Distribution Category:
Magnetic Fusion Energy
(UC-420)

ANL/FPP/88-1

ANL/FPP--88-1

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DE88 012661

ITER SHIELD AND BLANKET WORK PACKAGE REPORT

U.S. ITER Nuclear Group

June 1988

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Work supported by
Office of Fusion Energy
U.S. Department of Energy
Under Contract W-31-109-ENG-38

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ITER SHIELD AND BLANKET WORK PACKAGE REPORT

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CHAPTER 1

SUMMARY

1.0 SUMMARY

1.1 INTRODUCTION

This report summarizes nuclear-related work in support of the U.S. effort for the International Thermonuclear Experimental Reactor (ITER) Study. The purpose of this work was to prepare for the first international ITER workshop devoted to defining a basic ITER concept that will serve as a basis for an in-depth conceptual design activity over the next 2-1/2 years. Primary tasks carried out during the past year included: design improvements of the inboard shield developed for the TIBER concept, scoping studies of a variety of tritium breeding blanket options, development of necessary design guidelines and evaluation criteria for the blanket options, further safety considerations related to nuclear components and issues regarding structural materials for an ITER device.

The design guidelines were developed for a range of possible ITER design parameters and included material design limits and documentation of material properties for the first wall, blanket and shield systems. The blanket evaluation criteria considered a broad-based assessment related to performance, economic, safety, environment and other programmatic factors.

The blanket concepts considered in this evaluation include: the aqueous/Li salt solution blanket developed for TIBER; a water-cooled, solid-breeder blanket; a helium-cooled, solid-breeder blanket; a blanket cooled by helium containing lithium-bearing particulates; and a blanket concept based on breeding tritium from He^3 .

Additional work on the inboard shield concentrated on optimizing the material composition of the inboard shield between stainless steel, tungsten, and other neutron moderators. In addition, the use of a thin beryllium region behind the inboard first wall was considered for enhancing tritium breeding in the outboard region for added neutron multiplication. Additional attention was given to ameliorating the effects of loss-of-coolant and loss-of-flow accidents.

1.2 RESULTS OF PRELIMINARY BLANKET/SHIELD EVALUATION

Several options for an ITER tritium-production blanket/shield were examined and are listed in Table 1-1. They were designed in a preliminary fashion under a common set of design guidelines and evaluated in terms of a common set of evaluation criteria described below.

As shown in Table 1-1, the options can be classified by coolant type (water and helium) and tritium breeding method, either separate from the coolant or contained in the coolant. These options provide for several possibilities to meet the needs of a permanent blanket/shield for ITER, each with their own set of particular issues - see Table 1-2 and the rest of this report for further details.

A preliminary assessment of the options in Table 1-1 was carried out using the evaluation criteria listed (as described below) in Table 1-3. A □ corresponds to the best and a ■ corresponds to the worst.* "Worst" here means in relative comparison to the other concepts. Symbols such as ◻ or ◼ reflect a split group opinion between □ and ◻ or between ◻ and ■.

All concepts were judged to be possible candidates for ITER. We have the most confidence in these judgments for the aqueous-salt option and water-cooled and helium-cooled solid breeder options because more work has been devoted to them so far.

Based on this preliminary and qualitative evaluation, one can make the following observations:

- There are several interesting candidates; no one option stands out as clearly superior to the others.
- The concepts are judged similar with respect to impact on reactor costs, R&D requirements and maintenance.
- The issues of accidental activated product release and decay heat response appear to be dominated by the inboard shield, which, in this preliminary evaluation, is taken to be similar for all blanket concepts.

* In subsequent chapters, □ = 3, ◻ = 2 and ■ = 1.

Table 1-1. U.S. ITER Blanket/Shield Options

<i>Form of Incorporating Breeder Material into Blanket</i>	<i>Type of Coolant</i>	
	<i>Water</i>	<i>Helium</i>
Separate from Coolant	Solid breeder with helium purge	Solid breeder with separate helium purge or *Separate ³ He purge
In Coolant	Li-containing salt in water (self-cooled aqueous salt)	*Li-containing particulates in helium

* Concept has recently been added to options under consideration

Table 1-2. Some Key Particular Issues for ITER Blanket/Shield Options*

• Self-cooled aqueous/salt	<ul style="list-style-type: none"> - stress corrosion cracking of austenitic steel by aqueous/salt mixture - cost effective tritium recovery from water - flow control in large tank configuration
• Water-cooled solid breeder	<ul style="list-style-type: none"> - aqueous stress corrosion cracking of austenitic steel - predictable gap conductance between breeder and clad, including irradiation effects - irradiation effects on solid breeder materials - ability to accommodate first wall neutron power loading variations
• Helium-cooled solid breeder	<ul style="list-style-type: none"> - predictable gap conductance between clad and breeder, including irradiation effects - irradiation effects on solid breeder materials - helium containment and manifolding
• Li particulates in helium	<ul style="list-style-type: none"> - helium containment and manifolding - erosion of helium ducts and helium pumps
• ^3He purge stream	<ul style="list-style-type: none"> - helium containment and manifolding - source of ^3He

* All designs employ austenitic steel as the structural material and beryllium as a neutron multiplier. They share common concerns such as low temperature irradiation effects on the mechanical properties of structural materials.

Table 1-3. Preliminary ITER Blanket & Shield Evaluation

<i>Blanket/Shield Option</i>	Aqueous Salt	H₂ O/SB	He/SB	Li Particulate	³He
1. Overall Reactor Cost
2. R&D Requirement
3. Design and Fabrication Complexity
4. Accidental Tritium Release
5. Accidental Activated Product Release
6. Routine Tritium Release
7. Compatibility with Phased Operation
8. Maintenance
9. Tritium Breeding
10. Chemical & Thermal Reaction Potential
11. Decay Heat Response
12. Tritium Extraction Technology
13. Waste Disposal Rating
14. Power Reactor Relevance

= best, = worst, = intermediate

- The aqueous-salt concept is simpler from a fabrication point of view when the coolant serves as the tritium breeding medium. The self-cooled options generally result in less design complexity.
- The helium-cooled options in general have safety and environmental advantages. The aqueous-salt has the most safety/environmental concerns, largely resulting from tritium control issues.
- The helium-cooled/solid breeder option appears to have the most reactor relevance.
- Helium cooling permits operating the structure at optimum temperatures.
- Based on preliminary analysis, the various blanket/shield options appear to have similar and sufficient tritium breeding capability. Some disagreement remains on this point, however, because there was not sufficient time to do a detailed 3-D neutronic analysis based on a common approach.*

1.3 DESIGN GUIDELINES

The main purpose of the design guidelines is to provide a common basis for the design process of the different first wall/blanket/shield options. Specifically, the value or range of main design parameters are established including reactor dimensions, neutron wall loading distribution, and operational schedules. Uniform guidance on design issues that impact the performance of various options are defined. For example, operating temperature limits for structural materials, first wall and tiles, solid-breeder material, beryllium neutron multiplier, and shield materials are established. Protection criteria for superconductor coils are also defined.

The other part of the design guidelines is a comprehensive data base required for the design process. It includes physical, thermal, and mechanical properties for different materials. Also, tritium transport properties and models for calculating steady-state tritium inventory in solid breeders and beryllium multipliers are given in detail.

* A 3-D analysis has been performed for the aqueous salt and helium-cooled, solid-breeder options.

1.4 EVALUATION CRITERIA

Evaluation criteria were established to assist in indicating the main areas of difference in a comparative evaluation of the permanent blanket/shield options for ITER. These criteria (presented below) cover several aspects of performance, economic and safety/environmental considerations. The basic approach is to first apply the design guidelines which serve as absolute requirements. The evaluation criteria then help to distinguish between the relative strengths and weaknesses of the various blanket/shield options.

Fourteen criteria (not all of equal weight) have been defined as follows:

1. Overall cost
2. R&D requirement
3. Design and fabrication complexity
4. Compatibility with phased operation
5. Maintenance
6. Tritium breeding
7. Tritium extraction
8. Accidental tritium release
9. Accidental activated product release
10. Routine tritium release
11. Chemical & thermal reaction potential
12. Decay heat response
13. Waste disposal rating
14. Power reactor relevance

Each criterion is given a general definition and a ranking system. The ranking system gives a 3, 2 or 1 (3 being best, 1 being worst) based on a somewhat more quantitative definition which measures a particular attribute of an evaluation criteria. The evaluation criteria and their ranking system are as follows.

1.4.1 Overall Cost

Overall cost should include all plant direct costs and the lifetime operating cost.

- **Ranking:** Lowest overall cost = 100%

100% - 105%	3
105% - 110%	2
> 110%	1

1.4.2 R&D Requirement

Qualitative assessment based on 1988 materials and technology data base and 1988 level of understanding of the critical technical issues.

- **Ranking:**

Low requirement (data could be available in the next three years with present facilities) with cost less than 1 M\$.	3
--	---

Intermediate requirement (data only available if new small-to-modest facilities are built within three years) with cost between 1 and 10 M\$.	2
---	---

High requirement (data will only be available from new major facilities no sooner than 1993) with cost greater than 10 M\$.	1
---	---

1.4.3 Design and Fabrication Complexity

Qualitative assessment based on the number of fluid systems, operating pressure, general design simplicity for accommodating penetrations, test modules and the divertor region, number and length of welds.

- **Ranking:**

Low complexity	3
Intermediate complexity	2
High complexity	1

1.4.4 Compatibility with Phased Operation

Required downtime to replace blankets between physics and technology phases.

- **Ranking:**

Ability to change from nonbreeding to breeding operation without removing the blanket/shield and produce no tritium during nonbreeding phase.	3
Ability to change from nonbreeding to breeding operation without removing the blanket/shield.	2
Must remove and replace blanket/shield structure to change from nonbreeding to breeding operation.	1

1.4.5 Maintenance

Qualitative assessment of time required to changeout of a single blanket module based on the weight, number of coolant connections and the complexity of disengagement from supporting structure.

- **Ranking:**

< 1 month	3
1 month - 3 months	2
> 3 months	1

1.4.6 Tritium Breeding

Net tritium breeding ratio (TBR) including possible breeding in inboard and divertor zones and excluding test modules.

- **Ranking:**

TBR > 1.1	3
TBR 1 - 1.1	2
TBR 0.9 - 1.0	1

1.4.7 Tritium Extraction

Qualitative assessment of tritium extraction issues such as continuous or batch process and demonstrated large scale extraction at relevant ITER conditions based on 1988 understanding of the technology.

- **Ranking:**

Currently demonstrable	3
Demonstrable within 2 - 5 y	2
Demonstrable > 5 y	1

1.4.8 Accidental Tritium Release

Defined as the maximum tritium released under the worst meteorological conditions as HTO.

- **Ranking:**

Maximum individual 50 y dose < 10 mrem	3
Maximum individual 50 y dose < 100 mrem	2
Maximum individual 50 y dose < 1 rem	1

1.4.9 Accidental Activated Product Release

Activated product release for a maximum credible accident at end of life (3 MWy/m²).

- **Ranking:**

Maximum individual prompt dose < 1 rem	3
Maximum individual prompt dose < 25 rem	2
Maximum individual prompt dose < 200 rem	1

1.4.10 Routine Tritium Release

Total estimated tritium release to the environment from the blanket system.

- **Ranking:**

Worst case tritium release rate < 10 Ci/d	3
Worst case tritium release rate between 10 and 25 Ci/d	2
Worst case tritium release rate between 25 and 40 Ci/d	1

1.4.11 Chemical and Thermal Reaction Potential

Calculated potential for production of explosive or flammable mixtures from interaction of blanket/shield materials with test module materials and divertor coolant.

- **Ranking:**

No explosive or flammable materials	3
Combustible material potentially present, but less than flammable limit if spread throughout building (< 4% H ₂)	2
Combustible material potentially present, but less than the explosive limit if spread throughout building (< 70% H ₂)	1

1.4.12 Decay Heat Response

Time after simultaneous LOCA/LOFA in both inboard and outboard blankets to reach half the melting temperature of any blanket/shield material.

- **Ranking:**

> 7 days	3
1 day - 7 days	2
< 24 hours	1

1.4.13 Waste Disposal Rating (WDR)

WDR of blanket/shield materials (averaged over blanket volume at end of life).

- **Ranking:**

WDR = Class A	3
WDR = Class C	2
Requiring deep burial	1

1.4.14 Power Reactor Relevance

Qualitative assessment of reactor relevance on the basis of extrapolation from ITER operating conditions to a power reactor.

- **Ranking:**

Uses prototypical materials and design concepts at power reactor parameters	3
Uses prototypical materials and design concepts at reduced parameters	2
Uses other materials and concepts at reduced parameters	1

1.5 SAFETY CONSIDERATIONS

The blanket will probably play the dominant role among design units in establishing the safety and environmental characteristics of ITER. The blanket will likely include the largest activation product inventory and the most mobile and vulnerable tritium inventory.

Basic material choices (coolant, tritium breeder, and structure) largely determine the safety and environmental features of the blanket. These choices (coupled with the ITER fluence) will set the range of several safety parameters, including the degree of waste management difficulty, activation product inventory, energy sources for release of radioactivity, and the form and amount of the tritium inventory. Once materials are selected, these safety parameters are generally difficult and expensive to alter. Thus, early attention to safety issues, before material selection, should optimize safety performance at minimum cost and with fewer add-on design fixes.

Overall, among the blanket concepts examined by the U.S. team, the helium-cooled concepts all appear more attractive from the safety standpoint than water-cooled concepts.

1.6 MATERIALS ASSESSMENT

1.6.1 Structural Materials

Austenitic stainless steel has been selected as the reference material in virtually every ITER-type design study and is proposed as the reference material for the ITER first wall/blanket structure. The ferritic/martensitic steels are not recommended for low temperature operation ($< 300^{\circ}\text{C}$), frequent shutdown operating scenario projected for ITER. Further analyses should be conducted to evaluate the potential/desirability of selected stainless steels (e.g., modified Type 304) and of manganese stabilized austenitic steels as low activation options.

