N, the latter being extremely corrosive. The reaction rate and the exothermic energy released would be significant from a safety standpoint. NaNO₃ is also extremely reactive toward carbon, and it is likely that intermixing of HTS with graphite would pose a safety hazard.

NaOH-KOH-LiOH. The most serious limitation of the hydroxides as coolants is their corrosivity. No satisfactory container material has been found for containing hydroxides for a reasonable length of time above about 600°C. Below 600°C, alloys based on nickel and molybdenum appear to withstand molten caustic, but only a few experiments to verify this have been run. Corrosion in hydroxide melts occurs by direct oxidation of the container metal to an oxysalt accompanied by the reduction of hydroxyl ions to form hydrogen and oxide ions. Complicating side reactions can include the formation of a metal hydride, the reduction of alkali metal ions, and the production of water. The corrosion of nickel by fused sodium hydroxide can be inhibited at 800°C by a 1 atm overpressure of hydrogen. Leakage of steam into hydroxides is tolerable from a safety standpoint but would produce an environment conducive to stress corrosion cracking.

Li₂CO₃-Na₂CO₃-K₂CO₃. Alkali metal carbonates are being evaluated as molten electrolytes for fuel-cell applications. Corrosion of metals in carbonate mixtures occurs by direct oxidation, but compatibility data do not presently exist which can be applied to forced convection systems.

Chlorides. Molten chlorides are generally similar to fluorides in their corrosive attack on metals. Thus, the discussion in 4.2.2.2.3 is directly relevant. The lower melting chloride systems, based on ZnCl₂ or AlCl₃, have relatively low binding energies and, hence, should be relatively corrosive. Although corrosion data do not exist, it is doubtful that these lower melting chlorides would be compatible with Path A or B alloys. Acceptable container materials would probably be limited to graphite, nickel, and molybdenum.

4.2.2.2.3.2 Purity Control. The corrosion properties of most molten-salt systems, particularly fluorides and chlorides, are strongly affected by trace impurities in the melt. This requires that the salts be chemically processed to remove undesirable impurities prior to

reactor service. Such methods are presently available for both chloride and fluoride systems, but requirements and procedures for other salt systems have yet to be determined. Those salts that undergo significant transmutation reactions will also require some form of purification or make-up during reactor operation.

4.2.2.2.3.3 Thermal, Radiation, and Chemical Stability of Coolant. LiF-BeF₂ mixtures and molten hydroxides are stable under gamma and neutron radiation at elevated temperatures but are subject to decomposition in the solid state. The behavior of molten nitrate-nitrite salts under irradiation has not been experimentally determined; however, thermal decomposition of HTS becomes significant above about 500°C, and it is likely that radiation would promote decomposition at still lower temperatures. Since carbonates are thermodynamically quite stable, it is probable that they would resist radiolytic decomposition. Also, those chloride salts with relatively high binding energies should afford good stability under irradiation. As discussed in Section 4.2.2.2.3.1, the chemical stability of molten salt mixtures will be strongly affected by the rate at which the salts travel across magnetic fields. All of the above salts have relatively low vapor pressures up to 450°C, and, in the case of LiF-BeF₂ and hydroxides, to temperatures upwards of 800°C.

4.2.2.2.3.4 Tritium. Oxygen and hydrogen-containing salt systems such as the hydroxides and carbonates will act as a large reservoir or sink for tritium, i.e., the escaping tendencies for tritium, once in the salt would be practically nil. Chlorides and fluorides, on the other hand, afford only a limited reservoir for tritium, and tritium will readily pass from the salt to a surrounding vacuum air, or steam environment. Processing of tritium would require a heavy expenditure of energy in the case of hydroxides and carbonates; much less energy in the case of fluorides or chlorides.

4.2.2.2.3.5 Neutronics. The neutronic properties of molten salts, excepting LiF-BeF₂ mixtures, have not yet been assessed under fusion-reactor conditions. Tritium generation and parasitic capture by fluorine are the principal neutronic factors affecting the use of LiF-BeF₂ as a coolant.

4.2.2.2.3.6 Thermal Properties. Properties relating to the heat transfer capabilities of selected salt systems are summarized in Table 4.2.2.3.

4.2.2.2.3.7 Pumping Requirements. Because of inherently low thermal conductivities, molten salts must develop turbulent flow to cool efficiently. Calculations of the Hartman number for Li₂BeF₄ indicate that the imposition of a strong magnetic field will not increase the laminar-turbulent transition Reynolds number above that normally specified. This implies that molten Li₂BeF₄, and molten salt coolants in general, could be pumped through magnetic fields using conventional flow circuitry and pump designs. Of greater concern is the electrical potential induced in flowing salt mixtures by magnetic fields, which

Table 4.2.2.3. Thermal properties of molten salts

Salt	Thermal conductivity (watt. m ⁻¹ K ⁻¹)	Heat capacity (JK _g ⁻¹ K ⁻¹)
2LiF-BeF ₂	1.0	2385
KNO ₃ -NaNO ₃ -NaNO ₂	0.61	1560
LiOH-NaOH-KOH	1.05	2123
Chloride	0.84	1261
LiCO ₃ -Na ₂ CO ₃ -K ₂ CO ₃	0.35	1728

may place an upper limit on flow velocity to avoid undue corrosion (Section 4.2.2.2.3.1).

4.2.2.2.4 Water. Water in some form has been used as a coolant for many engineering applications. For the purposes of this assessment, both pressurized water and steam are considered in this section. Because of its extensive technology base and favorable heat-transfer properties, water has been proposed as a coolant in near-term fusion devices, e.g., the experimental power reactor. However, serious questions arise as to the viability of water (pressurized or steam) as a coolant in tritium-breeding commercial power reactors. The two major problems relate to compatibility with the breeding material and poor tritium processing characteristics. Besides the broad technology base, water possesses favorable neutronics properties and has low pumping power requirement.

4.2.2.2.4.1 Compatibility and Purity Control. An extensive technology base on the compatibility of pressurized water and steam with commercial structural alloys has been developed for the fission reactor programs. The compatibility limitations for structural materials such as austenitic stainless steels, ferritic steels, aluminum alloys, certain nickel-base alloys, and Zircaloy have been well documented. The compatibility of these alloys is acceptable for fusion reactor applications. However, most of the Path C alloys, e.g., vanadium and niobium alloys, cannot be used with water coolant under conditions of interest.

It has been well established that water chemistry is an important factor relative to the compatibility of potential structural materials. For example, hydrogen overpressures are typically used in light-water fission reactors to maintain the oxygen potential at low levels in pressurized water systems. Because of the importance of impurities in the plasma and the high permeation rates of hydrogen, it is questionable whether this procedure is appropriate for fusion reactor applications.

The compatibility of water with candidate breeding materials is a major concern. Lithium, in particular, is known to react vigorously with water. Although a precedent for an alkali metal-water combination has been set in the LMFBR steam generator, one would probably not use this combination in a high radiation environment, particularly where

space is a premium. Water is less reactive with the other candidate breeding materials; however, much less data are available on reaction kinetics.

4.2.2.2.4.2 Thermal, Chemical and Radiation Properties. The heat-transfer properties of both water and steam are well documented. Water in both phases is an excellent heat-transfer fluid for many applications. The critical point for water is 374°C at 218 atm. The radiation properties of water present no special problems. In fact, water in the liquid phase serves as a good moderator.

4.2.2.2.4.3 Tritium. Tritium processing is a major disadvantage of water coolant. It is extremely difficult and expensive to remove tritium from water at low concentrations. Since water is an excellent sink for tritium at low pressures, substantial amounts of tritium will end up in the coolant. Although this problem does not appear to be prohibitive for small experimental devices, it is questionable whether water is acceptable as a coolant in commercial power reactors for this reason.

4.2.2.2.4.4 Recirculating Characteristics. The pumping or recirculating power requirements for a pressurized water or steam coolant are very favorable. Less than 1% of the reactor thermal power is needed for this purpose. Techniques and equipment for recirculating water have also been well developed.

4.2.2.3 Summary of Critical Problem Areas

Establishing the viability of the various coolant concepts should be a major objective of the fusion materials program, since the coolant must interface closely with both breeding and structural materials. Identification of potentially suitable blanket/structure/coolant combinations is an important factor in assessing the priorities of all materials development programs. From the preceding assessment, the breeding and structural materials combinations that are considered feasible with the most viable coolants are given in Table 4.2.2.4. A summary follows of the critical areas where additional research and development are required to demonstrate the suitability of these concepts.

Table 4.2.2.4. Most viable coolant/structural breeding materials combinations

Coolant	Breeding material	Structure
Liquid Lithium	Liquid Lithium	Path C (reactive/refractory metal alloys)
Sodium	Liquid Lithium	Path A (stainless steel)
Helium	Solid Lithium Compound	Path A (stainless steel) or Path B (Fe-Ni-Cr alloys)

4.2.2.3.1 Liquid Lithium. Based on the preceding assessment, three key areas are identified where additional analytical and experimental information is required before a final assessment can be made on the feasibility of a liquid lithium coolant.

4.2.2.3.1.1 Electromagnetic Effects. Information regarding the behavior of a highly conductive recirculating liquid metal in a strong magnetic field is essential. What are the pumping power requirements? How important are induced hydraulic pressures? What are the magnetic field effects on heat and mass transfer?

4.2.2.3.1.2 Compatibility with Structure. Compatibility limitations are important in establishing the temperature limitations and viability of blanket concepts that utilize liquid lithium with Path A or B structural alloys. Determination of the rate-controlling mechanisms and temperature limitations should be a major factor influencing alloy development programs. Lithium purity requirements and purification methods required for lithium-Path C alloy concepts should be established, particularly with respect to dissimilar metal mass transfer.

4.2.2.3.1.3 Tritium Processing. Capabilities of proposed tritium processing methods for liquid lithium must be established. The acceptability of these processes will influence other materials considerations and operating parameters.

4.2.2.3.2 Sodium. The technology required for the use of sodium as a coolant is relatively well developed. However, some development work is required to better assess the tradeoffs between sodium and the other coolants. These areas, which relate directly to the fusion reactor application, involve electromagnetic effects, compatibility, and tritium processing.

4.2.2.3.2.1 Electromagnetic Effects. This problem, which is similar to that described for lithium, relates to the behavior of a highly conducting fluid in a strong magnetic field. What are the pumping power requirements? How important are induced hydraulic pressures? What are the magnetic field effects on heat and mass transfer?

4.2.2.3.2.2 Compatibility with Structure. Sodium coolant is considered most viable with liquid lithium breeding materials. Problems of interfacing these two materials with a common structural material arise at high temperatures. As indicated in the previous assessment, the primary incentive for utilization of a sodium coolant relates to the possibility of using conventional Path A and B alloys for containment at higher temperatures. The compatibility problems of most concern involve dissimilar-metal mass transfer caused by the use of different containment materials internal and external of the reactor.

4.2.2.3.2.3 Tritium. Ease of tritium processing may be a significant advantage of sodium over a lithium coolant. However, further work is required to demonstrate the adequacy of tritium removal from sodium by tritide precipitation processes.

4.2.2.3.3 Helium. A substantial technology base relevant to the use of helium as a heat transfer medium has been developed in gas-cooled fission reactor programs. As a result, the favorable properties and materials limitations have been fairly well established and the major problem areas have been identified.

4.2.2.3.3.1 Compatibility. The compatibility of helium with Path A and B alloys appears acceptable at least to temperatures of 700°C, which is probably above that allowable from mechanical properties or radiation damage limitations. However, it must be demonstrated that acceptable leakage rates of helium can be achieved in large, high-pressure piping systems of the type being proposed. It does not appear that helium with anticipated impurity concentrations can be used with the Path C alloys at temperatures of interest because of compatibility limitations. The problem essentially involves a correlation of acceptable helium impurity levels with levels that can be achieved in large recirculating systems. The aspects of in-leakage from the atmosphere or steam generator during off-normal conditions must be considered. As pointed out in the assessment, the process of tritium removal from helium by the oxidation process probably cannot be used with Path C structural alloys under conditions of interest. Development of coatings, which might alleviate the compatibility problems associated with the Path C alloys, would require considerable effort.

4.2.2.3.3.2 Neutronics. Shielding of the magnets in reactors that utilize helium as a coolant will be more difficult than for other coolants because of the larger coolant channel size requirements and the fact that helium is relatively inert to neutrons. Determination of the magnitude of this problem is important to the assessment of the viability of helium as a coolant.

4.2.2.3.3.3 Recirculating Characteristics. The problem of relatively high pumping power requirements for helium should be examined. Methods for improving pumping efficiencies would have important consequences.

4.3 MATERIALS FOR TRITIUM CONTROL

4.3.1 Questions on Materials for Tritium Control

- a. Are barrier materials required? Where?
- b. How would one fabricate and test barrier materials?
- c. Can the problem be avoided by selection of operating temperature, structure, coolant, or system?
- d. Does He production from tritium decay affect the barrier material or interface?
- e. Are tritium windows required or desired? Under what conditions would they operate? Would dissimilar-metal mass transfer destroy their effectiveness?
- f. What other materials problems are envisioned?

4.3.2 Responses to Questions Concerning Tritium Permeation Barriers

4.3.2.1 Permeation Barriers

There was a general consensus of opinion among the subtask group members that tritium permeation barriers would be needed in fusion reactor systems (particularly in elevated temperature environments) and that their development should be a high priority item. The tritium migration problem in heat transfer and energy conversion systems is well recognized, but the same problem in other reactor subsystems may be equally as serious and is, we find, in need of detailed analysis. The recommended approach to a better understanding of tritium migration and a definitive verification of barrier performance is to conduct studies with tritium under realistic reactor conditions. The need was expressed for a more expanded permeation data base including effects of bulk and surface impurities, influence of alloying elements, and studies at low driving pressures (i.e., $\leq 10^{-6}$ torr) using tritium. Also, the effects of neutron and photon damage, energetic particle injection, and helium formation (from transmutation and tritium decay) on permeation rates through metals and on the integrity of permeation barriers is in need of study. These effects are expected to be most acute in the vicinity of the first wall and blanket, but some of them may influence materials performance in outlying regions as well. Several other points raised by members of the subtask group are listed below:

a. The most practical approach to developing permeation barriers seems to be utilization of in situ forming, self-healing oxide (or other impurity) coatings that develop on the surfaces of many conventional structural alloys.* If the effectiveness of these "natural" coatings proves to be degraded by thermal stress, vigorous fluid environments, or aging, then the implementation of (i) applied surface coatings and/or alloying elements and (ii) composite materials (for special high temperature applications) should be pursued as alternative means of increasing barrier layer resiliency.

b. The magnitude of barrier protection beyond which other factors (e.g., leakage and grain boundary effects) take over as the dominant tritium migration mechanism needs to be determined. It would be fruitless to over-develop permeation barriers if permeation does not turn out to be the only significant contributor to overall tritium losses - as is presently assumed by many in the field.

c. The ultimate utility of high permeability materials as "tritium windows" was discussed briefly. While the possibility exists that they may find application in blanket processing, the method appears to be fraught with difficulties associated with dissimilar metal mass transfer, high temperature operation, and deteriorating performance with time.

*Providing of course that useful materials with low intrinsic bulk permeabilities cannot be identified or developed.

However, a small scale effort to add more perspective in this area would not be considered imprudent.

4.3.2.2 Tritium Interactions with Organic Materials

Although the primary tritium containment systems for fusion reactors will undoubtedly be constructed almost exclusively of metals, organic materials are likely to be used in many applications throughout the secondary containment systems (e.g., pump oils, seals, lubricants, gloves, gaskets, etc.). Radiation damage to pump oils and seals (caused primarily by the tritium that gets impregnated into them) is recognized as a serious problem. There is considerable experience in dealing with this problem, but test data are available only for a limited variety of organic materials and the amount of R and D currently ongoing is minimal. Since this problem has received less attention in the fusion program than the one associated with tritium permeation, the need exists for a comprehensive assessment of the potential for and consequences of tritium interactions with organic materials in fusion reactor systems. Hopefully, out of this activity would come an assemblage of existing data that could be factored into many of the near-term device design activities (e.g., TFTR, TSTA, TNS) and, more importantly, a firmer definition of the areas where R and D is most needed.

4.3.2.3 Materials Selection, Fabrication Methodology, and Quality Assurance

There is a considerable amount of experience and associated documentation in this area (e.g., at LLL, LASL, SRL, MLM, etc.). (LLL, for example, has established its own certified materials store for constructing tritium handling systems.) What would be most helpful here in the near-term is a comprehensive assessment of the present status of (a) materials procurement and certification practices; (b) fabrication standards, (c) quality assurance methods, and (d) hardware selection criteria. This assessment should be carried out in the light of presently conceived requirements for tritium handling in the fusion devices of the 1980s. Out of this activity could come (a) a handbook or guidance-type document that would be extremely useful to ongoing design studies and (b) in all likelihood, a much clearer definition of areas where important fusion-related R and D needs exist.

Several other points raised during discussions of this topic are worthy of note.

a. There appear to be two basic philosophies at the major tritium handling laboratories regarding secondary containment practices -- one advocating high quality, essentially fail-safe primary containment and the other advocating extensive use of secondary containment (double jacketing, local enclosures, processible atmospheres, etc.). The relative merits of each philosophy from the standpoint of system effectiveness and cost-benefit may have to be evaluated on a subsystem by subsystem basis.

b. There would seem to be at least two incentives for constructing primary containment systems so that migration of tritium to the secondary containment is minimal. High levels of tritium release to secondary containment on a routine basis would surely beg concern and criticism over the consequences of secondary containment system failure. And, high tritium levels in the secondary systems may exacerbate maintenance problems beyond practical limits.

c. The question of leak detection and prevention was discussed. This question is clearly difficult to scope because it is highly design dependent. In the long run, the reduction and control of leaks will depend largely on the effectiveness of the quality assurance program.

d. Some interest was expressed in the development of coatings, paints, and other surface preparations that would resist tritium absorption and impregnation. These might be particularly useful in guarding large surface areas (e.g., like the walls of the reactor hall itself) against massive tritium soaking following any sizable tritium release.

e. There is much need for a decision on the part of DMFE concerning the extent to which materials performance problems associated with clean-up system operation will be covered in the Materials and Radiation Effects Branch programs.

4.3.2.4 Facilities

Only a limited amount of discussion was generated with regard to facilities for conducting tritium/materials research. The key points of these discussions are summarized below:

a. A wealth of expertise exists at the major tritium handling laboratories that can and should be called upon.

b. The TSTA will undoubtedly serve as a primary test bed for development and qualification of materials for tritium service.

c. Several subtask group members recommended the establishment of a permeation screening facility capable of studying large numbers of samples simultaneously. At the present level of capability within DMFE programs, only a small fraction of the required sampling program will get carried out in this decade.

4.4 ASSESSMENT OF GRAPHITE, BORONATED GRAPHITE, AND SILICON CARBIDE FOR FUSION REACTOR BLANKETS

4.4.1 The following questions have been posed:

- a. Where has its use been proposed?
- b. What are the proposed operating conditions?
 - Temperature
 - Stress levels (steady state and transient)

- Neutron flux and fluence
 - Coolant (co-patibility)
 - Container (compatibility)
- c. What are the performance limits for graphite?
- Temperature
 - Stress levels
 - Neutron flux and fluence (helium content, hydrogen content significance)
 - Coolants
 - Containers
- d. If not, what are the advantages and disadvantages to its use?
- e. What development program is required to assure the satisfactory performance of graphite for the proposed applications?
- f. What are your recommendations?
- g. Where do you do your testing and how long does it take?

4.4.2 The following answers are supplied:

4.4.2.1 Usage and Operating Conditions

The potential usage of graphite, B_4C , and SiC in fusion system blanket designs is summarized in Table 4.4.1 as it was visualized in early 1976. It will be noted that the materials requirements cover the entire gamut of possible fluences and temperatures. At this relatively early time, the acute need is to perform sufficient screening experiments to delineate performance limitations and, thus, potential ranges of useful application. Development of engineering data bases should await more definitive blanket designs, utilizing these materials within their feasibility range.

4.4.2.2 Material Limitations

4.4.2.2.1 Graphite Limitations. Currently known limitations on the use of graphite are summarized in Figure 4.4.1 and Table 4.4.2. Figure 4.4.1 plots the lifetime of several graphites as a function of irradiation temperature in terms of first-wall days, or in terms of total fluence for other applications. During damage, typical graphites tend to shrink, to the extent of 2-3% in linear dimension for isotropic materials. The graphite then tends to swell until it reaches and exceeds its original dimensions, accompanied by increasing mechanical deterioration. Lifetime is defined by the time (fluence) for the material to return to its original dimensions, at which point mechanical degradation is not yet severe. NC-8 is an anisotropic "typical" nuclear graphite. Grade H-451, discussed below, represents an isotropic material with somewhat improved lifetimes and mechanical properties.

Table 4.4.1. Assessment of proposed uses of graphites in fusion reactor blankets
as of March, 1976

Device	Material	Use	Conditions	Potential problems
ANL Tokamak EPR	Graphite B ₄ C SiC	First wall	Environment: adjacent to plasma Temperature: ~465°C Stress: NS* Fluence: 2.7×10^{21} n/cm ² ·yr	a, b, c, d, e, f.3, h
	B ₄ C	Inner shield	Environment: in ss Temperature: ~20°C Stress: mild Fluence: mild	d, h
	B ₄ C	Outer shield	Environment: in graphite with contact to ss Temperature: ~20°C Stress: mild Fluence: mild	d, h
BNL graphite blanket designs (EPR design)	Graphite	First wall/ moderator	Environment: adjacent to plasma, contacts breeder (e.g. LiAlO ₂) Temperature: ~2200°C max. Stress: 500 psi thermal Fluence: 2.4×10^{21} n/cm ² max.	a, b, d, e, f.2, g
	Pyrolytic graphite or fibrous graphite	Low thermal cond. layer facing plasma	Environment: adjacent to plasma Temperature: 2000°C Stress: NS Fluence: 2.4×10^{21} n/cm ²	a, b, f.1, g

Table 4.4.1 (continued)

Device	Material	Use	Conditions	Potential problems
BNL minimum activity aluminum blanket designs	SiC	Coolant tubes	Environment: adjacent to plasma Temperature: NS Stress: NS Fluence: NS	a, b, c, d, f.3
	B ₄ C	Shield	Not Specified	
	Graphite	Filler pieces between Al canister	Environment: adjacent to plasma Temperature: NS Stress: NS Fluence: 2.7×10^{21} n/cm ² ·y (EPR design) 1.3×10^{22} n/cm ² ·y (commercial design)	a, b, c, d, e, f.3 (commercial design only)
	Graphite fiber (e.g., cloth, felt)	Thermal insulation inside Al canister	Environment: contacts He and Al or SAP Temperature: NS Stress: NS Fluence: 2.7×10^{21} n/cm ² ·y (EPR design) 1.3×10^{22} n/cm ² ·y (commercial design)	e, f.1, g
	SiC B ₄ C (design only)	Moderator	Environment: contacts He, Al, and breeder material (e.g. L:AlO ₂ , Li ₇ Pb ₂ , etc.) Temperature: 1450°C Stress: NS Fluence: NS	e, f.2, h

Table 4.4.1 (continued)

Device	Material	Use	Conditions	Potential problems
	B ₄ C	Shield	Not specified	
GA EPR	SiC	First wall	Environment: adjacent to plasma Temperature: 1350°C Stress: 600 psi Fluence: 2×10^{22} n/cm ² ·yr	b, c, d, e, f.3
	Graphite	Blanket filler	Environment: vacuum and ss support Temperature: <1650°C Stress: NS Fluence: 4×10^{22}	a, c, d, e, f.2, f.3, g
	B ₄ C	Shield	Environment: NS Temperature: 165° Stress: mild Fluence: 10^{13} n/cm ² ·sec	h
GA noncircular Tokamak (DPR)	SiC	First wall	Environment: adjacent to plasma Temperature: 1600°C Stress: NS Fluence: 5×10^{22} n/cm ² (4y)	b, c, d, e, f.3
	Graphite	Moderator	Environment: Li breeder salt and inconel. Temperature: ~700°C Stress: NS Fluence: 6×10^{21} n/cm ² (4y)	d, e

Table 4.4.1 (continued)

Device	Material	Use	Conditions	Potential problems
	B ₄ C	Shield	Environment: Contact with ss and/or graphite Temperature: NS Stress: NS Fluence: mild	e, h
LASL RTPR	Graphite	Moderator	Environment: canned in Nb-1Zr Temperature: 900-975°C Stress: NS Fluence: 10 ²² n/cm ² ·y	e, f.3
Mirror blankets	Graphites SiC	Moderator	Not specified	
MIT BRSR blanket	Graphite	First wall shield and moderator	Environment: adjacent to plasma contacts Nb-1Zr Temperature: 850°C max. Stress: NS Fluence: 1.4 × 10 ²¹ n/cm ² max.	a, b, c, d, e, f.3, h
ORNL EPR	Graphite	Blanket reflector	Environment: canned in ss Temperature: 550°C Stress: NS Fluence: NS	e
UWMAK I	B ₄ C	Shield	Environment: contact with ss Temperature: NS Stress: NS Fluence: <10 ²¹ n/cm ² (15y)	e, h

Table 4.4.1 (continued)

Device	Material	Use	Conditions	Potential problems
UWMAK II	Fibrous graphite	First wall (curtain)	Environment: adjacent to plasma Temperature: 1100°C Stress: NS Fluence: 2.5×10^{22} n/cm ² (2y)	a, b, f.1, f.3
	Graphite	Reflector	Environment: canned in ss Temperature: 660°C Stress: NS Fluence: 1.9×10^{22} n/cm ² (15y)	e
	B ₄ C	Shield	Environment: contact with ss Temperature: NS Stress: NS Fluence: mild	e, h
UWMAK III	Graphite	ISSEC (partial)	Environment: adjacent to plasma Temperature: 1900°C Stress: NS Fluence: $\sim 5 \times 10^{22}$ n/cm ² (3y)	a, b, c, d, f.2, g
	Fibrous graphite	First wall (curtain)	See UWMAK II	
	Graphite	Moderator	Not specified	
	B ₄ C	Shield	Environment: NS Temperature: <200°C Stress: NS Fluence: mild	

*Not specified.