Primary advantages of the austenitic steels are their ease of fabrication/welding and more extensive data base compared to other alloys. However, there remain several unresolved critical issues, some of which are feasibility issues for the ITER objectives. These include:

- Aqueous stress corrosion cracking
 - "pure" water
 - aqueous salts (LiNO_3)
- Low temperature ($< 300^{\circ}\text{C}$) fracture toughness after irradiation
- Mechanical properties of weldments after irradiation
- Choice of thermal-mechanical-treatment (e.g., cold work)
- Fabrication issues
- Feasibility of providing adequate data base for modified Ni-stabilized or Mn-steels in ITER time schedule.

The aqueous stress corrosion cracking (SCC) issue is a serious feasibility issue for water-cooled and aqueous-salt concepts. The SCC issue is particularly important since water is generally regarded as the only reasonable coolant for other components (e.g., heating and fueling systems, etc.) that will probably be constructed of austenitic steel and must operate in the same fusion environment. Several additional considerations make the

SCC concern even greater for the fusion environment than for fission (LWR) applications:

- low threshold fluence for irradiation assisted SCC (< 1 dpa)
- high H generation (H/dpa ~ 50) and low temperature operation (100-200°C)
- radiolysis and/or electrolysis
- high thermal stress in first wall (cyclic)
- extensive weld requirements

Radiation-induced swelling does not appear to be a serious issue for austenitic steels at temperatures below 350°C and fluences < 50 dpa. This applies to weldments as well as both solution-annealed and cold-worked material. However, recent irradiation experiments on austenitic steels indicate that the tensile elongation is significantly reduced. This indicates that loss of low temperature fracture toughness may be serious. Also, since the fracture toughness of weldments is generally lower than that of the base metal, weldment/joint integrity under irradiation is also a major concern.

The thermal-mechanical-treatment (TMT) strongly influences the mechanical properties of austenitic steels. For example, cold-worked steel provides substantial advantages over annealed material with respect to allowable design stress. However, welding may negate part of this advantage. The effects of welding on the tensile properties of steels should be determined.

The feasibility of providing a structural material that meets 10CFR61 Class C waste disposal criteria for ITER conditions should be examined. Special heats of standard Ni-stabilized austenitic steels, e.g., Type 304, may meet the current specifications if the trace concentrations of Nb and Mo are adequately controlled. The extensive data base on Type 304 makes this an attractive option.

The Mn-stabilized austenitic steels should also be evaluated as a "low activation" structural material option for ITER. Most properties and critical issues for this alloy system are similar to those listed above for the conventional austenitic stainless steels. Significant uncertainties exist regarding the difficulties and properties of weldments. Major questions relate to whether an adequate data base on Mn-steels can be provided for ITER and whether a major R&D effort is justified for an alloy system that does not appear attractive for commercial applications. The low activation benefit

must also be balanced with the poorer safety rating compared to most other alloy systems. It is recommended that existing commercial alloys be evaluated in order to take advantage of the existing data base. Optimization of this alloy class does not seem feasible for ITER and it may not be warranted.

1.6.2 Beryllium

Beryllium is chosen for use in the breeding blanket because of its neutron multiplication properties. As an example, the TIBER study called for 32 Tonnes of beryllium at an estimated cost of \$19M. The beryllium was in the form of 1.25 million, 3 cm dia. spheres, at a cost of \$15 each. Other designs call for a similar quantity but in the form of plates. ITER is expected to use more beryllium because it will probably be a larger machine than TIBER. There are several issues concerning the use of beryllium in ITER.

The standard method of making beryllium parts, by machining from stock made from hot pressed powder, leads to considerable cost and material wastage from complex machining. The \$19M estimated for TIBER may be too low. The method of first cold pressing and then sintering should result in little or no final machining and little material wastage and therefore lower cost. There are no radiation damage data on this type of beryllium. The mechanical properties should be determined after irradiation up to a fluence of 2×10^{22} n/cm² above 1 MeV which corresponds to $> 10,000$ appm He in beryllium and a first wall fluence of ~ 4 MW.y/m² in ITER.

Tritium buildup in neutron-irradiated beryllium is also an issue. Both ⁶Li, ⁷Li and ³He are produced and neutron capture leads to tritium. Tritium is nearly impermeable in beryllium at low temperatures. If for any reason the temperature of the irradiated beryllium rises to a significant fraction of the melting point of 1278°C such as in a loss-of-coolant accident, the accumulated tritium could suddenly be released. This issue can be much better understood by measuring the tritium release rate versus temperature of irradiated beryllium.

Beryllium has been used successfully, unclad, exposed directly to water in a neutron and gamma flux of fission reactors for decades. Some versions of the tritium-producing blanket for ITER have this combination with LiNO₃ added. The presence of the dissolved nitrate salt might enhance corrosion. In addition, there is the issue of the effect of the magnetic field which,

with a flow will cause a voltage to be set up, which would have a tendency to enhance corrosion. Experiments to mock up these conditions and measure the corrosion rate would clarify these two issues. If the corrosion rate is too high, the beryllium would need to be clad. However, cladding raises its own questions.

Current neutron transport models and the nuclear data do not very accurately predict the neutron multiplication of beryllium in thick assemblies. Thus, it may be desirable to make some measurements of beryllium neutron multiplication. An accuracy of 3% appears achievable and would help in reducing the issue of predicting the tritium breeding in ITER.

1.7 INBOARD SHIELD DESIGN

The inboard (i/b) shield is responsible for protecting the inner legs of the toroidal field (TF) magnets. The thickness of the shield is dependent on the radiation damage allowance for the magnet, the shielding materials used, and the level of neutron production in the reaction chamber. The philosophy of the ITER shield design is to meet both the magnet protection requirements and safety-related criteria. The latter has imposed some constraints on the material selection so that the reactor components will not be harmed by the shield decay heat during any abnormal operation.

The baseline design of ITER calls for a major radius of 4.04 m and provides a 0.55 m space for the i/b shield. The configuration of the ITER i/b shield is based on earlier work done on the compact TIBER-II reactor, in which an extensive optimization study was performed for a tungsten-based shield cooled with the $\text{H}_2\text{O}/\text{LiNO}_3$ aqueous solution. Although the W provides excellent magnet protection, its high specific decay heat, particularly in the front layers of the shield, caused some concern in case of LOCA/LOFA accidents, as indicated by the activation and safety analyses. This suggested the selection of different materials and/or the relocation of various components.

Upon varying the i/b shield thickness over the range .5 - .75 m, several runs of the TETRA systems code have indicated that the major radius changes by 1.5 cm for each centimeter change in the i/b shield and the direct cost increase is about 5 M\$ and 4 M\$ per centimeter increase in the W and steel shields, respectively.

Satisfying the 10^{19} n/cm² fast fluence limit for the magnet, the direct cost is essentially the same for the PCA and W shields, although the inboard shield is 0.1 m thicker for the PCA shield. For this fluence, the higher cost of the W shield is offset by the lower cost of the smaller size TF magnets. For low fluence limits, the use of W instead of steel is beneficial in terms of smaller TF magnets, and thus, considerably lower direct cost. For ITER-type devices, the penalty for employing an order of magnitude lower fluence limit is about 65 M\$ increase in the direct cost and 0.25 m increase in the major radius, for any type of shield.

In the base case of ITER where the i/b shield is 0.55 m thick, the all PCA shield resulted in radiation effects that exceed the design limits for the TF coils. In order to meet the limits without increasing the size of the machine, the decision was made to combine both PCA and W in the i/b shield so that the PCA could be placed in the front layer of the shield. This offers the advantage of lessening the severity of the decay heat problem without altering the dimensions of the device. It was also found out that several centimeters of Be behind the first wall of the i/b shield will increase the tritium breeding capability of the reactor by about 10%. Therefore, the i/b shield is configured in 3 main layers: 0.05 m Be layer, followed by 0.18 m PCA layer, then 0.18 m W layer. Four coolant channels, each is 0.01 m wide, are distributed across the shield to reduce the maximum temperatures within the various layers below the permissible values. About 0.1 m thick layer of Li-aqueous solution at the back of the shield was found necessary to minimize the damage in the magnet.

1.8 AQUEOUS-SALT SELF-COOLED BLANKET

The aqueous self-cooled blanket concept used in the TIBER-II design was analyzed for the U.S. version of a preliminary ITER design. This concept is based on dissolving small amounts of lithium compounds in the water coolant. This makes it possible to breed tritium in all the shield zones, and results in tritium self-sufficiency as well as enhanced magnet protection due to strong neutron absorption in lithium. This is of particular significance in a compact technology test reactor.

The options of using LiNO₃ or LiOH in the aqueous solution were assessed. The design that maximizes the outboard tritium breeding ratio

utilizes a 40 cm thick zone of Be balls followed by 80 cm thick zone of steel balls. The salt concentration is 16 g/100 cm³ for LiNO₃ and 5 g/100 cm³ for LiOH. The lithium is enriched to 90% ⁶Li. Single size balls are used to minimize the pressure drop. An optimum ball diameter of 3 cm was found to yield a minimum system pressure of 0.27 MPa. Using balls in the design makes it possible to accommodate the complicated ITER geometry with the numerous penetrations. The overall tritium breeding ratio (TBR) excluding tritium bred in the test modules is 1.1 and falls only to 1.05 if LiOH is used. The inboard and divertor zone shields contribute 0.3 to the overall TBR. The end of life peak dpa in the first wall is 50 dpa and the peak power density is 15 W/cm³.

The first wall and blanket are cooled with a single pass where the coolant flows from the bottom up. The inlet coolant temperature is 40°C and the temperature rise is 15°C in the first wall and 35°C in the blanket. The coolant pressure is 0.27 MPa with 0.13 MPa pressure drop. The coolant flow rate is 3430 kg/s and the pumping power is 0.43 MW. The maximum first wall thermal stress is 400 MPa. The inboard shield has the same pressure as the outboard shield but with a temperature rise of 20°C in the front five coolant channels and only 4°C in the back channel. The coolant flow rate in this inboard shield is 1990 kg/s and the pumping power is 0.24 MW. This design can easily operate at 50% higher heating conditions at a coolant pressure of 0.41 MPa.

Detailed activation analysis was performed for the outboard and inboard blanket and shield. The options of using PCA or the low activation austenitic steel Tenelon were assessed. The total activity in the blanket and shield at shutdown is 1560 MCi if Tenelon is used and 1130 MCi if PCA is used. The outboard blanket and shield qualifies as class C low level waste. If Tenelon is used it can even qualify as class A waste. The inboard shield by itself qualifies as class C waste only if Tenelon is used. If PCA is used exclusively, then class C qualification for the inboard and outboard shields may be realized if the two are disposed of together.

The total amount of ¹⁴C produced in the coolant is 4650 Ci for LiNO₃ and 54 Ci for LiOH. Since 20% of the coolant will be processed per full power day for tritium recovery, ¹⁴C can be removed at the same rate leading to a coolant specific activity in the reactor of only 0.05 Ci/m³ of ¹⁴C for class C

waste. The volume of operational waste to be disposed of annually is 50 m³ for LiNO₃ and 0.6 m³ for LiOH.

The significant reduction in the amount of W used in the optimized inboard shield results in large reduction in decay heat compared to the TIBER-II design. The integrated decay heat values in one day after shutdown in the outboard region are 68 and 97 GJ for PCA and Tenelon, respectively. The corresponding values for the inboard shield are 86 and 141 GJ. The decay heat data were used to determine the temperature response of the different blanket and shield materials following a LOCA. Under adiabatic heat up conditions the structure can withstand a LOCA, and after 1 day the temperatures do not exceed 700°C. Allowing natural circulation of the reactor cover gas can reduce these temperatures significantly (perhaps to the 500°C level). In the case of LOFA, natural coolant convection will exist everywhere to dissipate the decay heat.

Long-term corrosion and the radiolytic decomposition of the aqueous solutions in contact with the shield materials is a critical issue for this blanket concept. While there is no experimental evidence to substantiate this concern, a R&D program will provide data in the next three years. The technology required for tritium recovery has been demonstrated on a large scale by Ontario Hydro. The 400 g average tritium inventory in the reactor coolant loop is roughly the same as that found in current CANDU reactors. This inventory can be substantially reduced at little or no penalty in TBR by using the aqueous solution only in the front zones of the outboard blanket and inboard shield.

1.9 WATER COOLED, SOLID BREEDER BLANKET

A water-cooled, solid-breeder blanket concept has been developed for ITER. The blanket makes use of beryllium to multiply the fusion neutron for tritium production. In order to utilize these neutrons, the solid breeder has to contain a large concentration of lithium-6 isotope. A 90% lithium-6 enrichment is used with any of the solid breeders (Li₂O, LiAlO₂, or Li₄SiO₄). Also, the use of high lithium-6 enrichment reduces the solid-breeder volume required in the blanket. This reduction results in lower tritium inventory and blanket cost.

For satisfactory tritium inventory, the solid breeder temperature has to be within a specific temperature range. The minimum temperature for all solid breeders is about 350 to 400°C which does not match the desired operating temperature range for a low-pressure water coolant. The adopted design approach is to locate the beryllium multiplier between the solid breeder and the water coolant to raise the breeder temperature to the required temperature range without relying on an adjustable gap conductance between the solid breeder and the clad material. This design approach increases the blanket reliability and simplifies the design configuration. Also, the use of a beryllium zone between the water coolant and tritium breeder increases the number of tritium barriers which reduces the tritium permeation from the solid breeder to the water coolant. Carbon is used at the back of the blanket as a neutron moderator with a low absorption cross section. Also, it operates as a thermal insulator between the water coolant and the solid breeder.

A multilayer configuration is used to reduce fabrication costs and to simplify the blanket configuration. Two thin layers of solid breeder (1.8 cm total thickness) with three layers of beryllium (22.5 cm total) are used. The use of thin breeder layers permits an increase in the neutron wall loading up to 50% from the nominal value without exceeding the maximum allowable temperature of any material.

The one-dimensional tritium breeding ratio of this blanket is 1.68 with 22.5 MeV per neutron deposited in the blanket. The total tritium inventory in the solid breeder (Li_2O) and the beryllium multiplier is 1.7 g under optimum conditions. The blanket materials satisfy Class C for waste disposal at end-of-life.