Table 4.4.1 (continued)

Potential Problems

- a. Degassing
- b. Interactions with plasma (e.g., physical and chemical sputtering)
- c. Thermal stress
- d. Irradiation induced stress
- e. Chemical compatibility with materials in direct contact
- f. Irradiation induced dimensional changes
 - f.1. Materials for which there are little or no data
 - f.2. Some data exist but not at temperatures of interest (e.g., above $\sim 1400^{\circ}\text{C}$ for graphite)
 - f.3. Data exist at temperatures of interest but lifetimes may be exceeded unless more radiation resistant materials can be developed.
- g. Irradiation induced changes in thermal conductivity
- h. He generation in B_4C loss of integrity

Figure 4.4.1. Minimum expected lifetime of graphite in first-wall fusion applications at 1 MW/m² wall loading.

Table 4.4.2. Factors potentially limiting the performance of graphite materials for fusion applications

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
H-451 graphite	Temperature	2100°C	2100°C	Carbon vaporization (vapor pressure exceeds 0.1 Pa)
	Stress	15 MPa (tension)	25 MPa (tension)	Mechanical failure
	Fatigue	9 MPa	Not known	Mechanical failure
	Max fast-neutron fluence at 1000°C	—	10^{22} n/cm ²	Swelling
	Thermal-stress parameter	3×10^4 W/m	1.5×10^4 W/m	Thermal stress failure
	Compatibility with TZM	~900°C		Formation of Mo ₂ C layer
	Compatibility with stainless steel	700°C		Carburization of stainless steel
	Compatibility with lithium	<300°C (Depends on purity of liquid metal)		Swelling, carbon transport
	Compatibility with sodium	~400°C (Depends on purity of liquid metal)		Swelling, carbon transport
	Compatibility with molten salt	Decomposition temperature of salt		Swelling, carbon transport

In contradistinction, the advanced graphites (such as Poco AXF) show no shrinkage at all, and tend to go into rapid swelling and mechanical deterioration at 50-100% longer lifetimes than the so-called typical graphites. The Poco material, in particular, would probably cost ten times as much as typical graphite, and is available in useful but limited size ranges.

At ORNL, experimental graphites were developed in the late 1960s which equal or exceed the Poco graphite. These types of material have since been intensively developed for military applications, but no irradiation data on these latter materials exist. It can be said with confidence that graphites exceeding performance of the Poco material in nuclear or fusion applications can be developed, although at some cost penalty. In view of replacement costs, particularly off-stream time, the net cost should be negative.

Table 4.4.2 presents the known performance limits for H-451 graphite. This table covers only a small sampling of available data; the following comments may be made.

Constitutive Equations: Generally known to full life over the range 600-1200°C in the absence of creep.

Creep: Being determined over same temperature range. Creep now known to affect all mechanical properties.

Thermal Properties: Known over same range. The tabulated quantity "Thermal stress parameter" is calculated from the formula $UTS \cdot \lambda (1 - \mu) / \Sigma \alpha$, where UTS is the ultimate tensile strength, λ the thermal conductivity, μ the Poisson's ratio, Σ the Young's modulus, and α the linear thermal expansion coefficient, all as measured at 1000°C. This formula tends to underestimate the thermal stress resistance of graphitic materials.

Effect of Multiaxial Stress: Not yet defined.

Failure Mechanisms: Under investigation, but not yet defined.

Pyrolytic graphite has been considered as a possible coating material. Attempts to date to produce damage-stable coatings on massive graphite substrates have been unsuccessful due to the necessity to control void structure simultaneously in both coating and substrate. However, this work was never carried beyond the exploratory stage during the Molten-Salt Reactor Program. Table 4.4.3 summarizes known data on pyrolytic carbons primarily derived from coated fuel particles for HTGRs. An area not yet explored is the possibility of alloyed pyrolytic carbon and SiC, a material with both increased mechanical strengths and resistance to neutron damage.

Fibrous graphites have been extensively developed for military applications, with limited commercial markets where the high cost is not limiting. Their extensive use is as composite materials with one-,

Table 4.4.3. Factors potentially limiting the performance of pyrolytic carbon materials for fusion reactor application

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
Isotropic pyrolytic carbon	Temperature	2000°C	2000°C	Graphitization and shrinkage
	Stress	250 MPa (tension)	250 MPa (tension)	Mechanical failure
	Fatigue	> 200 MPa (tension, 10 ⁶ cycles)	Not known	Mechanical failure
	Maximum fast neutron fluence at 1000°C	—	>10 ²² n/cm ²	Swelling
	Thermal stress parameter	10 ⁴ W/m	3 × 10 ³ W/m	Thermal stress failure
	Compatibility with TZM	~900°C		Formation of Mo ₂ C layer
	Compatibility with stainless steel	700°C		Carburization of stainless steel
	Compatibility with lithium	<300°C	(Depends on purity of liquid metal)	Swelling, carbon transport
	Compatibility with sodium	~400°C		
	Compatibility with molten salt	Decomposition temperature of salt		—

two-, and three-dimensional lay-ups in carbon and metallic matrices. The application of composite materials in a significant neutron field appears limited due to very high axial fiber shrinkage. The existing information is given in Table 4.4.4. It is not yet evident that fibrous graphite without a supporting matrix could not be employed as a freestanding curtain, although the mounting problems would be severe.

4.4.2.2.2 Silicon Carbide. Information on silicon carbide is considerably limited compared to graphite, and what is known generally applies to co-vapor-deposited (CVD) material in thin ($<100\text{ }\mu\text{m}$) coatings. Known data are summarized in Table 4.4.5. In contrast to graphite, the following general observations can be made: silicon carbide tends to be less chemically inert in either reducing or oxidizing environment, and is less thermally stable and more subject to thermal shock. Conversely, within its temperature limits, it is essentially dimensionally stable and its lifetime far exceeds that of graphite. Extensive testing of bulk silicon carbide with controlled stoichiometry and grain size is required before its apparent advantages can be said to be established.

4.4.2.2.3 Boronated Graphite. The following discussion is limited to B_4C in graphite. The presence of boron in graphite considerably reduces the lifetime due primarily to recoil from the (n,α) reaction. This can be minimized by keeping the particulates of B_4C large, thus confining much of the recoil energy within the particulate. At temperatures in excess of 2300°C , boron rapidly goes into solution in graphite, reordering the microstructure and strongly enhancing the radiation damage. Boronated graphite has been investigated and used as both a control and shielding material. It is most attractive as a shield material because of its thermal stability, high heat content, acceptable resistance to damage, and low cost. Available data is summarized for two levels of boron content in Tables 4.4.6 and 4.4.7.

4.4.3 Recommended Development and Testing Program

As stated in Section 3.4.2, a feasibility study on graphite and bulk silicon carbide is the only program recommended prior to 1980, or until specific application requirements come into focus. The premises on which this conclusion is drawn are as follows:

- a. At temperatures below 1400°C , the lifetime of graphite generally will be well below that of metals. In applications near the front wall, graphite must be easily replaceable or operated presumably at temperatures well above 1200°C . The radiation behavior in this high-temperature domain must be quickly established to determine feasibility of ISSEC (Internal Spectral Shifter and Energy Converter) or similar design concepts.
- b. Very little is known about bulk SiC , other than it appears to have definite advantages over graphite in terms of lifetime. Again, feasibility needs to be quickly established for front-face applications, with material of highly defined composition and texture.

Table 4.4.4. Factors potentially limiting the performance of fibrous graphite materials for fusion reactor applications

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
Graphite cloth	Temperature	2100°C	2100°C	Vapor pressure exceeds 0.1 Pa
	Stress	700-2500 MPa (tensile strength of fibers)	Unknown	Mechanical failure
	Fatigue	Unknown	Unknown	
	Maximum fast neutron fluence at 1000°C	—	Unknown ^a	Possible loss of integrity or unacceptably large dim. changes
	Thermal stress parameter	Unknown but very high	Unknown	
	Compatibility	Probably same as other carbons and graphites, but tendency if any would be towards more reactive.		

^aCloths irradiated to 1×10^{22} n/cm² @ 470°C shrank 18-27% in axial direction of fibers, yet some types of cloth maintained integrity very well.

Table 4.4.5. Factors potentially limiting the performance of silicon carbide materials for fusion reactor applications

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
CVD silicon carbide	Temperature	1600°C	1600°C	Silicon loss (vapor exceeds 0.1 Pa)
	Stress	400 MPa (tension)	400 MPa (tension)	Mechanical failure
	Fatigue	>300 MPa (tension, 10^6 cycles)	Not known	Mechanical failure
	Maximum fast neutron fluence at 1000°C	—	$> 5 \times 10^{22}$ n/cm ²	Swelling
	Thermal stress parameter	5×10^3 W/m	2×10^3 W/m	Thermal stress failure
	Compatibility with TZM	$\sim 1000^\circ\text{C}$		Chemical interaction (silicide formation)
	Compatibility with stainless steel	650°C		
	Compatibility with lithium	Incompatible		Dissolution and carbon transport
	Compatibility with sodium	Incompatible		
	Compatibility with molten salt	Not known		—

Table 4.4.6. Factors potentially limiting the performance of boronated graphite materials for fusion reactor applications

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
40 wt % B of B ₄ C in graphite	Temperature	2300°C	2300°C	Slump of compact oxidation of B ₄ C serious at 1000°C if high moisture present
	Stress	7 MPa UTS 28 MPa UCS	6 MPa UTS 24 MPa UCS	Fracture
	Fatigue	Unknown	Unknown	
	Maximum fast neutron fluence at 1000°C	—	$> 7 \times 10^{21}$ n/cm ²	Large ($\approx 5\%$) dimensional change possible fracture
	Thermal stress parameter	4×10^3 W/m	1×10^3 W/m	Tensile failure
	Compatibility with TZM	Reaction rate of $R = 3.56 \times 10^9 e^{-34801/T}$ R in mill/year, T in °R		Neutron capture by B ¹⁰ in metal leading to swelling in metal
	Compatibility with stainless steel			
	Compatibility with lithium	Unknown		
	Compatibility with sodium	650°C	650°C	Loss of structural integrity after ~ 1000 hours
	Compatibility with molten salt	Unknown		

Table 4.4.7. Factors potentially limiting the performance of boronated graphite materials for fusion reactor applications

Material	Limiting factor	Probable limit		Consequence of exceeding limit
		Unirradiated	Properties after irradiation to 10^{22} n/cm ² at 1000°C	
5 wt % B of B ₄ C in graphite	Temperature	2300°C	2300°C	Subsequent large irradiation — induced dimensional changes
	Stress	7 MPa UTS 28 MPa UCS	7 MPa UTS 28 MPa UCS	Fracture
	Fatigue	Unknown	Unknown	
	Maximum fast neutron fluence at 1000°C	—	1.5×10^{22}	Large (>5%) dimensional change
	Thermal stress parameter	Unknown	Unknown	
	Compatibility with TZM	Some T dependent reaction but limited by surface contact of B ₄ C with metal		Metal swelling
	Compatibility with stainless steel			
	Compatibility with lithium	Unknown		
	Compatibility with sodium	Limited by graphite reactions		Loss of boron
	Compatibility with molten salt			

- c. Although of lesser criticality, the behavior of fibrous graphite can be easily established as a potential curtain material by "piggyback" experiments with bulk graphite.
- d. The probability of significantly increasing graphite lifetimes with acceptable cost penalties for the material appear large. The lead time on such developments will be long; but the incremental program costs, small.

On this basis, the following program is recommended.

The only practical source of a fast-neutron spectrum to determine displacement damage is the HFIR target facility. Disadvantages of this facility are the inability to instrument capsules; hence, its utility is primarily for feasibility studies rather than engineering data. Fluence can be generated at the rate of 3×10^{22} peak per reactor year, and four 3-capsule sets are recommended on the following schedules:

Temperature:	1400°C	1600°C	1800°C	1500°C
Material:	← Graphites →			SiC
Cumulative Fluences:	1)	← 7 × 10 ²¹ →		
	2)	← 1.5 × 10 ²² →		
	3)	← 3 × 10 ²² →		
Properties:	← Dimensions →			
	← Thermal Expansion →			
	← Moduli →			

Properties would be measured at the termination of each irradiation; such measurements are essential, at least for graphites, because of their changing dimensions and hence requirements for progressively changing capsule gas gaps. Manpower requirements for this program would be relatively constant: three engineers per year with an additional three craft or technician assistance per year. The program could be completed about three years from inception date, but could be curtailed if materials fail to survive the higher fluences.

In conjunction with this irradiation program, a graphite development activity is also recommended which would take advantage of the irradiation program to test advanced structures. A minimum two-year program initiated with the irradiation testing is required, at a level of 1.5 engineers and 1.0 technician. Continuation of the program would be determined by potential success and need at the end of the two years.

The following problem areas have not been addressed, pending more definite requirements coupled with more detailed blanket design studies.

- a. If graphite is to be used at temperatures below 400°C and fluences above $2-3 \times 10^{21}$, the stored energy must be determined to prevent catastrophic temperature excursions.

- b. If graphite is to be used at temperatures above 1000-1200°C in a plasma environment, the acetylene reaction must be quantified and the D-T trapping evaluated.
- c. If graphite is to be used in fluxes where transmutation is significant, the He diffusion must be determined. At elevated temperatures, it appears to be an activated process, but the mechanism is uncertain.
- d. At the present time, graphite is being extensively investigated up to 1250°C and to 1×10^{22} n/cm². This engineering data bank may require extension in both fluence and temperature. This particularly applies to creep phenomena and their effect on shutdown stresses.
- e. The long-term reaction of Li salts with graphite must be determined if they are in contact with graphite.
- f. If pyrolytic graphite, SiC, or alloys of these are to be considered as coatings, the coating must be developed and demonstrated.
- g. If massive SiC is to be used, an engineering data bank must be established.
- h. If breeding salts are to be in contact with graphite, their compatibility must be demonstrated.
- i. If graphite is to be exposed to vacuum, its degassing properties must be established, including means to obtain the elevated temperatures required.

4.4.4 Additional Comments on Natural Resources and Industrial Capacity

For graphite, it is extremely doubtful that the envisaged use in fusion devices would ever become more than a small fraction of the current production of graphite electrodes for the steel industry. Typical nuclear graphites fall into this fabrication technology. The advanced graphites are based on what is loosely termed a "green-coke" scheme, utilizing highly chemically reactive cokes. This process is used to some extent today in the manufacture of specialty graphites for molds, the electronic industry, and discharge machining. These markets are comparable to the projected requirements of a fusion industry. The same remarks apply to bulk silicon carbide and boron carbide powders. The major disjoint would be the much more stringent requirements on silicon carbide stoichiometry and microtexture demanded by a nuclear application.

In terms of natural resources, boron and silicon are as common as the sands of the desert; and only the graphite industry's reliance on petroleum cokes may be considered a future problem. Coal-tar pitch can

certainly be developed into a suitable but expensive coke precursor. Today, electrodes are manufactured directly from coal primarily for the fertilizer and aluminum industries. At the present time, these would be totally unsatisfactory for nuclear or fusion applications. However, these applications do not require the thermal shock resistance normally essential to petroleum-coke-based electrodes; and coal may be considered as an evolutionary successor to petroleum for graphite. Such a move should not be required until the 1990s or beyond.

4.5 CERAMICS

4.5.1 Theta-Pinch First-Wall Insulator

a. What is the geometry?

The original RTPR design calls for a 0.3-mm-thickness of Al_2O_3 bonded to 1 mm of Nb-1% Zr. Subsequent designs have proposed a thin-insulator version, a continuously-graded laminate, a nonbonded system, an all-ceramic first wall, an all-ceramic blanket structure, and bumper protection.

b. What are the operating conditions?

1. Radiation fields

- (a) Neutrons — 8.1×10^{14} n/cm² sec (ave), $\sim 8 \times 10^{16}$ n/cm² sec (peak), 2.5×10^{22} n/cm² yr
- (b) Bremsstrahlung — 72 J/cm² pulse
- (c) Ionizing energy absorption rate in Al_2O_3 during burn — 7×10^8 rad/sec
- (d) Ionizing energy absorption rate in Al_2O_3 between burns — $\sim 2 \times 10^4$ rad/sec

2. Temperatures

The base operating temperature is ~ 1000 K. Temperature rises during a burn cycle for a number of first-wall designs are given in Table 4.5.1. In all cases but that of the bumper, temperatures return to the base value before the next pulse.

3. Surrounding medium

The inner surface will be exposed to D, T, and He particles, the energies of which will have been degraded by the neutral gas blanket. Particle energies, densities, state of dissociation, and state of ionization have not been quantified. The outer surface will be in indirect contact with the blanket coolant (probably liquid lithium); the intervening material will be either a thick metal structure or a thin protective metallic layer.

Table 4.5.1. Results of first-wall insulator parametric studies

Design	Temp. rise, K ^a	Max. tensile stress (strength), Pa × 10 ⁻⁸	Max. compr. stress (strength), Pa × 10 ⁻⁸
Laminar (0.3 mm Al ₂ O ₃)	326	1.50 (1.75)	6.60 (7)
Laminar (0.1 mm Al ₂ O ₃)	312	0 (1.75)	3.90 (7)
Bumper-protected 0.3 mm Al ₂ O ₃ laminar (1 mm Al ₂ O ₃ bumper)	55	0.16 (1.75)	0.67 (7)
All-ceramic (5 mm Si ₃ N ₄)	450	3.75 (7)	4.5 (>10)
Bumper (1 mm solid Al ₂ O ₃)	920		

^aBase temperature assumed to be 1000 K.

4. Stresses

Maximum thermally-induced tensile and compressive hoop stresses for a number of designs are given in Table 4.5.1. These values were calculated by 1-D and 2-D finite element analyses. First-wall insulator stresses from sources other than thermal throughput have not been quantified.

5. Electrical requirements*

(a) Surface dielectric strength — 2 kV/cm (pulsed)

(b) Bulk dielectric strength — 100 kV/cm (pulsed)

(c) Bulk resistivity — $10^2 \Omega\text{-cm}$ (pulsed).

c. Are there sufficient design data available to evaluate materials choices for the application?

Enough data are available to allow tentative choices to be made (without considering fabrication problems and long-term degradation effects). However, designs have not been optimized, e.g., with respect to materials compatibility and brittle materials design analysis, and at any rate designs are a function of reactor operating parameters which are still under study. The all-ceramic blanket concept might require major changes in blanket design, such as the use of He instead of liquid Li as a coolant and of solid Li as a T breeder to avoid Li corrosion of the ceramic. The hybrid design with a ceramic first wall and a metal blanket structure might be preferable.

d. Is a materials program required to establish a data base?

Yes. Sufficient information is available on short-term behavior to allow tentative estimates of initial material performance. However, more data are needed to allow such estimates to be made with confidence. The major need is for data on long-term behavior, i.e., performance after degradation by the severe environment. A balanced materials program should include cooperative efforts among reactor designers, fabricators, and material property investigators.

e. What testing is required?

Tests on irradiated materials at elevated temperatures are needed in the following areas:

ELECTRICAL PROPERTIES

bulk dielectric strength
 surface dielectric strength (after exposure to fuel gases)
 bulk electrical resistivity
 surface electrical resistivity
 the effect of repetitive pulses on the above

*Recent calculations of a new operating point for the RTPR result in a reduction of voltages by a factor of three and of thermal gradients by a factor of five to ten.

STRUCTURAL PROPERTIES

swelling
 strength and fracture (including long-term cyclic effects)
 thermal conductivity
 interface effects in bonded ceramic/metal systems

CHEMICAL PROPERTIES

chemical erosion in fuel gases
 effect of liquid Li and He coolants (where relevant).

The major testing problem is simulation of the radiation environment. Effects which must be simulated are:

displacement events
 H and He gas generation
 DPA
 DPA rate
 ionizing radiation
 metallic transmutation products.

Several of these must be studied together to determine if synergistic effects are present.

Testing can be carried out in the following facilities:

hard-spectrum fission reactors (for DPA)
 mixed-spectrum fission reactors (for combined DPA and transmutation effects in selected materials where doping with isotopes sensitive to thermal neutron absorption reactions can be utilized)
 14 MeV neutron sources (for displacement events, DPA, and transmutation effects)
 ion and electron accelerators (for DPA rate and ionizing radiation effects)
 high-voltage electron microscopes (for DPA, DPA rate, and ionizing radiation).

With certain facilities (heavy ion accelerators and 14 MeV neutron sources with higher-energy spectral tails) the possibility of damage by track mechanisms (massive electrostatic repulsion) must be considered.

For some tests (e.g., swelling), postirradiation evaluations will be adequate. With others (e.g., dielectric strength), measurements during bombardment are required. This will be difficult in many facilities, and impossible in some (e.g., HVEM). It may be that as studies proceed, a sufficient data base can be developed to allow various separate test results to be brought together and used to predict with confidence material behavior in a complex fusion environment. However, since such a data base does not at present exist and may not be achievable, testing in a FERF appears necessary.

Studies of the nature of radiation-induced defects must proceed in parallel with measurements on various physical properties, and experimental studies should be supported by theoretical calculations, so that damage effects can be adequately interpreted.

- f. Are some radiation damage effects in ceramics different from those in metals?

Yes. Certain ceramics are subject to amorphization (metamictization) at high doses, and bombarding particles which induce a localized high ionization density can cause track-type damage. The low-Z components of typical ceramics transmute relatively rapidly under 14 MeV neutron bombardment (roughly 2000 ppm gas/yr for Al_2O_3 in this application). Details of displacement events differ somewhat in low-Z materials from those in other materials. Although ceramics which suffer displacements from bulk ionizing radiation (the Pooley-Hersh effect) will not be used, the role of such radiation in the behavior of preexisting defects in candidate ceramics must nevertheless be considered.

- g. Is a fabrication development program required?

Yes, for two reasons: (1) fabrication capabilities needed are in some cases beyond the state of the art, and (2) physical properties tests such as those described earlier must be carried out on materials made by representative fabrication methods.

- h. Is a materials engineering program required to establish QA and QC requirements?

Yes. Quality assurance and quality control should be an integral part of the insulator development program. Plasma engineers must be involved, since the effect of material failure (e.g., insulator cracking) on plasma performance must be assessed. Only after this is done can the degree of material reliability required be specified.

- i. Can the ceramic be easily replaced?

Not in present designs, although this option should be considered as part of design optimization.

- j. What materials have been considered for this application?

Al_2O_3 , BeO , $\text{Y}_3\text{Al}_5\text{O}_{12}$, MgAl_2O_4 , Y_2O_3 , Si_3N_4 , Si_2ON_2 , SiAlON , and barium aluminosilicate glass have been studied.

- k. What are the properties of candidate materials relevant to the theta-pinch first-wall applications?

Relevant properties are summarized below:

Bulk dielectric breakdown strength. DBS in the single-pulse mode at elevated temperatures ranges from very high for single crystals (~ 2 MV/cm for Al_2O_3 , MgAl_2O_4 , and $\text{Y}_3\text{Al}_5\text{O}_{12}$) to moderate for technological ceramics (~ 100 kV/cm for Si_3N_4 and Si_2ON_2). Fabrication and impurity effects appear to control the behavior

of technological materials. Irradiation to $\sim 3 \times 10^{21}$ n/cm² ($E > 0.1$ MeV) at 1015 K in EBR-II appears to cause little or noⁿ degradation unless grain boundary separation occurs.

Bulk electrical resistivity. Resistivity of Al₂O₃ and BAS glass at ~ 1000 K is high ($\sim 10^9$ and 10^7 Ω -cm respectively). Calculations predict a tolerable reduction for Al₂O₃ under RTPR irradiation conditions, and measurements in ionizing fluxes up to 5×10^5 rad/sec generally confirm this.

Swelling. Al₂O₃ and BeO swell ~ 2 vol % under the EBR-II irradiation conditions described above (~ 3 DPA, a damage level equivalent to roughly a month's exposure at the theta-pinch reactor first wall). Y₃Al₅O₁₂, MgAl₂O₄, Y₂O₃, Si₃N₄, Si₂ON₂, and SiAlON all swell less than 0.5 vol % under these conditions. The temperature-dependence of swelling is not well-defined; however, ZrO₂ is known to show a swelling peak at ~ 875 K.