The water-cooled, solid-breeder blanket developed for ITER utilizes current technology and data bases. However, several key issues require special attention to insure a satisfactory performance. The issues for this class of blankets are the following:

- Irradiation effects and lithium-6 burnup impact on the properties of the solid breeder materials are not completely known. The main concern is the change in the thermal conductivity of the solid breeders and the gap conductance at the breeder-to-structure interface.

- Irradiation data for austenitic steel at a low temperature at appropriate He to dpa ratio need to be determined. (Applies to all blanket options for ITER.)
- Aqueous stress corrosion cracking of austenitic steel in fusion environment needs to be assessed.
- The ability of the blanket to accommodate a large change in the neutron wall loading greater than 50% of the nominal value needs to be addressed. The main concern is that the breeder temperature will exceed the nominal temperature range.

1.10 HELIUM-COOLED, SOLID-BREEDER BLANKET

The attractiveness of a helium-cooled solid breeder blanket for ITER rests on several factors. The design uses main line reactor-relevant materials and configuration with the solid breeder operating at high temperature and the coolant operating at moderate temperature and pressure for safety and reliability reasons. Helium can be run so as to optimize the structure temperature. Large power variations can be accommodated by utilizing only a portion of the temperature window for the solid breeder and by adjusting the rate of flow and/or inlet temperature of the helium without incurring any substantial pressure penalty. Also, helium precludes any chemical reactivity problems, which is a major advantage since safety is expected to be a key factor for ITER.

The blanket configuration consists of a number of canisters positioned side-by-side in the poloidal direction. Each canister contains a bundle of rods lying in a toroidal axis with the main helium coolant flowing radially, perpendicular to the rod axis. The rods contain the Li_4SiO_4 solid breeder and Be multiplier and a low-conductance gap must be included to provide the temperature drop between the high temperature solid breeder and the lower temperature helium coolant. A key feature of the rod design is its robustness with regard to rod failure. Even if a rod was to fail completely, the worst consequence would probably be tritium contamination of the main flow. However, the tritium produced per rod is small and the consequence of several rod failures can be easily tolerated.

Comprehensive analyses were performed to assess various aspects of the design such as the achievable tritium breeding ratio (TBR) and the predictability of the gap conductance required to maintain the desired solid breeder operating temperature. Conservative 3-D neutronics calculations yield a 1.27 TBR for breeding in all regions except for a 4m high space-restricted inboard shield. Even by including space losses due to penetrations, test modules and canister end walls and plenums, we can comfortably design for a TBR of at least unity.

The power variation that can be allowed passively is linked with the predictability of the temperature profile between the solid breeder and helium coolant, and, in particular, of the thermal resistance gap temperature drop. A novel idea has been proposed for this gap. Mixing Be particles with He increases the gap conductance and, hence, its dimension and results in a more forgiving design. In this design, the rods consist of an inner Li_4SiO_4 cylinder surrounded by an annular Be/He region whose dimensions and material volume fractions can be varied to obtain the solid breeder/Be mix maximizing the TBR. The solid breeder helium purge flow can also be used to purge the clad Be/He region, thus providing a strong effective barrier against tritium permeation from the solid breeder to the main flow. Such a gap conductance concept, however, needs to be verified experimentally.

Irradiation effects on solid breeder properties is an issue, particularly with respect to the tritium inventory. However, the estimated tritium inventory is very small, ~ 1.5g which allows for a large margin of up to a factor of 100 without losing the safety advantages of this design. In addition, it is believed that the R&D requirements for ITER could be obtained with a modest acceleration of the current solid breeder base program.

Helium containment and manifolding are other issues. It is difficult to accurately predict the helium leakage to the plasma and further effort is required in this area to find out the extent of the problem, and, if required, to develop acceptable solutions. The difficulty associated with the large piping dimensions for helium can be minimized by optimizing the configuration arrangement and maximizing the use of the outboard region for breeding while excluding breeding in the space-limited inboard.

1.11 HE-COOLED, LI PARTICULATE BLANKET

This blanket concept uses a dilute suspension of fine solid breeder particles in a carrier gas as the coolant and the lithium breeding stream. It has the potential to be a simple blanket design for ITER, with excellent performance and low tritium inventory in the breeder. It can also be extrapolated to power reactors. Similar particle and gas mixture coolants were proposed for fission reactors in the 1960's.

The design uses Li_4SiO_4 or other solid breeder materials as the breeder particles which have been an average size in the range of 2-10 microns. The helium gas at 20 to 30 atmospheres pressure carries the dilute particles and circulates around the blanket and heat exchanger loop. The solid breeder concentration in the helium stream is in the range of 1 to 5 volume percent. In order to produce an adequate amount of tritium, beryllium is used as the neutron multiplier. At a ^6Li enrichment of 90%, the 1-D poloidal model tritium breeding ratio is 2.03. The total thickness of the helium stream is 4 cm out of 50 cm total blanket thickness. The blanket module configuration is a lobe design, with coolant flowing in the poloidal directions from the bottom to the top of the reactor. At coolant inlet and outlet temperatures of 50 and 200°C, respectively, the maximum structural temperature at the first wall is 417°C. The bred tritium can be extracted by diverting one percent of the coolant to the tritium processing system. The estimated tritium inventory is less than 1 gm in the helium stream.

This concept has the capability of switching from non-breeding to breeding tritium without hardware changes in the reactor. Also, the blanket performance can be adjusted during operation by changing the helium stream characteristics (speed, pressure, breeder content, ^6Li enrichment, or inlet temperature).

However, there are few design issues which require further analyses. Structure erosion by the solid breeder particles is the main concern. The selection of a flow velocity less than 10 m/s and a particle size of less than 40 microns eliminates the concern about the 316 steel erosion. Attention must be given to the other components in the helium loop where the velocity may exceed the above threshold. The current design uses a blower and multistate cyclone loop design to avoid the erosion problem. Another design issue is the sticking of fine particles on cold surfaces.

1.12 HELIUM-3 BLANKETS

In this approach, terrestrial supplies of helium-3, rather than lithium, are used for tritium breeding. In order to assess the potential of the He-3 blanket, a reference configuration was adopted based on minor modifications to the helium-cooled blanket concepts considered in the Blanket Comparison and Selection Study. The chosen configuration assumes a stainless steel for structure and cladding and beryllium for neutron multiplication. In order to minimize He-3 inventories, as well as He-3 and tritium leakage rates, the He-3 is placed separate from the main helium (He-4) coolant. It flows through the beryllium in a manner similar to the helium purge through a solid breeder. This arrangement provides good tritium-breeding and also allows for removal of tritium bred in the beryllium. The He-3 circulates only fast enough to recover the tritium. Both the breeding and cooling loops operate at about 5 MPa. For 600 MWth, and an inlet temperature of 80C, a helium coolant flow rate of 510 kg/s (130 kmol/s) gives a temperature rise of 225C across the blanket.

The tritium breeding potential of the He-3 blanket configuration was assessed using a one-dimensional discrete ordinates transport code. The results indicate that tritium breeding ratios above 1.5 are attainable when the He-3 volume is about 6% of the total blanket volume.

For a 600 MW (fusion) machine with a tritium breeding ratio of unity, the tritium production rate of the blanket is 91 g T/d, which must be recovered from the He-3 carrier gas. Tritium loss from the breeder region across the cladding, into the main coolant and then across the heat exchanger was estimated. Due to the low temperatures of this blanket, a 10 Ci/d loss rate across the heat exchanger is easily met without assuming oxide barriers. Direct loss of tritium through leaks at valves and flanges will be very small. For the coolant and breeder gas these losses are estimated to be 160 Ci/yr and 10 Ci/yr, respectively.

The total tritium inventory in the breeder gas is 0.06 g, and in the main coolant, 0.9 g. In the event of an accidental release of this entire inventory, the maximum offsite dose would be only 50 mrem. The total tritium dissolved in beryllium is about 0.5 g. The 1% of tritium directly produced in the beryllium leads to 45 g T inventory for a 1 cm thick Be layer, but up to 1

kg T by the end of ITER operation if a BeO layer forms over much of the beryllium and inhibits diffusion.

The total He-3 circuit volume is about 15 m³, requiring 50 kg He-3. We assume that the He-3 breeder circuit has a 1%/yr loss rate due to its small size and assuming welded, leak-tight construction. This corresponds to a He-3 loss rate of about 0.5 kg/yr. The yearly burnup of He-3 is ~ 8 kg, assuming a 25% availability.

The He-3 blanket offers an attractive option for tritium breeding in ITER. The concept exhibits good tritium breeding potential, low tritium inventories, and low tritium leakage rates. Moreover, the concept retains the desirable safety and operational features inherent in gas-cooled blankets, including full compatibility with all reactor test blankets and the ability to operate the first wall at high temperatures for outgassing. There is no need to mechanically access the blanket in order to control breeding - the same blanket hardware can provide shielding or breeding controlled only by out-of-core addition of He-3. Finally, although the concept is not extrapolatable to power reactors (assuming we are limited to terrestrial supplies of He-3), it will provide reactor relevant information on helium-cooled blankets.

While the technical features of the He-3 blanket option are attractive, the ability to guarantee the required He-3 resources (50 kg inventory and 8.5 kg/yr makeup) emerges as the major issue associated with the viability of this concept. Tritium decay in nuclear weapons is the only potential source of He-3 which could satisfy ITER needs.

CHAPTER 2

DESIGN GUIDELINES AND EVALUATION CRITERIA

2.0

DESIGN GUIDELINES AND EVALUATION CRITERIA

2.1 DESIGN GUIDELINES

The main purpose of the design guidelines is to provide a common basis for the design process of the different first wall/blanket/shield options. Specifically, the value or range of main design parameters are established, including reactor dimensions, neutron wall loading distribution, and expected operational schedules. A uniform guidance on design issues that impact the design performance of various options is defined. For example, operating temperature limits for structural materials, first wall tile, solid breeder materials, carbon reflector, beryllium neutron multiplier, and shield materials are established. Protection criteria for superconductor coils, inventory concentration of tritium in the main coolant, and dose equivalent limits are also defined.

A comprehensive materials data base is also presented in an appropriate format for the design process. It includes physical, thermal, and mechanical properties for different materials. Also, tritium transport properties and models for calculating steady-state tritium inventory in solid breeders and beryllium multiplier are given in detail. The next subsections give these data in a condensed form.

2.1.1 Main Reactor Parameters

The main reactor parameters with direct relevance to the first wall/blanket/shield are given in this section.

Plasma Fusion Power, MW	631
Plasma Major Radius, m	4.036
Plasma Minor Radius, m	1.41
Plasma Elongation (κ)	2.0
Average Neutron Wall Loading, MW/m ²	1.224
Average First Wall Heat Flux ⁽¹⁾ , MW/m ²	0.215
Total First Wall Area, m ²	412.86
Test Section Surface Area, m ²	19.20
Divertor Surface Area Per Null ⁽²⁾ , m ²	40.78
Plasma Heating Surface Area, m ²	~10.0

Inboard First Wall Area, m ²	113.82
Outboard First Wall Area, m ²	217.52
Average Inboard Neutron Wall Loading, MW/m ²	0.845
Maximum Inboard Neutron Wall Loading, MW/m ²	1.473
Minimum Inboard Neutron Wall Loading, MW/m ²	0.249
Average Outboard Neutron Wall Loading, MW/m ²	1.498
Maximum Outboard Neutron Wall Loading, MW/m ²	1.910
Minimum Outboard Neutron Wall Loading, MW/m ²	0.789
Nominal Total Inboard Tile First Wall/Blanket/Shield Thickness, m	0.57
Inboard Tile Thickness, m	0.02
Total Number of Pulses	
Pulse Inductive Ignition Operation	5 x 10 ⁵
Steady-state Current Drive Operation	5 x 10 ⁴
Burn Pulse Length, s	
Pulse Inductive Ignition Operation	200
Steady-state Current-drive Operation	2000
First Wall Disruption Parameters (Preliminary)	
Pulse Inductive Ignition Operation	
Total Number of Disruptions	300
Magnetic Energy Released to First Wall, MJ	160
Current Quench Time, ms	16.4
Kinetic Energy Released to First Wall, ms	260
Thermal Quench Time, ms	3
Peak Energy Density, MJ/m ²	3.8
Steady-State Current Drive Operation	
Total Number of Disruptions	30
Magnetic Energy Released to First Wall, MJ	142
Current Quench Time, ms	15.5
Kinetic Energy Released to First Wall, MJ	360
Thermal Quench Time, ms	3
Peak Energy Density, MJ/m ²	4.6
Maintenance Concept	Hor./Vert. Access
First Wall/Blanket/Shield L/R Time Constant, ms	10
On Axis Magnetic Field, T	4.98

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- (1) First wall heat flux distribution is assumed to be the same as neutron wall loading distribution.
 - (2) The design guideline is based on a double-null divertor. The most recent design uses a single null divertor which changes the values of the first wall area listed above.