Strength. As can be seen in Table 4.5.1, some first-wall designs result in only moderate ceramic stresses. Calculations have shown that strength of unirradiated Si₃N₄ is adequate for long-term operation under fatigue conditions. Ceramics are often strengthened by irradiation. However, anisotropic swelling of noncubic polycrystalline aggregates can lead to grain boundary separation, as can precipitation of insoluble gases at boundaries.

Compatibility with fuel gases. The gas-blanket-protected theta-pinch first-wall insulator is expected to be in contact with atomic hydrogen isotopes. Studies with single-crystal Al₂O₃ in atomic hydrogen at 2000 K show that significant erosion may result. The likelihood of surface reduction has not yet been evaluated. Details of the gaseous environment (density, energy, state of charge or dissociation of particles) have not been analyzed.

Compatibility with coolant. Most ceramics are subject to chemical attack by hot liquid Li, and must be protected by a metallic interlayer. Even then, certain metals such as Nb are permeable to anions and this can allow partial reduction of the ceramic. He coolant is more benign, but trace impurities can cause either deleterious or beneficial effects.

Thermal conductivity. A first-wall insulator suffers a high thermal energy throughput, and must have a reasonably high thermal conductivity in order that temperatures and thermal stresses be kept at reasonable levels. Glasses are inadequate unless a protective bumper is used. Thermal conductivity of ceramics is reduced by irradiation; those irradiated in EBR-II suffered room-temperature reductions of from 8% to 68%. For the first-wall application, thermal stresses are inversely proportional to the square root of thermal conductivity. In Al₂O₃, point defects contribute about two-thirds of the reduction and pores one-third.

Ceramic/metal interface effects. Such interfaces may suffer degradation under irradiation due to: (1) overlap of knock-on damage, (2) enhanced interdiffusion, (3) swelling mismatch, or (4) precipitation of insoluble gases. Where swelling mismatches are unavoidable, creep or the use of a soft (e.g., glassy) interface may alleviate the problem. $\text{Al}_2\text{O}_3/\text{Nb}$ and $\text{Al}_2\text{O}_3/\text{Mo}$ laminates remain bonded after irradiation in EBR-II. Accumulation of D and T at the interface will be dependent on permeability of the ceramic and metal used, and on T concentration in the liquid Li.

1. What are the operating limits, lifetime, and failure mode for candidate materials?

A number of materials appear to be able to meet operating requirements initially, if fabrication requirements can be met. Long-term behavior cannot yet be predicted, primarily because of lack of information on radiation-induced structural changes (which affect both electrical and mechanical performance) and on chemical erosion. Reactor operating variables are another source of uncertainty. Although electrical failure of the first-wall insulator is the prime concern, this may result from structural failure (e.g., spalling) or from chemical erosion.

- m. Is there a resource availability problem?

Most insulators are made up of combinations of Si, Al, Mg, O, and N, all of which are plentiful. If such materials as Y_2O_3 and ZrO_2 become serious contenders, the availability of the metallic constituents should be considered.

- n. Based on the above answers what materials are recommended and which should be excluded?

On the assumption that materials which have performed poorly in tests to date will perform even more poorly in more stringent tests, Al_2O_3 and BeO should be excluded from consideration for first-wall usage. However, since these ceramics clearly exhibit certain deleterious effects such as swelling, further studies of their behavior can be useful in understanding and possibly alleviating degradation effects in other materials. $\text{Y}_3\text{Al}_5\text{O}_{12}$, MgAl_2O_4 , Si_3N_4 , Si_2ON_2 , SiAlON , and possibly Y_2O_3 all show sufficient promise that further studies are warranted. As degradation mechanisms become better-understood, optimized materials should be introduced and evaluated. The need for improved materials is apparent with respect to the nitrides, which are presently in the developmental stage. Chances of success for any material will be improved if a protective bumper can be used.

- o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

It is not possible at this stage of materials and reactor development to answer this question. However, no overwhelming

problems in the original RTPR design have surfaced to date which cause materials people to despair, and the new operating point generally relaxes first-wall requirements. The Subtask Group feels that it is not reasonable to think that a material cannot be developed for this application.

4.5.2 Tokamak Current Breaker*

a. What is the geometry?

The current breaker is a ring of insulating material ~ 6 m in diameter installed in a gap in the torus, if possible at the outer diameter of the blanket. Other dimensions will be set by electrical and structural requirements. The gap will be offset so that the insulator does not look at the plasma. The ring may be mechanically clamped in place, and the lack of a vacuum seal compensated by placing the torus in a building evacuated to $\sim 10^{-8}$ torr. Mechanical vacuum joints to insulating coatings and hermetic sealing of rings are also being considered.

b. What are the operating conditions?

The voltage across the ring is expected to be ~ 1 kV DC, and the temperature $\sim 500^\circ\text{C}$. No resistivity specification has been set, but will be quite modest. The neutron flux is expected to be $\sim 1 \times 10^{13}$ n/cm² sec at the outer diameter of the blanket, and 100 times higher at the inner diameter. Level of ionizing radiation is not known. Cantilevered forces of unknown magnitude will be imposed on an offset insulating ring. Compressive preloading may be induced by clamping during assembly. The ceramic will be exposed to a residual D-T gas density of $\sim 10^{14}$ /cc when located at the outer diameter of the blanket.

c. Are there sufficient design data available to evaluate materials choices for the application?

Only tentative choices can be made on the basis of present design information.

d. Is a materials program required to establish a data base?

Yes. See answer to this question in the section on theta-pinch first-wall insulators.

e. What testing is required?

A number of tests specified for the theta-pinch first-wall insulator are appropriate here. However, environmental conditions are less severe for the current breaker. Emphasis should be on vacuum joint integrity (where required), long-term strength and fatigue effects, structural radiation damage, compatibility with fuel gases, ceramic/metal interface effects, and DC polarization effects. The last is a phenomenon resulting from cation or anion drift, and can result in electrical or structural degradation.

*The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

- f. Are some radiation damage effects in ceramics different from those in metals?

Yes. (See section on theta-pinch first-wall insulators.)

- g. Is a fabrication development program required?

Yes. If a monolithic ring is required, major scale-up problems will be encountered. Insulating coatings will require development of an application technique. Lesser fabrication problems are expected if hermetically-sealed hollow metal rings are specified.

- h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be assessed. Electrical requirements are so modest that only a massive structural failure seems likely to cause problems in this area. However, failure of a vacuum joint could have serious consequences both with respect to plasma contamination and to tritium contamination of surrounding areas.

- i. Can the insulator be easily replaced?

No. Problems associated with the massive size of a tokamak torus and with inaccessibility of the current breaker dictate that the insulator be replaced no more frequently than the schedule for major torus service.

- j. What materials have been considered for this application?

Candidates include anodized coatings, mica, powdered insulator encapsulated in thin-walled metal, gas-filled metal gaskets pressed against ceramic or glass inserts, a monolithic ring of refractory ceramic or glass clamped with insulated bolts, and a ceramic-to-metal sealed ring with welded bellows and ceramic spacers.

- k. What are the properties of candidate materials relevant to the application?

See the section on theta-pinch first-wall insulators (especially comments on strength, swelling, compatibility with fuel gases, and ceramic/metal interface effects). Little is known about DC polarization effects in commercial ceramics.

- l. What are the operating limits, lifetime, and failure mode for candidate materials?

Lack of detailed fabrication and design data prevent an estimate of short-term behavior, primarily because of uncertainties in strength and stress loading. Uncertainties with respect to degradation effects preclude identification of long-term failure modes.

m. Is there a resource availability problem?

No. Only a small quantity of material is needed, and most ceramics are made of plentiful elements.

n. Based on the above answers, what materials are recommended and which should be excluded?

Uncertainties in fabrication and design preclude a detailed answer to this question. Anticipated behavior of specific materials in a fusion reactor environment is described in the section on theta-pinch first-wall insulators.

o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

Yes. Operating conditions are not intolerably severe, and material requirements are modest (unless vacuum integrity is required). The major problem appears to be fabrication; design choices should be made on the basis of recommendations by fabricators.

4.5.3 Neutral-Beam Injector Insulators (with Direct Conversion on Injectors)*

a. What is the geometry?

Injector insulators are rings ~ 1 m dia $\times$ ~ 1 cm thick, with a radial thickness of ~ 1 cm. The direct converter insulators have not been designed. Insulators can be recessed or shielded from ions. Unless an outer vacuum chamber is specified, vacuum seals will be needed.

b. What are the operating conditions?

The full voltage of the injector will be ~ 1 MV DC, and of the direct converter, ~ 100 kV DC. The voltage will be divided among several insulators, with a maximum bulk field of perhaps 50 kV/cm. Water cooling will be utilized, so that insulator temperature should not exceed $\sim 200^\circ\text{C}$. Resistivity should remain above $\sim 10^8 \Omega \text{ cm}$ to reduce ohmic heating effects. Neutron flux will be 10^{10} – 10^{12} n/cm² sec, with the direct converter receiving the higher dose. Gamma flux will be 10^5 – 10^6 R/hr from (n, γ) reactions. Other ionizing radiation has not been quantified. Residual Cs vapor and D-T gas will be present in the injector and direct converter, at maximum pressures of 10^{-2} and 10^{-4} torr respectively.

c. Are there sufficient design data available to evaluate materials choices for the application?

Only tentative choices can be made on the basis of present design information. See note following question o.

*The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

d. Is a materials program required to establish a data base?

Yes. See answer to this question in the section on theta-pinch first-wall insulators.

e. What testing is required?

Tests are needed in the areas of bulk dielectric strength, resistivity, surface flashover in the presence of ionizing radiation and possible sputtered metal deposition, and DC polarization. Long-term integrity of vacuum seals (where specified) must be evaluated.

f. Are some radiation damage effects in ceramics different from those in metals?

Yes. (See section on theta-pinch first-wall insulators.)

g. Is a fabrication development program required?

Yes. Scale-up of present fabrication techniques will require attention. It may be necessary to develop glazes and appropriate coating techniques so that controlled surface leakage can be obtained. Vacuum bonding techniques will have to be developed after both ceramic and adjoining metal have been specified.

h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be assessed. The effect of material quality on long-term dielectric strength and polarization are major areas of concern. Quality and durability of vacuum seals will also need attention.

i. Can the insulator be easily replaced?

No. If bonded vacuum seals are specified, insulator lifetime must be that of the device itself. Even with demountable seals, replacement will be difficult and tritium contamination may be a problem.

j. What materials have been considered for this application?

Candidates include Al_2O_3 , Y_2O_3 , MgO , SiO_2 , and BeO . Glasses may be preferable to ceramics from the standpoint of fabricability, but will be more difficult to water-cool. Titanium or manganese oxide glazes are being considered to obtain controlled surface leakage.

k. What are the properties of candidate materials relevant to the application?

See the section on theta-pinch first-wall insulators (especially comments on bulk dielectric breakdown strength, bulk resistivity, and strength). DC breakdown strength is lower than that in the pulsed mode at elevated temperatures. Surface flashover is

often a property of the gaseous environment rather than of the ceramic surface. Little is known about DC polarization effects in commercial ceramics, although the effect may not be significant at 200°C.

1. What are the operating limits, lifetime, and failure mode for candidate materials?

Lack of detailed fabrication and design data prevent an estimate of short-term behavior, primarily because of uncertainties with respect to bulk dielectric strength and surface flashover. Uncertainties with respect to polarization, sputtered metal effects, and long-term vacuum integrity prevent identification of long-term failure modes.

- m. Is there a resource availability problem?

No. The amount of material involved is not large, and most ceramics are made of plentiful elements.

- n. Based on the above answers, what materials are recommended and which should be excluded?

Uncertainties in fabrication and design disallow an answer to this question.

- o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

Yes. While material problems are not well-defined, the present judgment is that they are not insurmountable. The greatest problems may lie in the realm of device operating parameters, e.g., ionized gas-induced surface flashover or sputtered metal deposition.

Note: Devices added to the neutral-beam injector will themselves have materials problems. If laser photo-stripping is deemed necessary, the laser and associated dielectric mirrors (used to reflect the beam across the gas stream several times) will be subjected to environmental conditions similar to those for the injector insulators. Also, electronic circuitry (with associated insulators) may be required in order to control surges of breakdown current. This circuitry will operate in a flux of $\sim 10^{10}$ n/cm² sec.

4.5.4 Mirror Direct Converter Insulators*

- a. What is the geometry?

These insulators are parts of vacuum-sealed feedthroughs in the wall of the direct converter. Dimensions have not been fixed, but large units can be used to reduce electrical requirements.

*The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

Insulators will be recessed and shielded against ion bombardment and sputter-deposition.

b. What are the operating conditions?