2.1.2 Design Limits

Design limits were established to satisfy the ITER mission and insure satisfactory performance for the different components. The different limits (with a short clarification note if it is needed) are listed below:

First Wall/Blanket/Shield Life, MW.y/m ²	
Average	3.0
Peak	4.7
Total Outboard Tile or First Wall Erosion, mm	<1
Maximum Load Structural Temperature During Normal Operation for Type 316 Stainless Steel ⁽¹⁾ , °C	350
Maximum Load Structural Temperature During Off-normal Conditions for Type 316 Stainless Steel ⁽²⁾ , °C	750
Allowable Stress Intensity ⁽³⁾ , S _m , for Type 316 Stainless Steel at 350°C, MPa	
Annealed	91
Cold worked	205
Structural Temperature for Type 316 Stainless Steel at the Interface with Other Materials Based on Compatibility Issues, °C	
H ₂ O ⁽⁴⁾	<150 or >250
H ₂ O/LiNO ₃ /LiOH ⁽⁴⁾	<120
Be ⁽⁵⁾	<380
Li ₂ O ⁽⁶⁾	<600
LiAlO ₂ ⁽⁶⁾	<750
Li ₄ SiO ₄ ⁽⁶⁾	<750
Operating Temperature Range for Solid Breeders During Steady-state Operation, °C	
Li ₂ O ⁽⁷⁾	400 to 800
LiAlO ₂ ⁽⁸⁾	400 to 1000
Li ₄ SiO ₄ ⁽⁹⁾	350 to 850
Be (Pure) ⁽¹⁰⁾	≤600
Be (1% BeO) ⁽¹⁰⁾	350 to 600
Maximum Operating Temperature for Shielding Material, °C	
W ⁽¹¹⁾	2673
B ₄ C ⁽¹²⁾	1905
TiH ₂ ⁽¹²⁾	300
Pb ⁽¹¹⁾	209
Steel ⁽¹¹⁾	1045

Radiation Dose Limits

Maximum total operating personnel doses under normal conditions, rem/y-person	1
Maximum dose level during operation and after shutdown in radiation zones for normal personnel operation (no maintenance), mrem/h	0.5
Maximum dose level one day after shutdown for planned personnel access, mrem/h	2.5

Outboard Shield Design Criterion

Complete remote maintenance operation	
No consideration for the radiation dose inside the reactor hall after shutdown	
Use the available space between the first wall and the outboard TF coils for shielding	
Insure no significant increase in the TF coil nuclear heating	
Remote maintenance operation with personnel access to the reactor hall	
Maximum dose level one day after shutdown with all shields in place, mrem/h	2.5

Tritium

A net tritium breeding ratio close to unity is the design goal	
Tritium concentration in the water coolant ⁽¹³⁾ , Ci/l	
Average over time	10
Peak	30
Maximum tritium loss from First Wall/Blanket/Shield, Ci/D	40
Maximum total tritium loss, Ci/D ⁽¹⁴⁾	50
Maximum tritium inventory in the water coolant (to satisfy 1 rem off-site accidental prompt dose assuming 2 - 10% release), g	1000 - 200

Toroidal Field Coils Design Limits

Maximum insulator dose ⁽¹⁵⁾ , rads	
Polyimides	10^{10}
Epoxy-type G10-CR or G11-CR	2×10^9
Maximum insulator dose for polyimides, assuming only a static compression load, rads	4×10^{11}
Maximum fast neutron fluence ($E > 0.1$ MeV) in the Nb ₃ Sn superconductor, ⁽¹⁶⁾ n/cm ²	10^{19}
Maximum total nuclear heating ⁽¹⁷⁾ kW	<72

Maximum total copper stabilizer resistance including
radiation contribution⁽¹⁸⁾, $\Omega\text{-cm}$

3×10^{-7}

-
- (1) Swelling consideration defines this limit.
 - (2) Structure deformation at this temperature is the limiting factor.
 - (3) Use $1.5 S_m$ for primary membrane plus bending stress intensity; and $3 S_m$ for primary membrane, plus bending, plus secondary stress intensity. Use Von Mises effective stress on the stress intensity for multiaxial stress.
 - (4) Stress corrosion cracking is the basis for this limit.
 - (5) This temperature limits the interaction layer (assumed to have no strength from a structural analysis viewpoint) to 0.1 mm. At 550°C, the interaction layer increases to 0.25 mm.
 - (6) This temperature limits the interaction layer to <0.1 mm.
 - (7) For purge and interconnected porosity partial pressure of 34 Pa for HTO, LiOT will precipitate out as a separate phase for $T \leq 400^\circ\text{C}$. For $T \leq 350^\circ\text{C}$, LiOT will precipitate out as a separate phase at 5 Pa HTO partial pressure. In order to avoid excessive inventory of T in the form of separate phase LiOT and in order to provide some flexibility in purge flow design, it is recommended that $T_{\min} = 400^\circ\text{C}$ and $P_{\text{H}_2\text{O}} + P_{\text{HTO}} < 34 \text{ Pa}$ for Li_2O . The upper limit is based on thermal stability and mass transport concerns. Thermal anneal tests at 900°C for 8 hours on unirradiated Li_2O show that this material (which probably has some moisture in the form of LiOH) experiences grain growth (6 μm to 80 μm), weight loss (1.34%) and densification (2.4% volume decrease) in a static environment. Based on the FUBR-1A irradiation test, grain growth (6 μm to 34 μm) was observed in closed-capsule Li_2O samples with calculated beginning of life temperatures of 850 - 885°C. Mass transfer from the Li_2O to the cladding walls was also observed. However, any possible densification of as-fabricated porosity was masked by helium induced swelling and cracking which led to a net volumetric swelling of about 6%. Purge flow tests (in a non-nuclear environment) showed that Li_2O exposed to He with 1 ppm H_2O and 1 ppm H_2 experienced mass loss rates of 9 $\text{mg/m}^2\cdot\text{h}$ at 550°C and 63 $\text{mg/m}^2\cdot\text{h}$ at 650°C. Two possible problems associated with such mass transport are loss of Li from the breeder and constricting of the purge stream due to LiOT condensation in low temperature regions. The recommended temperature limit of 800°C for Li_2O is a compromise between the acceptable performance of Li_2O in the temperature range of 850 - 885°C in closed-capsule, in-reactor tests and the problematic behavior of Li_2O at 650°C in purged-capsule, out-of-reactor tests.
 - (8) The diffusive inventory of tritium becomes large as the minimum temperature of LiAlO_2 decreases below 400°C. The maximum temperature limit is set to avoid excessive in-reactor sintering of LiAlO_2 .

- (9) The limit of $T_{\min} = 350^{\circ}\text{C}$ is somewhat arbitrary. It assumes that Li_4SiO_4 is thermodynamically like Li_2O which has a limit of 400°C , but not as hygroscopic. The upper limit is based on thermal stability.
- (10) Hydrogen isotope retention in Be appears to be high for deuterium-bombarded Be which contains $\sim 1\%$ BeO at $T < 350^{\circ}\text{C}$. If it is desired to retain tritium in beryllium, then it is recommended that Be (+ 1% BeO) have a $T_{\max} = 300^{\circ}\text{C}$ and no $T_{\min}$. If on-line tritium recovery from the Be is desired, then it is recommended that $T_{\min} = 350^{\circ}\text{C}$ and/or the Be be kept relatively free from oxygen. The maximum temperature for Be is based on swelling and corrosion concerns. He-induced swelling increases dramatically from $\sim 1 \text{ vol}\%$ at 740°C to $> 10\%$ at 800°C . Corrosion (which could seriously impact tritium recovery) due to the presence of oxygen, moisture, or CO_2 becomes significant for $T > 600^{\circ}\text{C}$. Thus, a $T_{\max} = 600^{\circ}\text{C}$ seems reasonable to avoid excessive swelling and oxidation of the Be.
- (11) The maximum temperature is taken to be 0.8 the melting point.
- (12) The dissociation of TiH_2 and B_4C is the basis for the temperature limit.
- (13) The maximum off-site dose due to 100% release of vulnerable tritium and the tritium system cost are the basis for these limits.
- (14) The off-site dose to the public is the basis for this tritium loss.
- (15) A 25% decrease in the mechanical properties is the basis for this limit.
- (16) No change in the critical current density and 0.9 the original value of the critical temperature are the basis for this limit.
- (17) Heat removal considerations are the basis for this limit.
- (18) The conductor design requires this limit.

2.1.3 Design Data Base

Solid Breeders and Neutron Multipliers

Theoretical density (ρ) and ^6Li number density (N_0) vs fractional ^6Li enrichment (δ)

Li_2O

$$\begin{aligned}\rho &= 2.0338 (1 - 0.06665\delta), \text{ g/cm}^3 \\ N_0 &= 8.1581 \times 10^{22} \delta, \text{ atoms/cm}^3\end{aligned}$$

Li₄SiO₄

$$\rho = 2.3993 (1 - 0.03332\delta), \text{ g/cm}^3$$

$$N_O = 4.8112 \times 10^{22} \delta, \text{ atoms/cm}^3$$

 γ -LiAlO₂ (Tetragonal)

$$\rho = 2.6171 (1 - 0.01516\delta), \text{ g/cm}^3$$

$$N_O = 2.3885 \times 10^{22} \delta, \text{ atoms/cm}^3$$

Be

$$\rho = 1.85 \text{ g/cm}^3$$

$$N_O = 1.238 \times 10^{23}, \text{ atoms/cm}^3$$

Melting and/or Phase Change Temperature

<u>Material</u>	<u>Phase Change Temp. °C</u>	<u>Melting Temp. °C</u>
Li ₂ O	--	1432 ± 6
Li ₄ SiO ₄	655 α + γ	1255 (Decomposes)
LiAlO ₂	--	1610
Be	--	1283

Latent Heat of Melting

<u>Material</u>	<u>Latent Heat, J/g</u>
Li ₂ O	1954
Li ₄ SiO ₄	----
LiAlO ₂	1332
Be	1303

Thermal Conductivity**Li₂O**

$$k = f_p [71.4 (1 + 1.306 \times 10^{-2} T)^{-1}], \text{ W/m-K}$$

$$f_p = (1-P) [1 + (2.08 - 6.6 \times 10^{-4} T)P]^{-1}$$

Li₄SiO₄

$$k = f_p [6.34(1 + 1.44 \times 10^{-3} T)^{-1}], \text{ W/m-K}$$

$$f_p = (1-P) [1 + (2.08 - 6.6 \times 10^{-4} T)P]^{-1}$$

LiAlO₂

$$k = f_p [20.6(1 + 6.42 \times 10^{-3} T)^{-1}], \text{ W/m-K,}$$

$$f_p = (1-P) [1 + (2.08 - 6.6 \times 10^{-4} T)P]^{-1}, \text{ and}$$

Be

$$k = f_p [279(1 - 1.344 \times 10^{-3} T + 7.168 \times 10^{-7} T^2)], \text{ W/m-K}$$

$$f_p = (1 - P)/(1 + 11 P^2)$$

where: T is temperature in K and
P is porosity volume fraction

Sphere-Pac Thermal Conductivity

The above correlations assume that the solid breeder is a continuous phase with interspersed porosity. This is what one would get with a sintered product. If the breeder smear density were achieved with a sphere-pac breeder, then a further reduction in effective conductivity would be realized because of the resistance at particle/particle interfaces. Following the approach adopted by Tam and Liu, the data for UO₂ sphere-pac (86.6% packing fraction with 98% dense particles of three particle sizes) thermal conductivity are compared to the data for sintered UO₂ of the same smear density. Let f_{sp} be the ratio of thermal conductivity of sphere-pac material to sintered material of the same density. In the temperature range of interest ($T > 600$ K) f_{sp} is sensitive to helium pressure and relatively insensitive to temperature. Recommended values for f_{sp} are 0.49 at 0.101 MPa (1 atm), 0.60 at 0.203 MPa (2 atm) and 0.76 at 0.608 MPa (6 atm). Interpolation may be used for intermediate pressures. For pressures above 0.608 MPa, $f_{sp} = 0.76$ is recommended as the improvement of thermal conductivity with pressure tends to saturate. Extrapolation below 0.101 MPa should be exercised with caution and probably limited to 0.05 MPa because of high uncertainties associated with interface heat transfer as the gas fill becomes more rarified.

Thermal Expansion Coefficient (Instantaneous)

$$\text{Li}_2\text{O}: 10^6 \alpha = 20.75 + 1.72 \times 10^{-2}T$$

$$\text{Li}_4\text{SiO}_4: 10^6 \alpha = 18.80 + 1.66 \times 10^{-2}T$$

$$\text{LiAlO}_2: 10^6 \alpha = 9.66 + 4.61 \times 10^{-3}T$$

$$\text{Be}: 10^6 \alpha = 11.09 + 1.983 \times 10^{-2}T - 8.928 \times 10^{-6}T^2$$

where: α = Coefficient of thermal expansion ($^{\circ}\text{C}^{-1}$).

T = Temperature, $20 < T < 950$, $^{\circ}\text{C}$.

Mean Thermal Expansion Coefficient (α_m)

$$\alpha_m = \frac{\Delta L/L_0}{T - T_r} = \frac{1}{T - T_r} \int_{T_r}^T \alpha dT$$

where: α is the instantaneous thermal expansion coefficient, T_r is a reference temperature (e.g., 298K), and T is temperature in K.

$$\text{Li}_2\text{O}: \alpha_m = 18.7(1 + 4.49 \times 10^{-4}T), \mu\text{m/m-K}$$

$$\text{Li}_4\text{SiO}_4: \alpha_m = 16.7(1 + 4.96 \times 10^{-4}T), \mu\text{m/m-K}$$

$$\text{LiAlO}_2: \alpha_m = 8.98(1 + 2.76 \times 10^{-4}T), \mu\text{m/m-K}$$

$$\text{Be}: \alpha_m = 8.43(1 + 1.36 \times 10^{-3}T - 3.53 \times 10^{-7}T^2), \mu\text{m/m-K}$$

Specific Heat

$$\text{Li}_2\text{O}: C_p = 2.5179 \times 10^3 + 0.3328T - 8.3820 \times 10^7 T^{-2} \quad (306 < T < 1073)$$

$$\text{Li}_4\text{SiO}_4: C_p = 9.3994 \times 10^2 + 1.4577T + 4.0110 \times 10^6 T^{-2} \quad (300 < T < 973)$$

$$\text{LiAlO}_2: C_p = 1.0467 \times 10^3 + 0.4019T - 1.8003 \times 10^7 T^{-2} \quad (300 < T < 973)$$

$$\text{Be: } C_p = 2.432 \times 10^3 + 0.6378T - 7.11 \times 10^7 T^{-2} \quad (300 < T < 1553)$$

where: C_p = Specific heat, J/kg-K
 T = Temperature, K

Recommended Properties and Models for Calculating Steady-State Tritium Inventory in Breeders and Multiplier

Total Inventory

$$I = I_d + I_s + I_{sol}, g$$

where: I , I_d , I_s , and I_{sol} are the total, diffusive, surface, and solubility tritium inventories, respectively, in g.

Diffusive Inventory

$$I_d = (15 D_o)^{-1} \int_{V_b} \dot{g} r_g^2 \exp(Q_d/RT) dV$$

where: $\dot{g}$ = local tritium generation rate in g/cm³-s
 r_g = grain radius in cm
 V_b = total volume of the breeder (or multiplier)
 D_o = pre-exponential for diffusion coefficient in cm²/s
 Q_d = effective activation energy for diffusion in kJ/mol
 R = ideal gas constant = 8.314×10^{-3} kJ/mol-K

and

T = local temperature in K

Approximate formulas for diffusive inventory which are good for quick estimations of tritium inventory for rod and plate geometries are:

Plate (thickness ℓ)

$$I_d = (\dot{G} \overline{r_g^2} / 15 D_o) \int_{-1/2}^{+1/2} \exp[Q_d/RT(n)] dn$$

where: $\dot{G}$ = total tritium generation rate in blanket breeder (or multiplier)

$\overline{r_g^2}$ = volume-averaged grain-radius-squared in cm^2

$T(\eta)$ = temperature distribution in plate

η = normalized position (x/L)

Rod (radius r_0)

$$I_d = (\dot{G} \overline{r_g^2} / 15 D_0) \int_0^1 \exp [Q_d / RT(\eta)] d\eta^2 ,$$

where: η = normalized radial position (r/r_0)

<u>Material</u>	<u>$D_0, \text{cm}^2/\text{s}$</u>	<u>$Q_d, \text{kJ/mol}$</u>	<u>Data Base</u>
			<u>Temp. Range, °C</u>
Li_2O	6.81×10^{-3}	84.8	350 - 927
Li_4SiO_4	1.37×10^{-7}	63.0	500 - 630
LiAlO_2	8.90×10^{-3}	198.0	400 - 900
Be	3.00×10^{-7}	18.5	400 - 900

Surface Inventory

Diffusion, molecular combination and decomposition, desorption, and adsorption all play a role in how much tritium resides on free surfaces. In addition to the obvious variables of free surface area, tritium generation rate, and temperature, other variables which are important are the initial hydroxide (OH) impurity levels on the surfaces and the chemistry of the purge gas (e.g. partial pressures of H_2O , HTO , H_2 , HT , etc.).

No model or set of rate constants are currently available for this problem in its full generality. The following approach is recommended as interim guidelines until more data and better models become available. For the case of near vacuum ($\ll 1$ Pa) levels of moisture in the purge, a "desorption" inventory is defined based on thermal decomposition of LiOT ($\text{LiOT} + \text{LiOH} \rightarrow \text{Li}_2\text{O} + \text{HTO}$). Further, it is assumed that there is sufficient LiOH present on the breeder surfaces, so that the reaction rate kinetics are first order in tritium. For the case of moisture partial pressures $\gtrsim 1$ Pa, an "adsorption" inventory is defined.

Surface Desorption Inventory (I_{ds})

For the case of surfaces with a large concentration of LiOH compared to

LiOT and very small partial pressures of HTO and H₂O in the gas phase, the desorption coefficient will be pseudo first order in tritium with a desorption rate constant of

$$k_{ds} = k_o \exp (-Q_{ds}/RT), s^{-1}$$

If the reaction rate for OH is second order, the reaction rate for OT can be pseudo first order if ample moles of OH are present at the surface. Let K_{20} (in 1/mol-s) be the pre-exponential for the second-order kinetics reaction. Then, k_o can be derived from K_{20} to be $k_o = K_{20} M_{OH}$ where M_{OH} is the number of moles of OH on the surfaces.

For $M_{OH} \gg M_{OT}$ on the surfaces, the following formula may be used for the desorption inventory.

$$I_{ds} = (1/k_o) \int_{V_b} \dot{g} \exp (Q_{ds}/RT) dV \quad (2-1)$$

Simplified versions of Eq. 2-1 for plate and rod geometries with 1-D temperature profiles are:

Plate (thickness λ)

$$I_{ds} = (\dot{G}/k_o) \int_{-1/2}^{+1/2} \exp [Q_{ds}/RT(\eta)] d\eta \quad (2-1a)$$

where η = normalized position (x/λ)

Rod (outer radius r_o)

$$I_{ds} = (\dot{G}/k_o) \int_0^1 \exp [Q_{ds}/RT(\eta)] d\eta^2 \quad (2-1b)$$

where η = normalized radius (r/r_o)

<u>Material</u>	<u>k_o</u>	<u>Q_{ds}, kJ/mol</u>	<u>Basis</u>
Li ₂ O		2.8 x 10 ⁷	125
Li ₄ SiO ₄		1.0 x 10 ⁷	125
LiAlO ₂		1.6 x 10 ⁴ M _{OH}	62
Be		--	--
			Kudo data Estimate Fischer's Data Data Unavailable

Surface Adsorption Inventory I_{as}

As the partial pressure of HTO builds up in the interconnected porosity and the purge, the surface inventory will increase. The formula for calculating I_{as} is

$$I_{as} = A \int_{V_b} \exp (Q_a/RT) (P_{HTO}/P_{H_2O})^{1/2} a_{fs} \rho_b dV \quad , \quad (2-2)$$

where: a_{fs} = the specific free surface area, cm²/g
 ρ_b = the breeder (or multiplier) density, g/cm³
 Q_a = effective activation energy, kJ/mol
 P_{HTO} = local HTO partial pressure, Pa
 P_{H_2O} = local H₂O partial pressure, Pa
 A = constant, g/cm²-Pa^{1/2}

Simplified formulas for I_{as} for plate and rod geometries and 1-D temperature profiles are:

Plate (thickness ℓ)

$$I_{as} = A (P_{HTO}/P_{H_2O})^{1/2} \bar{a}_{fs} M_b \int_{-1/2}^{1/2} \exp [Q_a/RT(\eta)] d\eta \quad , \quad (2-2a)$$

where: η = normalized position (x/ ℓ)
 M_b = mass of the breeder, g

Rod (outer radius r_o)

$$I_{as} = A (P_{HTO}/P_{H_2O})^{1/2} \bar{a}_{fs} M_b \int_0^1 \exp [Q_a/RT(\eta)] d\eta^2 \quad (2-2b)$$

where: η = normalized position (r/ r_o)
 M_b = mass of breeder, g

<u>Material</u>	<u>A, g/cm² · Pa^{1/2}</u>	<u>Q_a, kJ/mol</u>	<u>Basis</u>
Li ₂ O	---	---	Should be small compared to solubility in bulk
Li ₄ SiO ₄	---	---	Should be small compared to solubility in bulk
LiAlO ₂	4.0 × 10 ⁻²⁰	139	Fischer's data for 773 and 873°K*
Be	---	---	Data unavailable

* If these parameters are used for T < 773K, then results will be an upper-bound as surface adsorption decreases from 773 K to 673 K for LiAlO₂.

Solubility

The solubility of hydroxide in Li₂O and LiAlO₂ has been measured. The results are generally expressed in the following form:

$$\log X_{OH} = -A + B \log (P_{H_2O}) ,$$

where X_{OH} is the mole fraction of OH in the solid, A and B are temperature dependent parameters and P_{H_2O} is the partial pressure of H₂O in the gas phase. This result can be generalized to the case of HTO and H₂O in the gas phase. The mole fraction of X_{OT} dissolved in the solid is

$$X_{OT} = 0.5 \exp (-2.303 A) P_{H_2O}^B \{ [1 + P_{HTO}/P_{H_2O}]^B - 1 \}. \quad (2-3a)$$

For $P_{HTO}/P_{H_2O} \ll 1$, the following approximation is reasonable

$$X_{OT} = 0.5 B \exp (-2.303 A) P_{HTO}/P_{H_2O}^{1-B} . \quad (2-3b)$$

Let S_T be the solubility of tritium (really OT) in wppm of breeder (or multiplier). Then, S_T can be expressed as

$$S_T = (3.016 \times 10^6 / W_b) X_{OT}, \text{ wppm}, \quad (2-4)$$

where W_b is the molecular weight of the breeder (or multiplier).

The solubility inventory (in g) of the whole blanket can be found by integrating S_T over the whole blanket volume:

$$I_{sol} = 10^6 \int_{V_b} S_T \rho_b dV, \quad (2-5)$$

where S_T is a function of partial pressures (P_{HTO} and P_{H_2O}) and temperature which may vary from volume element (dV) to volume element and ρ_b is the density of the breeder (or multiplier).

Simplified versions of Eq. 2-5 for plate and rod geometries and 1-D profiles are:

Plate (thickness ϵ)

$$I_{sol} = 10^{-6} M_b \int_{-\frac{\epsilon}{2}}^{\frac{\epsilon}{2}} S_T dn, g,$$

where M_b is the total mass of the breeder (or multiplier) and n is the normalized position (x/ϵ).

Rod (outer radius r_o)

$$I_{sol} = 10^{-6} M_b \int_0^1 S_T dn^2, g,$$

where n is the normalized radial position (r/r_o).

The restrictions on these equations are that the partial pressures of HTO and H_2O should be less than the limiting partial pressure for precipitating out the separate phases ($LiOT$ and $LiOH$) and that I_{sol} be less than the tritium generated.

Li_2O

Measurements have been made for X_{OH} in Li_2O . Using this data, the following expression has been derived for the solubility of tritium in Li_2O in the presence of gas phase H_2O and HTO:

$$S_T = \exp (-34.784 + 5.566 \times 10^{-2}T - 2.215 \times 10^{-5}T^2) P_{H_2O}^B \cdot [(1 + P_{HTO}/P_{H_2O})^B - 1], \text{ wppm}, \quad (2-6a)$$

where $B = (0.427 + 1.700 \times 10^{-4}T)$.

For the case that $P_{HTO}/P_{H_2O} \ll 1$, Eq. 2.6a reduces to

$$S_T = \exp (-34.784 + 5.566 \times 10^{-2}T - 2.215 \times 10^{-5}T^2) \cdot B P_{HTO}/P_{H_2O}^{1-B}, \text{ wppm} \quad (2-6b)$$

The restriction on the use of Eqs. 2-6a and 2-6b is that

$$P_{H_2O} + P_{HTO} < \begin{cases} (3.49 \times 10^{11} \text{ Pa}) \exp (-129 \text{ kJ} \cdot \text{mol}^{-1}/RT) & \text{for } T \leq 744 \text{ K} \\ (1.89 \times 10^8 \text{ Pa}) \exp (-84 \text{ kJ} \cdot \text{mol}^{-1}/RT) & \text{for } T > 744 \text{ K} \end{cases}$$

Li₄SiO₄

No data are currently available for calculating the solubility of OH in Li₄SiO₄. For estimation purposes, it is recommended that some fraction (e.g. 1/3) of the OH solubility in Li₂O be used.

LiAlO₂

Fischer has measured the solubility of OH in LiAlO₂ at 400°C and 500°C. Using his data and an approximate fit to his A and B coefficients, the following expression can be derived:

$$S_T = \exp (2.884 - 5.343 \times 10^{-3}T) P_{H_2O}^B [(1 + P_{HTO}/P_{H_2O})^B - 1], \text{ wppm}, \quad (2-7a)$$

where $B = (-0.274 + 1.0 \times 10^{-3}T)$.

For the special case in which $P_{\text{HTO}}/P_{\text{H}_2\text{O}} \ll 1$, the approximate formula is:

$$S_T = \exp (2.884 - 5.343 \times 10^{-3}T) B (P_{\text{HTO}}/P_{\text{H}_2\text{O}})^{1-B}, \text{ wppm} \quad (2-7b)$$

Be

Jones and Gibson have measured the solubility of H in Be as a function of H partial pressure. The effects of oxygen activity are not discussed. Allowing for P_{H_2} and P_{HT} in the gas phase gives

$$S_T = 3.3 \times 10^{-3} P_{\text{H}_2}^{1/2} [(1 + P_{\text{HT}}/P_{\text{H}_2})^{1/2} - 1], \text{ wppm} \quad (2-8a)$$

for the general expression and

$$S_T = 1.7 \times 10^{-3} P_{\text{HT}}/P_{\text{H}_2}^{1/2}, \text{ wppm}, \quad (2-8b)$$

for the particular case in which $P_{\text{HT}}/P_{\text{H}_2} \ll 1$.

Structural Material

Annealed Type 316 Stainless Steel Structural Material

Density

$$\rho = a + bT$$

where: ρ = density, lb/ft³
 T = temperature, °F (100-1500°F)
 a = 4.985886×10^2
 b = -1.537035×10^{-2}

Thermal expansion (mean)

$$\alpha_m = 11.397 \times 0.71828^{0.9989T}$$

where: α_m = mean thermal expansion coefficient, $10^{-6} \text{ } ^\circ\text{F}^{-1}$
 T = temperature, °F (100-1600°F)

Thermal conductivity

$$K = a + bT + cT^2 + dT^3$$

where: K = thermal conductivity, Btu/h ft°F
 T = temperature, °F
 a = 7.336869
 b = 5.567335×10^{-3}
 c = -4.511739×10^{-7}
 d = $-1.905863 \times 10^{-11}$

Thermal expansion (instantaneous)

$$\alpha = a + bT + cT^2 + dT^3$$

where: α = instantaneous thermal linear expansion coefficient, 10^{-6} F^{-1}
 T = temperature, °F (70 to 1600°F)
 a = 8.09139
 b = 5.496968×10^{-3}
 c = 2.679852×10^{-6}
 d = 4.95473×10^{-10}

Specific heat

$$C_p = a + bT + cT^2 + dT^3$$

where: C_p = specific heat, Btu/lb -°F
 T = temperature, °F
 a = 1.071939×10^{-1}
 b = 6.034521×10^{-5}
 c = -4.397679×10^{-8}
 d = 1.438572×10^{-11}

Thermal diffusivity

$$\alpha = a + bT + cT^2 + dT^3$$

where: α = thermal diffusivity, ft²/h
 T = temperature, °F
 a = 1.310488×10^{-1}
 b = 4.447860×10^{-5}
 c = 1.509035×10^{-8}
 d = $-7.068003 \times 10^{-12}$

Irradiation induced swelling

$$\frac{\Delta V}{V_0} = R \left[\phi t + \frac{1}{\alpha} \ln \left\{ \frac{1 + \exp [\alpha(\tau - \phi t)]}{1 + \exp (\alpha \tau)} \right\} \right]$$

where: $\frac{\Delta V}{V_0}$ = fractional volume change
 ϕt = fluence, 1 unit = 10^{22} n/cm² ($E > 0.1$ MeV)
 R = $0.01 \exp (-49.8592 + 0.195283T - 1.87409 \times 10^{-4}T^2)$
 α = $-1.117 + 6.889 \times 10^{-3}T$
 τ = $1.0 / (7.98769 - 2.98448 \times 10^{-2}T + 2.87279 \times 10^{-5}T^2)$
 T = temperature, °C

Poisson's ratio

$$\nu = a + bT$$

where: ν = Poisson's ratio
 T = temperature, °F ($100 < T < 1500$)
 a = 0.2624811
 b = 4.265787×10^{-5}

Young's modulus

$$E = a + bT + cT^2 + dT^3$$

where: E = Young's modulus, Mpsi °F
 T = temperature, °F ($100 < T < 1500$)
 a = 0.2833669
 b = -2.882211×10^{-3}
 c = -3.697849×10^{-6}
 d = 7.709188×10^{-10}

Shear Modulus

$$G = a + bT + cT^2 + dT^3$$

where: G = shear modulus, Mpsi
 T = temperature, °F ($100 < T < 1500$)
 a = 11.24760
 b = -1.642750×10^{-3}
 c = -1.330810×10^{-6}
 d = 3.567476×10^{-10}

Melting temperature of type 316 stainless steel, °C	1375
Latent heat of melting type 316 stainless steel, J/g	268

2.2 EVALUATION CRITERIA

Evaluation criteria were established to assist in a comprehensive and objective evaluation of the permanent blanket/shield options for ITER. These criteria (presented below) cover several aspects of performance, economic and safety/environmental considerations. The basic approach is to first apply the design guidelines which serve as absolute requirements. The evaluation criteria then help to distinguish between the relative strengths and weaknesses of the various blanket/shield options.