The voltage to be held off will be 180 kV DC or less. Fluxes of all forms of radiation are expected to be low, except possibly for ionizing radiation. Values for the latter have not been calculated. If thermal conversion as well as direct conversion is utilized, insulator temperature could be quite high ($\sim 1000^{\circ}\text{C}$). Without thermal conversion, temperatures will be lower. Water cooling may be utilized to reduce insulator temperatures. D-T gas density has not been specified.

c. Are there sufficient design data available to evaluate materials choices for the application?

Only tentative choices can be made on the basis of present design information.

d. Is a materials program required to establish a data base?

Yes. The size of the effort needed will depend on whether insulator temperature is high enough to degrade electrical properties. If temperatures are forced downward by water cooling, the major problem may be structural (thermal stresses) rather than electrical.

e. What testing is required?

If insulator temperatures are high, primary problem areas are bulk DC breakdown, resistivity, and polarization. Gaseous erosion is possible if gas density is high. Thermal stresses may be significant. Long-term integrity of vacuum seals should be evaluated.

f. Are some radiation damage effects in ceramics different from those in metals?

Probably not applicable. Particulate damage doses are expected to be low.

g. Is a fabrication development program required?

If conventional ceramics can be used, fabrication development requirements should be modest. If it is found that a special ceramic is needed (e.g., to resist polarization), a greater fabrication effort will be needed.

h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be assessed. The effect of material quality on long-term electrical and mechanical properties are primary areas of concern. Quality and durability of vacuum seals will also need attention.

i. Can the insulator be easily replaced?

Probably so, if the design is such that the insulator/metal feedthrough can be mechanically attached to the wall of the direct convertor. However, tritium contamination may be a problem.

j. What materials have been considered for this application?

No specific candidates have been designated to date. A number of conventional insulating ceramics appear to be worthy of consideration. Glasses may be preferable to ceramics from the standpoint of fabricability, but will be more difficult to water-cool.

k. What are the properties of candidate materials relevant to the application?

See the section on theta-pinch first-wall insulators. DC breakdown strength and resistivity can be quite low at 1000°C, and polarization high. Thus lower operating temperatures will probably be required. Surface flashover is often a property of the gaseous environment rather than of the ceramic surface.

l. What are the operating limits, lifetime, and failure mode for candidate materials?

Lack of detailed fabrication and design data prevent an estimate of short-term behavior, primarily because of uncertainties with respect to bulk DC breakdown, resistivity, and surface flashover. Uncertainties with respect to polarization, thermal stresses, and long-term vacuum integrity prevent identification of long-term failure modes.

m. Is there a resource availability problem?

No. Only a small quantity of material is needed, and most ceramics are made of plentiful elements.

n. Based on the above answers, what materials are recommended and which should be excluded?

Uncertainties as to design and operating temperature disallow an answer to this question.

o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

Yes, if operating temperatures can be kept low enough to avoid severe problems with respect to electrical properties. It appears likely that this can be achieved, although at the cost of increased thermal stresses. However, the latter problem appears (without benefit of reference to stress calculations) to be tolerable.

4.5.5 Low-Z First Walls and Liners*

a. What is the geometry?

A number of geometries have been proposed. Low-Z first walls might be the inner surface of an all-ceramic blanket, or a ceramic or glass liner bonded to a metal blanket structure. Separate protective liner concepts include ceramic tiles, woven ceramic cloth, or a venetian blind bumper structure. Dimensions have in most cases not been specified.

b. What are the operating conditions?

First wall and bonded liner structures, being cooled by liquid Li or gaseous He, will operate near blanket temperatures (roughly 800-1000 K). Separate protective liners must be cooled by radiation (unless heat pipes can be utilized) and will operate in the range of 1300-2000 K. Neutron fluxes will be high ($\sim 10^{15}$ n/cm² sec), as will D-T and He fluxes. Without a gas blanket, ion wall loadings for a tokamak will be roughly:

D-T ---- 1×10^{14} /cm² sec, 23 keV

He ---- 5×10^{12} /cm² sec, 100 keV.

Tiles for a tokamak may use a flowing D-T gas layer across the surface to sweep away impurities. Details of this gas layer have not been developed.

The theta-pinch first wall will be protected by a gas blanket; density and energy of gas particles have not been calculated in detail, but the predominant damaging species is expected to be atomic hydrogen. Ionizing radiation loading will be high. Thermal throughput will be significant, especially in the theta-pinch. For this reactor, wall loadings are pulsed, and pulsed thermal stresses will be high. Startup and shutdown (end-of-pulse) conditions for tokamaks may cause thermal load problems.

c. Are there sufficient design data available to evaluate materials choices for the application?

Only tentative choices can be made on the basis of present design information.

d. Is a materials program required to establish a data base?

Yes. The data base for this application is clearly inadequate. Tests of the type described in the next question are needed to address both short-term and long-term problems.

e. What testing is required?

Tests should be conducted at medium and high temperatures on D-T chemical erosion; ion sputtering; radiation damage effects

*The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

on strength, swelling, thermal stability, and thermal conductivity; and emissivity changes. Electrical resistivity of theta-pinch bumpers must be evaluated.

- f. Are some radiation damage effects in ceramics different from those in metals?

Yes. (See section on theta-pinch first-wall insulators.)

- g. Is a fabrication development program required?

Yes. Development of fabrication techniques for successful coatings may be difficult. Ceramic fabrics are available, but have not been developed with this application in mind. Tile fabrication should present the fewest problems, unless composites, foam, or honeycomb structures are desired.

- h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be addressed. Input from plasma engineers will be needed to ascertain the significance of various failure modes to reactor performance. Designs which minimize the consequences of failure should be identified and adopted so that QA and QC requirements can be minimized.

- i. Can the ceramic be easily replaced?

A major goal for separate liner concepts is easy replacement. Bonded liners cannot easily be replaced.

- j. What materials have been considered for this application?

BeO, graphite, B₄C, SiC, Al₂O₃, Si₃N₄, Be₂C, TiC, BN, SiAlON, Si₂ON₂, AlN, MgO·Al₂O₃, and low-Z composites have all been mentioned. Glasses have less potential for this use than do ceramics where high thermal conductivity is needed.

- k. What are the properties of candidate materials relevant to the application?

See the section on theta-pinch first-wall insulators. Vaporization becomes a problem for Si₃N₄ in the range 1100-1600°C. Swelling of SiC is moderate (0 to 2 vol %) between 400 and 1400°C, at a fission neutron fluence of 10^{22} n/cm² ($E_n > 0.18$ MeV). Chemical erosion of graphite at high temperatures appears to be high. Thermal conductivity is reduced by radiation damage-induced defects. Electrical resistivity of insulators at bumper temperatures may be rather low; however, the minimum acceptable value has not been specified and may be modest.

- l. What are the operating limits, lifetime, and failure mode for candidate materials?

Not enough materials data are available to allow this question to be answered. Lifetime of replaceable liners can be

comparatively short. Major uncertainties are associated with D-T chemical erosion, bulk irradiation effects, thermal stress conditions, and plasma tolerance of impurity effects,

m. Is there a resource availability problem?

See the section on theta-pinch first-wall insulators. Less ceramic material is needed here than is required for the theta-pinch, which also calls for blanket intersegment insulators.

n. Based on the above answers, what materials are recommended and which should be excluded?

Lack of materials data precludes a detailed answer to this question. Si_3N_4 and perhaps some other ceramics may not be useful for very high temperature applications due to high vaporization rates. Anticipated behavior of specific materials in a first-wall environment is discussed in the section on theta-pinch first-wall insulators.

o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

See the answer to this question in the section on theta-pinch first-wall insulators. The use of detachable rather than bonded liners will improve chances for success if the higher operating temperatures can be tolerated.

4.5.6 Insulators for Tokamak RF Heating Systems*

a. What is the geometry?

Two systems are under consideration. One utilizes waveguides to bring power through the magnets and blanket to the first wall; the other utilizes a coaxial cable. A design for the former specifies a $0.5 \text{ m} \times 0.5 \text{ m}$ cross-section waveguide 3 to 10 m long, filled with dense (i.e., not powdered) ceramic. The waveguide will terminate after passing through the first wall, except that some tailoring of the ceramic surface might be required (e.g., a convex curvature). The metal structure of the waveguide will be Li-cooled. A shielding door may be closed over the waveguide surface after a 1 to 10 sec startup, to minimize radiation damage. It may be possible to recess the end of the ceramic away from the first wall, if RF transmission requirements can be met.** A coaxial system will utilize an RF antenna or hoop (which can have a variety of shapes) inside the

*The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

**An RF system operating at 100 GHz could perhaps be located outside of the blanket.

plasma chamber, connected to the center lead and the chamber wall. A dielectric coating will be required in order to suppress arcing. A limiter may be used around the antenna to at least partially divert plasma ions.

b. What are the operating conditions?

Typical electrical specifications are: 100 to 200 MW of 60 to 100 MHz RF power, 1 to 10 kV/cm electric field across the dielectric, ~ 50 kW/cm² power density, and a loss tangent $\leq 10^{-4}$. Operating temperature will be nominally that of the blanket and first wall (~ 500 – 600°C) but as-yet uncalculated RF, neutron, and other heating effects may raise the temperature of the ceramic considerably. Radiation fields (without a shield) will be those characteristic of a tokamak (see the section on Low-Z first walls and liners). The inner surface of the ceramic will be exposed to residual D-T gas. Stresses from thermal and swelling gradients may be large. Coatings on RF antennas will be subjected to severe abuse.

c. Are there sufficient design data available to evaluate materials choices for the application?

Only tentative choices can be made on the basis of present design information.

d. Is a materials program required to establish a data base?

Yes. This application is perhaps the most difficult addressed in this report. Data needs are similar to those described in the section on theta-pinch first-wall insulators. Both short-term and long-term problems appear to be severe.

e. What testing is required?

Tests needed for this application are similar to those described for the theta-pinch first-wall insulator. An additional requirement (low loss tangent at high temperatures under irradiation) is applicable here. The large mass of ceramic material needed will magnify thermal and stress problems.

f. Are some radiation damage effects in ceramics different from those in metals?

Yes. (See section on theta-pinch first-wall insulators.)

g. Is a fabrication development program required?

Yes. Comments in the section on theta-pinch first-wall insulators are applicable, but the need for massive dense insulators may make RF fabrication requirements even more severe.

h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be addressed. Because of the massive

insulators required, QA and QC will be difficult. RF engineers must specify flaw tolerance of the system.

i. Can the ceramic be easily replaced?

The massive ceramic pieces cannot be easily replaced. Periodic replacement of the antenna by remote techniques may be feasible.

j. What materials have been considered for this application?

Macor (a Corning machinable glass-ceramic) has been used successfully in a coaxial system. Other candidates which have been mentioned are Al_2O_3 and Y_2O_3 . Mo and Cu metal parts have been considered. Glassy antenna coatings have been used, but their thermal conductivity is probably too low for reactor usage.

k. What are the properties of candidate materials relevant to this application?

See comments in the section on theta-pinch first-wall insulators. A question exists as to whether any technological ceramic can meet the loss tangent requirements at elevated temperatures in the irradiated condition.

l. What are the operating limits, lifetime, and failure mode for candidate materials?

The comments in the section on theta-pinch first-wall insulators are generally applicable here. However, even short-term performance is in question if (a) heating in massive ceramic parts is excessive, (b) ceramics with acceptably low flaw contents cannot be produced, or (c) abuse of insulating coatings on antennas is intolerably high.

m. Is there a resource availability problem?

See the section on theta-pinch first-wall insulators.

n. Based on the above answers, what materials are recommended and which should be excluded?

See the section on theta-pinch first-wall insulators. However, the additional requirement of a low loss tangent has not been assessed for the materials discussed there. A glassy rather than ceramic insulator may be preferable for the bulk needs in waveguides or coaxial systems, if the lower thermal conductivity of glasses can be tolerated. A protective bumper for the RF port should be considered.

o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

The answer to this question in the section on theta-pinch first-wall insulators is generally applicable here. More pessimism should perhaps be applied to the RF application, but this should be qualified by the fact that to date little effort

has been put forth to bring materials and electrical experts together to optimize the system. The use of a shield or bumper can be expected to reduce material problems.

4.5.7 Insulators for Magnetic Coils^{*†}

a. What is the geometry?

Most coil insulators are interleaved between metallic windings. A variety of geometries is being considered, e.g., monolithic ceramics, encapsulated powdered ceramics and strips of organic insulator with or without ceramic inserts. The theta-pinch implosion-heating coil is different from the above; here the coil is a top-hat-shaped ceramic body with thin conducting strips bonded to the inner and outer surface and coated with insulator. Hat size is 1.8 m, with a 20-30 cm brim.

b. What are the operating conditions?