Fourteen criteria have been defined as follows:

1. Overall cost
2. R&D requirement
3. Design and fabrication complexity
4. Compatibility with phased operation
5. Maintenance

6. Tritium breeding
7. Tritium extraction
8. Accidental tritium release
9. Accidental activated product release
10. Routine tritium release
11. Chemical & thermal reaction potential
12. Decay heat response
13. Waste disposal rating
14. Power reactor relevance

Each criteria is given a general definition and a ranking system. The ranking system gives a 3, 2 or 1 (3 being best, 1 being worst) based on a somewhat more quantitative definition which measures a particular attribute of an evaluation criteria. The evaluation criteria and their ranking system are as follows.

2.2.1 Overall Cost

Overall cost should include all plant direct costs and the lifetime operating cost.

- **Ranking:** Lowest overall cost = 100%

100% - 105%	3
105% - 110%	2
> 110%	1

2.2.2 R&D Requirement

Qualitative assessment based on 1988 materials and technology data base and 1988 level of understanding of the critical technical issues.

- **Ranking:**

Low requirement (data could be available in the next three years with present facilities) with cost less than 1 M\$.	3
--	---

Intermediate requirement (data only available 2
if new small-to-modest facilities are built
within three years) with cost between 1 and 10 M\$.

High requirement (data will only be available from 1
new major facilities no sooner than 1993) with cost
greater than 10 M\$.

2.2.3 Design and Fabrication Complexity

Qualitative assessment based on the number of fluid systems, operating pressure, general design simplicity for accommodating penetrations, test modules and the divertor region, number and length of welds.

- **Ranking:**

Low complexity	3
Intermediate complexity	2
High complexity	1

2.2.4 Compatibility with Phased Operation

Required downtime to replace blankets between physics and technology phases.

- **Ranking:**

Ability to change from nonbreeding to breeding operation without removing the blanket/shield and produce no tritium during nonbreeding phase.	3
Ability to change from nonbreeding to breeding operation without removing the blanket/shield.	2
Must remove and replace blanket/shield structure to change from nonbreeding to breeding operation.	1

2.2.5 Maintenance

Qualitative assessment of time required to changeout of a single blanket module based on the weight, number of coolant connections and the complexity of disengagement from supporting structure.

- **Ranking:**

< 1 month	3
1 month - 3 months	2
> 3 months	1

2.2.6 Tritium Breeding

Net tritium breeding ratio (TBR) including possible breeding in inboard and divertor zones and excluding test modules.

- **Ranking:**

TBR > 1.1	3
TBR 1 - 1.1	2
TBR 0.9 - 1.0	1

2.2.7 Tritium Extraction

Qualitative assessment of tritium extraction issues such as continuous or batch process and demonstrated large scale extraction at relevant ITER conditions based on 1988 understanding of the technology.

- **Ranking:**

Currently demonstrable	3
Demonstrable within 2 - 5 y	2
Demonstrable > 5 y	1

2.2.8 Accidental Tritium Release

Defined as the maximum tritium released under the worst meteorological conditions as HTO.

- **Ranking:**

Maximum individual 50 y dose < 10 mrem	3
Maximum individual 50 y dose < 100 mrem	2
Maximum individual 50 y dose < 1 rem	1

2.2.9 Accidental Activated Product Release

Activated product release for a maximum credible accident at end of life (3 MWy/m²).

- **Ranking:**

Maximum individual prompt dose < 1 rem	3
Maximum individual prompt dose < 25 rem	2
Maximum individual prompt dose < 200 rem	1

2.2.10 Routine Tritium Release

Total estimated tritium release to the environment from the blanket system.

- **Ranking:**

Worst case tritium release rate < 10 Ci/d	3
Worst case tritium release rate between 10 and 25 Ci/d	2
Worst case tritium release rate between 25 and 40 Ci/d	1

2.2.11 Chemical and Thermal Reaction Potential

Calculate potential for production of explosive or flammable mixtures from interaction of blanket/shield materials with test module materials and divertor coolant.

- **Ranking:**

No explosive or flammable materials	3
Combustible material potentially present, but less than flammable limit if spread throughout building ($< 4\% \text{ H}_2$)	2
Combustible material potentially present, but less than the explosive limit if spread throughout building ($< 70\% \text{ H}_2$)	1

2.2.12 Decay Heat Response

Time after simultaneous LOCA/LOFA in both inboard and outboard blankets to reach half the melting temperature of any blanket/shield material.

- **Ranking:**

> 7 days	3
1 day - 7 days	2
< 24 hours	1

2.2.13 Waste Disposal Rating (WDR)

WDR of blanket/shield materials (averaged over blanket volume at end of life).

Ranking:

WDR = Class A	3
WDR = Class C	2
Requiring deep burial	1

2.2.14 Power Reactor Relevance

Qualitative assessment of reactor relevance on the basis of extrapolation from ITER operating conditions to a power reactor.

- **Ranking:**

Uses prototypical materials and design concepts at power reactor parameters	3
Uses prototypical materials and design concepts at reduced parameters	2
Uses other materials and concepts at reduced parameters	1

CHAPTER 3

SAFETY AND ENVIRONMENTAL CONSIDERATIONS

3.0

SAFETY AND ENVIRONMENTAL CONSIDERATIONS

This chapter summarizes current safety and environmental results and unresolved issues. The blanket will include the largest activation product inventory and the most mobile and vulnerable tritium inventory - hence the blanket will likely dominate ITER's overall safety and environmental performance. Material choices largely determine safety and environmental features. With neutron fluence, these choices will fix the range of key safety factors, including the form and amount of tritium inventory, activation product inventory, energy sources for releasing radioactivity, and difficulty of waste management. Once materials are selected, these factors are generally difficult and costly to alter. Attention to safety and environmental issues before material selection should optimize safety performance at minimum cost and with fewer add-on design fixes.

Among the blanket concepts studied by the U.S. team, the helium-cooled concepts appear significantly more attractive from the safety viewpoint than the water-cooled concepts. The least attractive concept is the aqueous salt. The major difference is tritium control. (See sec. 3.1.)

The U.S. blanket team has focused on outboard blanket concepts with relatively less attention to the inboard side. Only the aqueous salt design team has emphasized inboard design. An aqueous salt inboard option has been found that appears approximately optimized between decay heat (desire minimum tungsten) and shielding (desire maximum tungsten) (Sec. 3.3).

All the concepts include austenitic steel structure. The trade-off between nickel-stabilized and manganese-stabilized steels is complex. Development of an ITER manganese steel would produce a steel composition qualifying as low level waste (LLW) using U.S. criteria. Several uncertainties are unresolved concerning the value of ITER manganese steels (Sec. 3.4).

All the blanket concepts use beryllium as the neutron multiplier. Beryllium appears preferred over lead from the safety perspective. Beryllium chemical toxicity is manageable with routine technology. One unresolved issue is tritium produced within the beryllium and subsequent inventory buildup (potential accident source term) or routine release (potential tritium effluent). The tritium produced in beryllium could be as high as 1 kg. (See sec. 3.5.)

3.1 SAFETY ISSUES IN BLANKET CONCEPT SELECTION

Compared with traditional blankets, the aqueous salt concept is newer. The dominant safety concerns are (a) tritium inventory and control, (b) C-14 production from the reference nitrate salt version, and (c) possible need for hydrogen gas overpressure to control radiolysis. Tritium control for this concept is inherently the most challenging because of producing tritium in the most hazardous and mobile form - water. Given enough money, adequate tritium control is possible. However, uncertainties must be resolved in the cost of tritium control - tritium removal from water and the building atmosphere and the leak-tightness of the coolant system and confinement building. At 10 Ci/liter-water, the average tritium inventory in the coolant will be 400 g (for 400-m³ coolant system) with a peak of 1.2 kg at a peak concentration of 30 Ci/liter. Unless coolant system resin beds are changed prematurely, the C-14 content will exceed U.S. LLW criteria; thus, ITER would be producing non-LLW during operation. The best understanding of U.S. LLW criteria is that premature resin bed change-out would not be allowed since it would be an attempt to get around the letter of the regulation. 1988 Canadian tests should clarify the radiolysis issue.

When designed with only one loop, the aqueous salt is the simplest blanket concept, likely leading to higher reliability and lower failure rates. If adequate tritium control could be demonstrated at acceptable cost, the combination of good tritium performance and lower failure rates would be attractive. However, the concept would still suffer from general water coolant safety issues, see sec. 3.2.

The safety issues with the water/solid breeder (SB) concept are well known, especially tritium inventory prediction and potential releases during operation or accidents. Three candidate solid breeders are Li₂O, LiAlO₂, and Li₄SiO₄. Of these, the water/Li₂O case involves the mild water-Li₂O reaction. Whereas this reaction is unlikely to be a significant public safety concern, it could threaten the plant or reduce availability. For example, water leakage from coolant system to Li₂O would produce a corrosive mixture.

The safety issues for the helium/SB concept are also well known, especially predicting the tritium inventory and minimizing releases during operation or accidents. This concept has no chemical reaction safety issues.

The He^4/He^3 concept is new and innovative. Although it has received less attention, the basic material choices make it attractive from the safety perspective, without any special safety issues. Tritium control and removal from helium streams must be demonstrated on a large scale to reduce technical risk for this concept. However, because of different chemistry, this may be different (easier?) than tritium removal from helium purge streams through solid breeders. Hydrogen is a likely additive to solid breeder purge streams to facilitate tritium release from the solid breeder. This would not be necessary for the He^3 concept, and oxygen could be added to form good oxide barriers to lower tritium permeation through structures.

The He/SB-particulate concept is also new and innovative. The basic materials are the same as the regular He/SB concept. Safety issues are to (a) insure that particulate erosion of structure is acceptable, (b) demonstrate tritium control and removal from helium streams on a large scale, and (c) insure adequate control of radioactive particulate contamination during operation and maintenance. Several uncertainties in predicting the tritium inventory in solid breeders are irrelevant for this concept, e.g. proper temperature control of the solid breeder.

3.2 POTENTIAL BLANKET INTERACTIONS WITH OTHER COMPONENTS

The ITER test program includes test modules for blanket concepts attractive for eventual commercial fusion. Two of these are elemental lithium and $^{17}\text{Li}^{83}\text{Pb}$.⁽¹⁾ Both react with water. Per volume, the lower lithium content in $^{17}\text{Li}^{83}\text{Pb}$ results in $^{17}\text{Li}^{83}\text{Pb}$ -water reactions releasing 1/8th the energy and hydrogen gas as lithium-water reactions.⁽²⁾ However, $^{17}\text{Li}^{83}\text{Pb}$ will become 10^2 - 10^3 times more radioactive. Limited testing with both materials appear possible in a water-cooled ITER. Replacing water cooling with helium (totally or partially) would increase the amount of lithium and $^{17}\text{Li}^{83}\text{Pb}$ allowed in test modules, thereby increasing the value of the test program. See the U.S. Test Program Briefing Paper.

The plasma-side of plasma chamber components will be quite hot. A primary candidate for first wall protective tiles is graphite, which has an endothermic reaction with water, producing hydrogen gas. Simple calculations for TIBER-II⁽³⁾ suggest that a significant amount of hydrogen gas could be

produced in the event of a blanket water leak spraying water on hot graphite. Even without chemical reactions, cold water on hot surfaces could thermally shock those surfaces. A leak is expected during the ITER lifetime.

There are apparently some innovative divertor concepts involving liquid metal surfaces. Some candidate liquid metals may be reactive with water. Additional design details and analyses are required to determine the severity of each of the above issues.

3.3 SAFETY CONSIDERATIONS IN SELECTING INBOARD BLANKET/SHIELD COMPOSITION

Tungsten is an effective inboard shield material. However, tungsten also produces high levels of decay heat following plasma shutdown, sufficient to pose a threat during loss-of-coolant accidents. The current U.S. inboard shield concept uses steel structure, tungsten for better shield effectiveness, beryllium for neutron multiplication, and aqueous salt coolant.

The current design has half the tungsten of the TIBER-II design, about 190 mm of 550 mm total. There is now 50 mm of beryllium; the rest is steel and salt solution. The decay heat and temperature rise during loss-of-coolant accidents appear acceptable. When other inboard coolants - regular water (not salt solution) or helium - are considered, the inboard shield will be redesigned and decay heat calculations will have to be repeated.

3.4 DECAY HEAT AND WASTE MANAGEMENT CONSIDERATIONS IN SELECTING STEEL TYPE

Subject to feasibility and cost, it would be advisable if the blanket materials could be disposed as LLW, using near-surface land burial. The austenitic phase in steels can be stabilized by either nickel or manganese. Nickel austenitic steels, e.g. PCA and 316SS, are not LLW when used as the first wall of a commercial fusion plant. Manganese austenitic steels could qualify as LLW by substitution of manganese for nickel, substitution of tungsten or vanadium for molybdenum, and reduction in niobium impurity.

Although neither type of austenitic steel has good overall performance for commercial fusion,⁽¹⁾ both are possible ITER candidates. Several safety and environmental issues are relevant in comparing these steel types. First, manganese steels produce about 1.5 times more decay heat over the critical

first day after plasma shutdown, making loss-of-coolant accidents more of a concern. The consequences of higher decay heat cannot be quantified without more detailed design and transient analyses.

Second, versus commercial fusion, ITER would have an order of magnitude lower fluence. Properly selected nickel-based steels might qualify as LLW. The key elements are molybdenum and niobium, not nickel. That is, the research and development expense of manganese steels, and their higher decay heat, might not be needed for ITER to meet LLW.

Third, there is no international consensus on waste management LLW criteria. Meeting U.S. LLW criteria may not be relevant for ITER; some other countries' LLW criteria are significantly more restrictive and possibly neither nickel nor manganese steels would qualify as LLW. Until an ITER policy on international criteria is adopted, one cannot assess the motivating force for developing manganese steels. In summary, one cannot currently make a strong case for or against manganese steels for ITER.

3.5 SAFETY CONSIDERATIONS IN NEUTRON MULTIPLIER SELECTION

To obtain a tritium breeding ratio near 1.0 in ITER, some neutron multiplier, either beryllium or lead, appears needed. Lead could be used as elemental lead, $^{17}\text{Li}^{83}\text{Pb}$, or PbBi . The key benefits of beryllium are to (a) produce more tritium, reducing tritium procurement costs, (b) avoid the high activation hazards of lead, and (c) provide an additional heat sink during loss-of-coolant accidents. Besides high cost, the key safety and environmental hazards of beryllium are (a) fabrication and construction hazard due to chemical toxicity, (b) potential accident source term (chemical toxicity and tritium inventory), and (c) waste management.

Overall, beryllium chemical hazards are adequately known, and techniques are routinely used to control those hazards. Standard economic calculations should adequately internalize appropriate penalties for safety and environmental hazards of beryllium, provided (a) the cost of installed beryllium reflects the difficulties in mining, refining, and fabricating beryllium, (b) the cost of beryllium waste management reflects hazards in handling and specific limits on beryllium transport (about $13,000 \text{ } \$/\text{m}^3$ for beryllium low-level waste versus $1,000 \text{ } \$/\text{m}^3$ for generic low-level waste), and (c) the design

uses beryllium such that there are no special accident pathways that could mobilize beryllium dust. Beryllium balls or bulk pieces in the blanket may not be subject to beryllium dust production; however, beryllium as first wall coating likely would. Future activities should include (a) experiments to clarify the inventory and behavior of tritium in beryllium, in ITER-relevant forms, and (b) examination of candidate designs to insure that no significant pathway exists for mobilizing beryllium dust.

REFERENCES

(More information is in Appendix 5 of the U.S. Safety Briefing Paper.)

1. D. L. Smith, et al., "Blanket Comparison and Selection Study - Final Report," ANL/FPP-84-1, September 1984; especially section 5.4.2.
2. S. J. Piet, et al., "Liquid Metal Chemical Reaction Safety in Fusion Facilities," Fusion Engineering and Design, 5, 1987, pp. 273-298.
3. J. D. Lee, et al., "TIBER-II/ETR Final Design Report," UCID-21150, September 1987; especially chapter 5.

CHAPTER 4

STRUCTURAL MATERIALS ASSESSMENT FOR ITER FIRST WALL/BLANKET

4.0

STRUCTURAL MATERIALS ASSESSMENT FOR ITER FIRST WALL/BLANKET

Austenitic stainless steel has been selected as the reference material in virtually every ETR-type design study and is proposed as the reference material for the ITER first wall/blanket structure. The ferritic/martensitic steels are not recommended for the low temperature ($< 300^{\circ}\text{C}$), frequent shutdown operating scenario projected for ITER. Further analyses should be conducted to evaluate the potential for structural material options that meet "low activation" waste disposal criteria. Special heats of conventional stainless steel, e.g., Type 304, and selected manganese-stabilized austenitic steels appear acceptable for the ITER operating scenario.

4.1 CONVENTIONAL AUSTENITIC STEELS (316, PCA)

Primary advantages of the conventional austenitic steels are their ease of fabrication/welding and more extensive data base compared to other alloys. However, there remain several unresolved critical issues, some of which are feasibility issues for the ITER objectives. These include:

- Aqueous stress corrosion cracking
 - "pure" water
 - aqueous salts (LiNO_3)
- Low temperature ($< 300^{\circ}\text{C}$) fracture toughness after irradiation
- Mechanical properties of weldments after irradiation
- Choice of cold-worked versus solution annealed material
- Fabrication issues of wall components
- Feasibility of providing adequate data base for modified Ni-stabilized or Mn-stabilized steels in the ITER time schedule.

4.1.1 Aqueous Stress Corrosion Cracking

Aqueous stress corrosion cracking (SCC) of austenitic steels has been/is a serious problem in water-cooled fission reactors (LWR's) and has been identified as a serious feasibility issue for all water-stainless steel systems proposed for fusion applications. Extensive study in recent years indicates that virtually all austenitic steels are susceptible to SCC under

certain conditions. These studies have shown that the following three factors strongly influence SCC:

- water chemistry (O_2 , H_2O_2 , etc.),
- steel composition and microstructure, and
- stress state.

Recent information has also indicated an additional effect due to irradiation (IASCC), which occurs at neutron fluences in excess of $\sim 10^{21} \text{ cm}^{-2}$ at $\sim 300^\circ\text{C}$ (corresponding to $\sim 0.5 \text{ dpa}$ or about two weeks of continuous operation in the ITER first wall). Several additional considerations that make the SCC concern much greater for the fusion environment compared to the LWR applications include:

- low threshold for IASCC ($\sim 0.5 \text{ dpa}$ at 300°C),
- high hydrogen transmutation rate ($H/\text{dpa} \sim 50$) with low temperature ($< 200^\circ\text{C}$) operation,
- radiolysis at high radiation levels,
- electrolysis of flowing liquid in the magnetic field if electrical conductivity is significant,
- high thermal stresses that must be accommodated, particularly in the first wall,
- extensive weld requirements.

Certain design features, e.g., crevices, weldments, and dissimilar metal systems are particularly sensitive to SCC and should be avoided. The existing data base is insufficient to conclude whether either "pure" water-cooled concepts or the aqueous salt concepts are viable. The low operating temperature and low pressure coolant provide some advantages. Also, flexibility in steel composition should be beneficial. In general, SCC in the aqueous salts is expected to be more severe because of the more complex chemistry and the known effects of small additions of anions to "pure" water.

In determining priorities for R&D, one should consider that water is generally regarded as the preferred coolant for other reactor systems, e.g., heating and fueling systems, that may be constructed of austenitic steel and must operate in the same fusion environment. One should also consider

modifications in composition and thermomechanical treatment to select specific austenitic steels that are more resistant to SCC. For example, some information indicates that low carbon steels are more resistant to SCC. Also, highly sensitized steels are generally more susceptible to SCC.

4.1.2 Radiation Effects

Radiation induced swelling is not perceived to be a serious issue for austenitic steels at temperatures below 350°C and fluences < 50 dpa. This applies to weldments as well as both solution annealed and cold-worked material. However, recent irradiation experiments on austenitic steels indicate that the tensile elongation at temperatures below ~ 350°C is significantly reduced (uniform elongation < 1%) even at modest fluences (~ 15 dpa).

Since the fracture toughness typically correlates with tensile ductility, loss of low temperature fracture toughness is a major concern. Also, since the fracture toughness of weldments (including heat affected zone [HAZ]) is generally lower than that of the base metal, weldment/joint integrity under irradiation is also a major concern.

4.1.3 Fabrication, Joining, and TMT

The thermal-mechanical-treatment (TMT) strongly influences the mechanical properties of austenitic steels. For example, the yield stress of 20% cold-worked 316 SS is over three times higher than the yield stress for solution annealed material at temperatures < 400°C. Therefore, cold-worked material provides significant advantages with regard to allowable design stress. However, concerns have been raised with respect to annealing of the cold-worked structure in regions near weldments. Further work is required to define design stress limits for weldments in cold-worked material. A proposed design solution is to avoid weldments in maximum stress regions. Similarly, certain joining processes such as hot isostatic pressing (HIP) and high temperature brazing will tend to anneal cold-worked structures. Either these fabrication methods must be avoided or design allowable stresses must be determined if cold-worked material is to be used.

4.2 LOW ACTIVATION AUSTENITIC STEELS

Development of "low activation" materials is an important goal of the current fusion materials program. A key question is whether acceptable low activation materials can be qualified in a time frame consistent with the ITER schedule. Two possible options should be considered for ITER:

- Conventional Ni-stabilized austenitic steels (e.g., Type 304) with controlled Mo and Nb concentrations.
- Mn-stabilized austenitic steels with low N and low Ni concentrations.

4.2.1 Ni-Stabilized Austenitic Steels

There is currently considerable uncertainty as to the allowable nickel concentration in a steel structure that will meet the waste management guidelines for Class C disposal under 10 CFR 61. The allowable limits depend on total fluence, neutron spectrum, and method of averaging or compaction of total material. It appears that for the limited lifetime neutron fluences projected for ITER, conventional Ni-stabilized steels, (e.g., Type 304 with 8-10% Ni) may be acceptable if the molybdenum and niobium concentrations are adequately controlled.

It should be no more difficult to control these trace impurities in Type 304 than in other austenitic alloys. The primary advantage of a modified Type 304 or a similar alloy is the extensive data base available. In general, the key issues are the same as those discussed above for the conventional steels. Detailed analyses of proposed blanket concepts should be conducted to determine whether this class of alloy will meet the desired waste management guidelines. If so, the possibility of providing an adequate data base in the ITER time schedule is more favorable for this class of alloys than it is for the Mn-steels.

4.2.2 Mn-Stabilized Austenitic Steels

The Mn-stabilized austenitic steels should be evaluated as a "low activation" structural material option for ITER. Most properties and critical issues are the same as for the conventional austenitic stainless steels. An important question regarding the Mn-steels relates to the relatively high nitrogen concentration contained in many alloys to help stabilize the

austenite phase. Since carbon-14 produced from the nitrogen is a major contributor to the long-term activation problem, the nitrogen must be maintained at relatively low concentrations to meet Class-C waste disposal criteria. The irradiation induced swelling behavior at low temperature is expected to be similar to that for conventional austenitic steels, however, the data base is very limited. The SCC and low-temperature fracture toughness (after irradiation) are the most serious issues. Significant uncertainties exist regarding the difficulties and properties of weldments. Major questions relate to whether an adequate data base on Mn-steels can be provided for ITER and whether a major R&D effort is justified for an alloy system that does not appear attractive for commercial applications. The low activation benefit must also be balanced with the perceived poorer safety rating compared to most other alloy systems. Manganese is one of the most volatile alloying elements in austenitic steels and ^{54}Mn is potentially a major contributor to radioactive effluents in the event of an accident.

The Mn-steels of interest should contain low nickel and nitrogen concentrations while the molybdenum and niobium should be removed. Although many Mn-steels contain significant amounts of nickel and nitrogen to stabilize the austenite phase, several alloys appear attractive. It is recommended that existing commercial alloys such as the French alloy AMCR be evaluated in order to take advantage of the existing baseline data. Two compositions of particular interest are:

AMCR: Fe-17Mn-10Cr-0.7Ni-0.2C-0.06N

AMCR-0035: Fe-19.9Mn-14.1Cr-0.26Ni-0.03C-0.05N

Recent investigations in the U.S. have provided valuable information on the phase stability of the Mn-steels. Based on this work, additional heats of special compositions could also be prepared if commercially available alloys of desired compositions are not available. Optimization of this alloy class does not seem feasible for ITER and it may not be warranted because these alloys are not considered leading candidates for commercial reactor applications.

4.3 FERRITIC-MARTENSITIC STEELS

Ferritic/martensitic steels such as HT-9 or low activation compositions are not recommended for the low-temperature ($< 300^{\circ}\text{C}$), frequent shutdown operating scenario projected for ITER. Primary concerns relate to:

- Radiation-induced increases in the ductile-to-brittle transition temperature (DBTT) to temperatures above the minimum operating temperature for most proposed concepts.
- The more difficult welding and post-weld heat treatment requirements for complex structures.
- Additional concerns related to the sensitivity to hydrogen embrittlement at low temperatures.

4.4 BERYLLIUM

Beryllium is chosen for use in the breeding blanket because of its neutron multiplication properties. As an example, the TIBER study called for 32 Tonnes of beryllium at an estimated cost of \$19M. The beryllium was in the form of 1.25 million, 3 cm dia. spheres, at a cost of \$15 each. Other designs call for a similar quantity but in the form of plates. ITER is expected to use more beryllium because it will probably be a larger machine than TIBER. There are several issues concerning the use of beryllium in ITER.

Current neutron transport models and the nuclear data may not accurately predict the neutron multiplication of beryllium in thick assemblies. Thus, it may be desirable to make some measurements of beryllium neutron multiplication. An accuracy of 3% appears achievable and would help in reducing the issue of predicting the tritium breeding in ITER.

The standard method of making beryllium parts by machining them out of stock made from hot pressed powder leads to considerable cost and material wastage from complex machining. The \$19M estimated for TIBER may be too low. The method of first cold pressing and then sintering should result in little or no final machining and little material wastage and therefore lower cost. There are no radiation damage data on this type of beryllium. The

mechanical properties should be determined after irradiation up to a fluence of $2 \times 10^{22} \text{ n/cm}^2$ above 1 MeV which corresponds to $> 10,000 \text{ appm He}$ in beryllium and a first wall fluence of $\sim 4 \text{ MW.y/m}^2$ in ITER.

Tritium buildup in neutron-irradiated beryllium is also an issue. Both ^6Li , ^7Li and ^3He are produced and neutron capture leads to tritium. Tritium is nearly impermeable in beryllium at low temperatures. If for any reason the temperature of the irradiated beryllium rises to a significant fraction of the melting point of 1278°C such as in a loss-of-coolant accident, the accumulated tritium could suddenly be released. This issue can be much better understood by measuring the tritium release rate versus temperature of irradiated beryllium.