These coils will be fluid-cooled as needed and will operate at or slightly above RT.** All will be located outside of blanket and/or shielding, so that radiation fields will be greatly reduced from those at first walls.*** Neutron fluxes for the theta-pinch implosion coil are $\sim 6 \times 10^{13}$ n/cm² sec, and for the compression coil, $\sim 5 \times 10^{13}$ n/cm² sec. Ionizing energy absorption rates in Al₂O₃ during and after burn for the former coil are $\sim 4 \times 10^6$ and $\sim 4 \times 10^3$ rad/sec. The corresponding values for the latter coil are $\sim 2 \times 10^6$ and $\sim 2 \times 10^3$ rad/sec. Radiation fields for most other applications should be lower, but are in most cases not well-defined. Stresses are expected to be high in several applications. The implosion coil requires a bulk dielectric strength of ~ 100 kV/cm (pulsed). Ohmic heating coils may operate at a few kV. For other applications the voltages are expected to be low.

c. Are there sufficient design data available to evaluate materials choices for the application?

For the theta-pinch coils, yes. For other applications the situation is variable.

*Excluding sc coils, which are being considered by another subtask group.

†The reader is referred to the section on theta-pinch first-wall insulators for a more detailed discussion of some questions.

**Cryogenic but non-sc coils are being considered for some applications.

***A possible exception is the tokamak F-coil, which in some designs is placed in a high-flux region near the first wall.

d. Is a materials program required to establish a data base?

A modest program seems necessary, but this may be a part of ongoing engineering development. Development of organic insulators may require more effort, but could be combined with work in this area for SC coils.

e. What testing is required?

Tests should be conducted on long-term strength* behavior and multipulse dielectric breakdown strength (for high-voltage applications) on materials in the as-fabricated condition. Radiation effects should be evaluated for the theta-pinch coils and for others which may be exposed to significant fluences. The much lower radiation tolerance of organic insulators compared with ceramics must be taken into account.

f. Are some radiation damage effects in ceramics different from those in metals?

Yes. (See section on theta-pinch first-wall insulators.)

g. Is a fabrication development program required?

Such a program is needed for the theta-pinch implosion coil, and may be required for other applications, depending on design and operating conditions. Fabrication development for the latter may be a part of ongoing engineering development.

h. Is a materials engineering program required to establish QA and QC requirements?

Yes. As with any engineering application, the consequences of material failure must be addressed. Even in those cases where operating conditions are not severe, the long lifetime requirements necessitate attention to quality of starting materials.

i. Can the ceramic be easily replaced?

No. Complexity of coils and their inaccessibility preclude easy replacement.

j. What materials have been considered for this application?

Castable ceramics are being considered for the implosion coil. Candidate interleaved insulators include epoxy fiberglass laminates, Mylar, Vespel, Kapton, mica, powdered MgO, and bulk ceramics (e.g., Si_3N_4).

k. What are the properties of candidate materials relevant to the application?

See the section on theta-pinch insulators in reference to structural radiation damage of the implosion coil. Dielectric strength may be more a function of fabrication parameters than

*Including friction, wear, galling, and fatigue.

of the ceramic itself, and so cannot be meaningfully addressed here. Where stresses are compressive, insulator strength is high; tensile stresses should be avoided or minimized. Organic insulators are generally limited to radiation fluences of $\sim 10^{18}$ n/cm² or $\sim 10^9$ rad.

1. What are the operating limits, lifetime, and failure mode for candidate materials?

Not enough design and fabrication data are available on interleaved insulators to allow this question to be answered. For the implosion coil, long-term degradation of mechanical or electrical strength is the likely failure mode.

- m. Is there a resource availability problem?

No. Although a significant quantity of insulating material may be required, such material is usually composed of plentiful elements.

- n. Based on the above answers, what materials are recommended and which should be avoided?

Organic insulators should be excluded from use in radiation fields high enough to cause significant degradation. Radiation fields for theta-pinch coils are high enough that high-swelling ceramics (Al₂O₃, BeO) should be avoided. A host of other insulators appear worthy of consideration for use in magnet coils.

- o. Is it reasonable to believe that a material can be developed to meet the needs of this application?

Yes. The theta-pinch implosion-heating coil requirement presents an only moderately severe materials challenge, and most other applications for coil insulators appear to be less difficult.

SUMMARY

This technology assessment has considered seven applications for insulators and ceramics in fusion reactors:

- theta-pinch first-wall insulator
- tokamak current breaker
- neutral beam injector insulators
- mirror direct convertor insulators
- low-Z first-walls and liners
- insulators for tokamak RF heating systems
- insulators for magnetic coils.

The most difficult appear to be the theta-pinch first-wall insulator and tokamak RF heating insulators. Nevertheless, there is no compelling

reason at this time to believe that these requirements cannot be met. Other applications also offer a significant materials challenge; however, all but the first-wall insulator lack the definition of design and operating conditions required for a detailed description of their problems.

A major observation of this technical assessment is that most fusion reactor insulator and ceramic problems are generic in nature, i.e., are common to several applications and reactor concepts. Consider the extent to which this is true for the major R & D problem areas:

<u>PROBLEM</u>	<u>NUMBER OF APPLICATIONS (of 7)</u>	<u>REACTORS AFFECTED</u>
swelling	5	T, θ
bulk dielectric strength	5	T, M, θ
surface dielectric strength	5	T, M, θ
bond integrity	7	T, M, θ
mechanical properties	6	T, M, θ
gas/solid reactions	6	T, M, θ
degradation in thermal conductivity	3	T, θ
14 MeV neutron damage	5	T, θ
DC polarization	3	T, M
thermal stresses	5	T, M, θ
synergistic damage effects	5	T, θ
ion sputtering	2	T, M
bulk resistivity	5	T, M
thermal stability	4	T, M, θ
degradation of organic insulators	1	T, M
loss tangent	1	T

These observations strongly suggest that initial R & D studies should be primarily generic in nature.

A major problem area common to all applications is fabrication. This critical topic is receiving little attention at this time; yet many applications present severe fabrication problems, sometimes beyond the state of the art. Lack of definitiveness of designs hampers identification of specific fabrication requirements; however, considerable progress could be made now in development of broad capabilities, e.g., ceramic-to-metal coating techniques. Early fabrication studies are needed for another reason — evaluation of relevant physical properties must be performed on realistically-made materials in order that results have engineering applicability.

The data base applicable to fusion reactor insulator and ceramic problems is rather weak, particularly in areas of radiation damage, surface electrical properties, polarization effects, gas/solid interactions, and fabrication. Some progress has been made in recent years

by the DMFE Insulator and Ceramic R & D projects, but support to date has been inadequate to allow anywhere near the rate of progress needed. An estimate has been made of the size of a program sufficient to support development of insulators and ceramics for fusion reactors. Staff for an R & D program to address generic problems builds to a total of 20 professionals and 8 technicians in FY 1987, with costs per year reaching ~\$1600 K in constant FY 77 dollars at that time. It is estimated that an effort roughly the size of that for R & D (10 professionals and 20 technicians by FY 1987) will be required for fabrication development.

Finally, problems of overlap of this program with other DMFE materials projects should be addressed. It became apparent during this exercise that some of the distinctions between programs are awkward. For example, development of a fusion reactor insulator might involve:

- studies of surface electrical properties by the Special Purpose Materials Program
- studies of surface degradation effects (which affect surface electrical properties) by the Plasma-Materials Interaction Program
- studies of strength and fatigue behavior by the Special Purpose Materials Program
- studies of structural design of brittle materials (which affect strength and fatigue behavior) by the Alloy Development Program.

Unless care is taken to achieve good cooperation and liaison among programs, materials development activities could suffer. Problems of overlap have to some extent been ignored in this technical assessment. Thus surface effects were included in discussions of long-term ceramic degradation, and low-Z first walls and liners were included as a problem area because the data base needed is similar to that for several other applications addressed here.

4.6 HEAT-SINK MATERIALS

a. Name the application.

It appears that heat sink materials, probably several different types, will be required in Tokamak machines for several applications. The uses will include beam stops, armour, limiters, divertor targets, calorimeters and others. Mirror machine direct convertors are also likely to use heat sink materials.

b. What are the operating conditions of temperature, stress, magnetic field, T concentration, coolant, container or support, neutron flux and fluence, gamma flux and fluence, other?

The heat sink materials will in general be exposed to the conditions of the first wall during operation, however, because of the direct bombardment with high energy particles heat sinks will experience much higher heat and particle fluxes. Peak

power densities will be high perhaps up to 8 or 10 kW/cm² for commercial reactors but numbers are not available at this time. Calculations made for LLL's FERF design gave 3.3 kW/cm² as the peak power impinging on the neutral beam stops and a peak particle flux density of 2.8×10^{17} particles/cm²/s. Most of the available information is for short ~0.5 s pulses rather than the long pulses required in commercial plants.

Perhaps the most severe operating conditions the heat sinks will experience will be the rapid temperature changes, ~15000°C/s, upon startup causing very high thermal stresses.

Severe sputtering conditions also will exist due to the impingement of high energy ions on divertor targets. For example UWMAK III has prepared a 0.1 thick TZM sacrificial plate to be sputtered away to protect the cooling arrangement. The erosion rate is 0.041 cm³/s.

- c. Are there sufficient design data available to evaluate materials choices for the application?

The answer is no. It is clear that cooling will be required regardless of the material chosen. In general the designs of beam stops, limiters, divertors, etc., have been largely ignored in the conceptual plant reports. It is important that such design studies be done with input from the materials community.

- d. What materials have been considered for the application?

Copper heads the list for present day machines. Other materials considered include the refractory metals, tungsten, molybdenum, tantalum and vanadium, the high melting carbide-carbon composites (carbides of tantalum, niobium, hafnium, zirconium, boron, titanium and mixtures of same). Graphite, aluminum, titanium and zirconium are also candidates. In addition to the above and their alloys consideration needs to be given to heat pipes and to composite structures such as copper-tungsten.

- e. List the properties of each relevant to the application.

Thermal stress resistance
 Thermal conductivity
 Coefficient of thermal expansion
 Specific heat
 Melting range
 Density
 Sputtering rate
 Radiation damage resistance
 Stability in operating environment
 Mechanical properties at operating temperature with radiation damage
 Fabricability
 Z number

- f. Is the material compatible with its environment?

Hydride formers such as carbon, tantalum, niobium, titanium and zirconium pose potential problems if they reach their hydriding temperature when in contact with plasma gases.

- g. What are the operating limits for each material?

They will probably be determined by the thermal stress requirements, with further limits set by melting points, sputtering and radiation damage resistance.

- h. What is the lifetime of each material?

Unknown. Might very well be the limiting materials in the reactors.

- i. What are the failure modes?

Melting, fracture and vaporization.

- j. What are the advantages and disadvantages of each material proposed for the application?

This is very much dependent upon design.

- k. Can the component be easily replaced?

Again design dependent but items like beam dumps and limiters would be very difficult to replace.

- l. Is a fabrication development program required?

Absolutely yes. It is almost certain that combinations of materials will be required or that refractory metals will be necessary and fabrication will be a severe problem.

- m. Is a material engineering program required to establish QA and QC requirements and a data base?

Yes, particularly for thermal stress, sputtering and radiation damage assessment.

- n. Is there a resource availability problem?

Probably not. Tantalum is in short supply but is not likely to be the top candidate.

- o. What testing is required?

The properties listed under question e should be determined. The final test will be in a reactor. There is no other way to really test these components.

- p. Where would the testing be done?

The National Laboratories have good facilities for the testing. Radiation testing hits the usual snags about spectra.

- q. Is this simulation adequate?

The only sure tests will be conducted with full size components in an operating fusion machine.

- r. Based on your answers to the above, what materials are recommended and which should be excluded?

This question is almost impossible to answer presently due to lack of designs. Copper will almost certainly be involved and probably one of the refractory metals as a coating. For applications such as armour plates of refractory metal may be used. One should not discard the carbide-carbon composites as potential heat sinks at this time.

- s. Describe the development program required for recommended materials, including schedule, manpower, and costs.

Several parallel R and D programs need to be undertaken to ensure that the engineering, fabricability and data base problems are in hand by the time they are needed for future machines. The machines under construction now are having problems due to lack of this type information on heat sinks. The program should include:

1. Design Studies
2. Materials Characterization
3. Fabrication and Engineering
4. Radiation Damage Effects

4.7 HIGH FIELD SUPERCONDUCTING MAGNETS

4.7.1 Introduction

This assessment addresses a very particular set of questions related to the applications of superconducting magnets in MFE devices. It is important to realize that, with very few exceptions, existing fusion devices do not use superconducting magnets. Furthermore, only a very limited number of large superconducting magnets have ever been built and these are dc coils using ductile NbTi superconductors. It is, thus, not unreasonable to expect that materials problems will be encountered in MFE applications requiring large coils, time varying fields and perhaps, the brittle high field superconductor Nb₃Sn. We have attempted to at least highlight some of these problems.

The remainder of the report indicates the specific questions considered by the group and a summary of their responses. In some instances, as one might expect, there was some disagreement on the answers and that is indicated in the "write-up." As an introduction to the overall materials problems associated with cold magnet structures, an Attachment is included, containing three outlines which list magnet components, environment, and operations on materials. In actual practice, these must be considered. This assessment treats only the most critical. There are also other serious questions related to copper coils operating at room temperature and above, which were not considered in this assessment.

4.7.2 Questions on High Field Superconducting Magnet Materials

- a. Is it possible to design a 12-16 T magnet at 0.1% conductor strain with existing structural materials?

The answer depends, of course, on the parameters of the coil such as the size, shape, overall current density and expected force distribution. It is questionable whether the 0.1% strain limit is a realistic limit; it may well be 0.2-0.3%. Cabling techniques for the conductor can be used to increase the allowable strain. The quality control needed for wire production and magnet winding would be very difficult to achieve if the lower strain level is required. For EPR-sized, and smaller, magnets one may not have the option of using sufficient structural material to keep the strain level down. If economically sensible materials are to be used magnets at the 16 T level may not be realistic. On the other hand, if the strain constraint can be relaxed, large high field magnets seem to be feasible.

- b. Should high field superconductors other than Nb₃Sn be considered?

Not by DMFE. The only reason to make an exception would be if it turned out that V₃Ga had significantly better resistance to strain than Nb₃Sn — an unlikely situation. High field superconductor research should be pursued by DPR or NSF.

- c. Is it possible to develop strain- and fatigue-tolerant Nb₃Sn conductors?

Yes. It appears that either clever cabling techniques or methods of modifying the internal Nb₃Sn layer (waves, twists, coils) will be necessary. In the large conductors, an effective resistivity of $\sim 10^{-12}$ Ω cm is adequate for most applications so that "perfect" superconducting behavior is not a requirement. The Tsuei-wire concept which involves short filaments created by drawing a bronze boule containing Nb precipitates, and reacting the resulting wire, is a technique which may also show promise. However, its state of development is such that it would be better treated as a basic research project.

One should note that strain tolerance and fatigue tolerance are two different things. At present there is a bit of data on strain effects in practical Nb₃Sn conductors but none on fatigue effects.

The effect of such a conductor on the other materials would be primarily to place the emphasis on strength rather than modulus in the materials chosen for coil containment. For instance, this development might allow the use of fiberglass-epoxy for pulsed coil confinement, thereby eliminating eddy current losses.

- d. What are the relative merits of aluminum and copper as a stabilizing metal for superconducting magnets?

The consensus is that copper is the most desirable of the two metals for this application in MFE magnets. In some particular applications the opposite may be true. The important point is that one must consider the final state of the stabilizer in service and not the purity of available metal. Sufficient data exist to allow the designer to make at least an initial choice. The trade-offs are: Aluminum is available with somewhat higher purity, its resistivity at 4 K is more sensitive to strain and radiation damage than that of copper. Neutron irradiation strengthens both metals. Warming to room temperature recovers the resistivity of Al 100% and all increase in strength is lost. Copper recovers only ~80%, but none of the increased strength is lost. Successive cycles of irradiation and annealing cause a continual degradation of the copper resistivity. Complete recovery occurs at 350°C. Copper has a higher magnetoresistance. Conductor joints, necessary in any large magnet, are likely to be more of a problem with aluminum. We have not addressed the problems of resource availability in this discussion. In theory at least, aluminum is available in greater supply than copper, a fact which might bear on this decision in the far future.

- e. Is it possible to combine the stabilizing function, now served by pure copper, and the strengthening function, now served by stainless steel, in one metal, thereby simplifying the structure?

Yes. Depending on the application and the allowable stress, cold worked copper is a possible material. There is a problem in soldering this copper to that carrying the superconducting filaments without annealing out the cold work. Dispersion hardening of the stabilizer should also be considered. Inorganic fibers (not boron) could possibly be added to the stabilizer to form a low resistivity metal matrix composite. Some of these schemes could have the added advantage of reducing eddy current losses in the pulsed magnets.

- f. To what extent will the magnet materials be influenced by the rise time of the poloidal fields in tokamak systems?

In most applications the material choice will not be affected but the geometry of the system will change drastically, cf., monolithic conductors in TF coils to a cable construction in poloidal field coils. In some cases, however, it will be a critical factor. The poloidal field coil confining structure, for example, may have to be made of a nonconducting material for devices with short rise times (<1 s).

- g. Should new alloys be developed for low temperature structural applications?

No. We must instead expand the data base on existing materials which are readily available in a variety of commercial products

forms. Particular attention must be paid to weldability and to weldment properties. In addition, reliable specifications are needed for metals to be used in low temperature applications.

- h. To what extent can structural composites be used in magnet systems?

Their extensive use in superconducting magnets to date has proven their desirability for insulating applications. It is imperative that the sensitivity of their thermal, mechanical and electrical properties to radiation damage (see question j, below) be determined soon. If they are to be required for large scale applications, such as coil forms, the existing data base on mechanical properties must be greatly expanded. Specifications are needed. It is not widely known, for instance, that "G 10" is an electrical specification *only*. It says nothing about the exact makeup of the composite in terms of resin or type of glass.

- i. How much of a data base and how much in-service experience is necessary before a new material will be accepted for application in MFE devices?

Unless a need arises which absolutely cannot be satisfied by existing materials, the requirements are such that one probably should not consider completely new materials without allowing at least ten years for large scale production and application. Clearly, if a need is truly critical, this time can be shortened appreciably by the application of large sums of money. Still, five years is a reasonable minimum.

- j. What determines the irradiation limits for superconducting magnets: a. conductor; b. stabilizer; c. insulators; d. structure; e. dewar superinsulation?

Here the usual consensus of the group breaks down a bit. One problem arises in defining how the term "limits" applies to the stabilizer. In a TNS device, frequent warming is to be expected and stabilizer damage will not be the problem that it would be in a power reactor system where down time is inversely related to profits. Another problem concerns how much one is willing to accept the "perfect shielding scheme" assumed in most damage evaluations — in reality, holes in the shield will be necessary and localized regions of greatly increased radiation damage will most likely exist.

The majority of this group feels that the organic insulators and the dewar superinsulation will prove to be the limiting materials. Secondary gamma radiation is the serious problem. Warming of the magnet to anneal the stabilizer could well result in total destruction of the insulation. The behavior of these materials under irradiation at low temperatures is such a serious question that it is difficult to understand why so little work has been done. It seems that a large scale screening project will be necessary in the near future.

Others of the group indicated that they felt that the superconductor itself was the limiting material. All available data show that neutron damage does affect the critical parameters of the superconductors. The damage can usually be removed by a room temperature anneal. More work is needed on "practical" wires with radiation closely simulating that expected in working devices.

There is no question but that neutron radiation effects are one of the major concerns of the entire superconducting magnet community.

k. Can superconducting magnets be fabricated with entirely inorganic electrical insulation?

The answer here is a guarded yes. So little information exists that a definitive evaluation is not possible. Anodizing of copper or aluminum has been proposed (and used), but conductor motion tends to damage the fragile oxide. Various slurries and materials with organic binders that are baked off were suggested. In general, a screening program to evaluate the material already developed for high temperature insulation applications at low temperatures is needed.

More strain tolerant Nb₃Sn superconductors are needed for high field (>10 T) applications in large magnets. Quality control and inspection techniques for these wires are in need of further development.

In order to provide a range of choices to the magnet designers, the low temperature data base on all pertinent existing materials is in need of expansion. In general, new materials will not be necessary, at least through EPR or DEMO time development and construction.

The rise times of the pulsed tokamak fields may be of crucial importance to magnet design and materials choices in some instances. We hope that a dependable minimum value for this parameter will be available in the near future. Anticipating that it may be quite short, a program to make a serious evaluation of the potential for application of composite materials in the devices should be initiated. Experimental programs to evaluate eddy current losses in various materials are also desirable.

As we have already pointed out, neutron and gamma ray damage studies in insulators at low temperatures are sorely needed. It is not yet completely clear that a problem exists, although the few experiments done to date indicate that at least some organic materials are completely useless in a high flux environment. A screening program, of significant size, to evaluate polymer films, bulk organic insulators and organic matrix composite materials for suitability for low temperature service in a radiation environment is needed. It would also be useful if a careful general evaluation of the *total* radiation seen by each of the superconducting coil types were available. This, of course, depends in

detail on the blanket and shield configuration, but nevertheless a thorough review, including some assessment of the expected effect of "holes" through the magnet region, could be made. The degradation of the superconducting composite due to neutrons will require more study, particularly in the case of Nb_3Sn conductors. Again, studies of properties while under irradiation are desirable.

The research requirements with regard to inorganic insulations for low temperature service are highly dependent on the final assessment as to the suitability of organics. It is unlikely that inorganics will be used unless it is absolutely necessary. Since a definitive evaluation of organics may not be available for some time, it seems prudent to commence a relatively small screening program to look at the properties of a variety of inorganic insulations at low temperatures and under thermal cycling.

Resource availability does not appear to be a serious problem in magnet systems. In a far-future "fusion economy" situation, economics may force certain trade-offs, aluminum instead of copper for example, but most materials required will be available or will have quite acceptable substitutes. The one exception may be niobium metal, but even there, it is the quality of the ore rather than its abundance which may create a problem, and one would hope that a number of techniques could be developed to use the lower grade ore if it becomes necessary.

ATTACHMENT

MFE DEVICE COMPONENTS ASSOCIATED WITH
SUPERCONDUCTING MAGNETS OR CRYOGENIC MAGNETS

A. Magnet Coils

- A. Conductor Components
 - 1. Normal magnets
 - Pure metals
 - Eddy current barrier
 - 2. Superconducting magnets
 - Superconductor
 - Pure metal stabilizer
 - Alloy for diffusion process
 - Diffusion barrier
 - Eddy current barrier
 - Strengthening material
 - Joining material
 - Metallic potting (cables)
- B. Electrical Insulation
 - 1. Turn to turn
 - 2. Coil to confining structure
- C. Bonding Agents
 - 1. Insulation to coil
 - 2. Insulation to confining structure
- D. Current Leads
- E. Spacers, Shims and Coolant Channels
- F. Coil Form and Confinement
 - 1. Structural material
 - 2. Joining materials (bolts, etc.)

B. Eddy Current Shields

C. Magnet Dewars

- A. Inner Wall
- B. Insulation
- C. Outer Wall
- D. Thermal Shields
 - 1. Within insulation space
 - 2. Nitrogen jacket

D. Support Structure

- A. Magnet to Dewar Inner Wall
- B. Dewar Inner Wall to Outer Wall
- C. TF Coil Central Support
- D. External Support Structure (warm)

MFE DEVICE ENVIRONMENTAL CONSIDERATIONS
IN THE MAGNET REGION

A. Stress

- A. As Constructed
 - 1. Gravitational
 - 2. Due to fabrication
 - 3. Due to construction
- B. Cooldown
- C. Operational
 - 1. Static
 - 2. Pulsed operation
 - 3. Abnormal conditions

B. Thermal

- A. Temperature
 - 1. Steady state temperature or gradient
 - 2. Excursions
 - Cooldown
 - Quench
 - Irradiation
 - Field induced
 - Current induced
- B. Heat Transfer
 - 1. To coolant
 - 2. Through electrical insulation (coil)
 - 3. Through thermal insulation (dewar)
 - 4. Through load-bearing structure

C. Electrical and Magnetic

(includes steady state, normal time dependent operating conditions
and off normal excursions)

- A. Voltage
- B. Current
- C. Field

D. Radiation

- A. Neutrons
- B. Other

OPERATIONS ON MFE MAGNET SYSTEM COMPONENTS
WHICH BEAR ON MATERIALS SELECTION

A. Fabrication

- A. Large Structures
 - 1. Joining
 - 2. Heat treatments
 - Base metal
 - Joints
 - 3. Machining
- B. Complex Structures
(including coolant channel material and basic superconductor)
- C. Magnet Winding
 - 1. Pretensioning
 - 2. Coating and cladding conductor
 - 3. Insulation and cooling channel insertion
 - 4. Joining conductor
 - 5. Curing epoxy materials

B. Transportation and Erection

C. Preoperation

- A. Vacuum Pumping and Leak Detection
- B. Instrumentation and NDE

D. Maintenance and Repair

- A. Remote Systems
- B. In Situ Repair