Beryllium has been used successfully, unclad, exposed directly to water in a neutron and gamma flux of fission reactors for decades. Some versions of the tritium producing blanket for ITER have this combination with LiNO_3 added. The presence of the dissolved nitrate salt might enhance corrosion. In addition, there is the issue of the effect of the magnetic field which with a flow will cause a voltage to be set up, which would have a tendency to enhance corrosion. Experiments to mock up these conditions and measure the corrosion rate would clarify these two issues. If the corrosion rate is too high, a solution would be to clad the beryllium. However, cladding raises its own questions.

CHAPTER 5
INBOARD SHIELD DESIGN

5.0 INBOARD SHIELD DESIGN

5.1 INTRODUCTION

The inboard (i/b) shield is responsible for protecting the inner legs of the toroidal field (TF) magnets. The thickness of the shield is dependent on the radiation damage allowance for the magnet, the shielding materials used, and the level of neutron production in the reaction chamber. The philosophy of the ITER shield design is to meet both the magnet protection requirements and safety-related criteria. The latter has imposed some constraints on the material selection so that the reactor components will not be harmed by the shield decay heat during any abnormal operation.

The baseline design of ITER calls for a major radius of 4.04 m and provides a 55 cm space for the i/b shield. The neutron wall loading peaks at the midplane at a value of 1.47 MW/m^2 . The reactor operates at a fusion power of 631 MW. During its planned 15 year life, 2.9 full power years (FPY) of DT operation are expected. The configuration of the ITER i/b shield is based on an earlier work done on the compact TIBER-II reactor, in which an extensive optimization study was performed for a tungsten-based shield cooled with the $\text{H}_2\text{O}/\text{LiNO}_3$ aqueous solution.⁽¹⁾ The shield is envisaged to consist of W layers oriented circumferentially with vertical coolant channels and the layering of the W is progressively thicker. Although the W provides excellent magnet protection, its high specific decay heat, particularly in the front layers of the shield, caused some concern in case of LOCA/LOFA accidents, as indicated by the activation and safety analyses. This suggested the selection of different materials and/or the relocation of various components. On this basis, a trade study was carried out in which W is replaced with steel and the sensitivity of machine size, cost, and magnet damage to such change was determined.

The radiation limits imposed by the TF magnet designers is discussed in Section 5.2. The tungsten versus steel trade study is covered in Section 5.3 and the selected i/b shield configuration for the reference ITER design is presented in Section 5.4. The thermal hydraulics analysis is provided in Section 5.5. The conclusions to be drawn from this work are presented in Section 5.6.

5.2 MAGNET RADIATION LIMITS

The radiation limits for the magnets determine the shield thickness that directly influences the cost. The superconducting magnet components most sensitive to radiation damage are the superconductor filaments, stabilizer, and organic insulators. In addition to its effect on winding temperatures, nuclear heating affects the economic performance of the reactor through increased refrigeration costs. Radiation effects are related as they are determined by the flux level in the magnet.

The effect of radiation on the critical properties of the superconductor material is related to the damage produced by fast neutrons through the production of defect cascades.⁽²⁾ The amount of produced damage is usually measured in displacements per atom (dpa) or, more practically, damage energy available per atom in eV/atom. Due to the steep variation of the damage energy cross section, the damage produced by the same neutron fluence in different facilities will be different depending on the neutron spectrum. The relative number of neutrons ($E > 0.1$ MeV) that produces equal damage in superconductors as that produced by one 14-MeV neutron in RTNS-II was calculated by Guinan et al.⁽²⁻⁴⁾ to be 5.74, 7, 3.68 and 4.7 for HFBR, TFCX, STARFIRE and MARS, respectively. We calculated the corresponding numbers for MINIMARS and ITER to be 5.1 and 4.3, respectively.

Most experimental data involve irradiation of Nb_3Sn at fission reactor ambient temperatures⁽⁴⁾ with limited low fluence 4 K irradiation data.⁽³⁾ The critical temperature (T_c) is nearly constant up to a fast neutron fluence of $\sim 10^{18}$ n/cm² and drops rapidly at higher fluences.⁽³⁾ All T_c data from irradiation in different facilities at different temperatures agree when compared on a damage energy basis.⁽³⁾ Recent data for 19-core and 10,000-filament Nb_3Sn wires⁽⁵⁾ agree also very well with previous monofilament data. The drop in T_c is less than 3% up to a fusion reactor fluence ($E > 0.1$ MeV) of $\sim 5 \times 10^{18}$ n/cm² and increases to $\sim 20\%$ at $\sim 3 \times 10^{19}$ n/cm².

Figure 5-1 shows the experimental data for the effect of irradiation on the critical current density (J_c) compared on a damage energy basis. In general, an initial rise in J_c was observed with a subsequent drop at high fluences. Comparing the results of higher temperature HFBR irradiation⁽⁴⁾ with the 4 K irradiation of the nearly identical monofilament sample⁽³⁾ indicates that high temperature irradiation yields larger J_c degradation.

Fig. 5-1 Experimental Data for the Effect of Irradiation on J_c Compared on a Damage Energy Basis.

This is due to defect mobility and subsequent cascade collapse during the high temperature irradiation resulting in lower flux pinning. Hence, using the high temperature irradiation data yields conservatively low fluence limits. Recently two commercial Nb_3Sn wires supplied by two manufacturing companies have been irradiated at 4.2 K in RTNS-II to a 14 MeV fluence of 1.3×10^{18} n/cm² ($\sim 6 \times 10^{18}$ n/cm² in ITER).⁽⁶⁾ At this fluence, the two wires have J_c values that are above their preirradiation values by factors of 1.8 and 1.5. Based on these data, a fast neutron fluence limit of 10^{19} n/cm² is used in ITER.

Experimental data for fiber-reinforced organic insulators indicate that the mechanical properties degrade at a lower dose than do the electrical ones. Polyimides are 5 to 10 times more radiation resistant than epoxies.⁽⁷⁾ Samples of several millimeters-thick cylindrical rods of glass-fiber-filled (gff) polyimide were irradiated by gamma rays at 5 K and tested for flexural and compression strength.⁽⁷⁾ The samples are representative of relatively thick sheets of insulators placed between conductors with both compression and interlaminar shear being important. Reviewing the available 5 K irradiation experimental data it was found that the best gff epoxy retained 75% of the mechanical strength for doses up to 10^9 rads. The corresponding dose for the best gff polyimide is 10^{10} rads.⁽⁸⁾ Recently, 0.5 mm-thick disks of gff organic insulators were irradiated at 325 K to a mixed gamma and neutron dose of $\sim 4 \times 10^{11}$ rad and tested in compressive fatigue.⁽⁹⁾ No failures were observed for the insulators containing S-glass when cycled to a 345 MPa stress for over 1.5×10^5 cycles. No failures were observed in static compression tests with 2750 MPa stress.⁽⁹⁾ In these experiments, no interlaminar shear occurs and the results are applicable to designs with compressive strains only. The dose limit recommended for ITER is 10^{10} rads for gff polyimides with the possibility of raising the limit to $\sim 4 \times 10^{11}$ rads if the magnet is designed with the insulator loaded in compression only. Notice that in this case the insulator dose will not be a design driver since this dose corresponds to an excessive fast neutron fluence of $\sim 4 \times 10^{20}$ n/cm².

Neutron irradiation at cryogenic temperature produces immobile point defects in the stabilizer resulting in a zero-field radiation induced resistivity $\Delta\rho_r$ which impacts the total resistivity at field. Based on Kohler's plot for Cu, we generated a chart relating $\Delta\rho_r$ to the total

resistivity at field given the field B and the residual resistivity ratio RRR of Cu.⁽¹⁰⁾ The effect of stabilizer resistivity increase on both magnet stability and protection is design dependent. The forced-flow cable-in-conduit conductors (CICC) used in ITER are less sensitive to a Cu resistivity increase. Stability against disturbances is more dependent on the heat capacity of helium and less limited by heat transfer and Cu conductivity as with typical cryostable conductor designs. The total Cu resistivity in ITER is limited to 3 n Ω by stability and protection considerations. For a field of 12 T and Cu RRR value of 100, the charts of ref. 10 imply a limit on radiation induced resistivity of 2.33 n Ω . This corresponds to a dpa limit of 6.3×10^{-3} dpa. Using this limit and the calculated peak dpa rate in the stabilizer one can determine the number of anneals required for the total resistivity not to exceed the 3 n Ω limit taking into account the 80-90% recovery of radiation induced defects with annealing.⁽¹⁰⁾

The main technical effect of nuclear heating in a force-cooled CICC design is the coolant and conductor temperature rise from inlet to outlet. This can be reduced by shortening the flow path and/or increasing the mass flow per path. The total nuclear heating limit required to maintain adequate temperature and stability margins need to be provided by the magnet designers. A total heating limit of 72 kW for TIBER-II resulted in a temperature margin of ≥ 1 K and a stability margin of ≥ 300 mJ/cm³.⁽¹¹⁾ This limit is used as a guide in the ITER shield design.

5.3 TUNGSTEN VERSUS STEEL TRADE STUDY

The neutronics analysis of the i/b shield has indicated that satisfying the fast neutron fluence limit for the TF magnets is the design driver for the i/b shield. On this basis, an optimization study was performed and the i/b shield that minimizes the fluence was found to consist of a thick W shield followed by an 8 cm thick H₂O/LiNO₃ layer. In order to meet the thermal hydraulics requirements (see Section 5.5), the W shield was configured into alternating W layers and coolant channels. The layering of the W shield is progressively thicker. Thickness constraints of 1 and 7 cm are imposed on the first and second layers of the W shield, respectively. The next W layers could be as thick as 15 cm. In addition to the 57 cm thick i/b shield and first wall, the rest of the space between the plasma edge and the winding pack

consists of 10 cm scrape-off zone, 6 cm gap behind the shield, 5 cm coil case and 0.5 cm electric insulator. The coil case is cooled with 5 vol% liquid helium and the winding pack is composed of 38 vol% 304 SS, 28 vol% Cu, 7 vol% Nb₃Sn, 7 vol% GFF polyimide, and 20 vol% liquid helium.

Although the W provides an excellent protection for the TF magnets, the activation and the safety analyses of the i/b shield have identified several drawbacks for using W. The W has higher specific decay heat than steel and that makes the i/b shield vulnerable to overheating under LOCA/LOFA accident conditions. In order to design an inherently safe i/b shield, the decision was made to replace the W by steel, particularly in the front layers of the shield where the neutron flux is high and the decay heat problem is most severe.

Three iron-based materials are considered for the i/b shield. These are the primary candidate alloy (PCA), boron steel (B-SS), and iron boride (FeB). The problem was modeled for radiation analysis as infinite toroidal cylinders around the machine axis and the calculations are performed using the one-dimensional (1-D) code ONEDANT⁽¹²⁾ with the MATXS5 data library based on ENDF/B-V in 30 neutron and 12 gamma groups, and the P₃-S₈ approximation. The radiation effects at the TF magnets are reported in Table 5-1 for the different shielding materials. Our estimates show that the nuclear heating in the straight legs of the TF coils amounts to ~ 70% of the total heating in the 16 TF magnets. The W shield yields the lowest radiation effects at the magnets, followed by the FeB shield, then the B-SS shield and the PCA shield. As indicated by the fluence, the W actually overshields the magnets while the PCA and B-SS do not provide the necessary shielding.

The effects of the i/b shielding material and thickness on the plasma parameters, overall size and cost of ITER were assessed using the TETRA systems code with the assumptions and constraints used to generate the ITER design base case. The i/b shield thickness was varied over the range 50-75 cm. The results show that the fusion power, reactor size, and thus cost increase with the i/b shield thickness. For each centimeter increment in the i/b shield the major radius changes by ~ 1.5 cm. Figure 5-2 shows the variation of the direct cost with the i/b shield thickness for both W and Fe based shields, assuming the same unit cost for PCA, B-SS, and FeB shields. The direct cost increase is ~ \$5 M and \$4 M per centimeter increase in the W

Fig. 5-2 Effect of the i/b Shield Thickness on the Direct Cost of ITER.

Table 5-1. Radiation Effects in Inner Legs of the TF Magnets
for the Nominal ITER Baseline Design

Shield Type	W	PCA	B-SS	FeB
Peak Fast n Fluence (10^{19} n/cm ² @ 2.9 FPY)	0.31	1.59	1.24	0.82
Peak Nuclear Htg. in W.P. (mW/cm ³)	1.81	10.49	4.69	2.45
Peak dpa rate in Cu (10^{-3} dpa/FPY)	0.97	5.03	3.88	2.41
# of Anneals Required for Cu resistivity < 3 nΩm	1	∞	10	4
Peak Dose to GFF Polyimide (rads @ 2.9 FPY)	4.26×10^9	2.33×10^{10}	1.34×10^{10}	7.69×10^9
Nuclear Htg. in Straight Legs (kW):				
Winding Packs	5.33	29.06	16.6	9.63
Coil Cases	9.13	54.01	20.49	9.70

and steel shields, respectively. Neutronics calculations were then performed for each i/b shield thickness using the different shielding materials to determine the peak radiation effects in the TF coils. This led to the attenuation curves shown in Fig. 5-3. These curves indicate that the required shield thicknesses to meet the 10^{19} n/cm² fluence limit are 47.5, 58, 56 and 53 cm for the W, PCA, B-SS, and FeB shields, respectively.

It is interesting to combine Figures 5-2 and 5-3 to generate the variation of the direct cost with the fluence at the magnets. This variation is given in Fig. 5-4 and there are several features that should be noted. Satisfying the 10^{19} n/cm² fast fluence limit for the magnet, the direct cost is essentially the same for the PCA and W shields, although the inboard shield is 10 cm thicker for the PCA shield. At this fluence, the higher cost of the W shield is offset by the lower cost of the smaller size TF magnets. Importantly, the PCA shield improves the reactor safety at no additional cost. The use of steel shield is cost effective only for higher radiation limits than those adopted for the ITER study. For lower fluence limits, the use of W instead of steel is beneficial in terms of smaller TF magnets and,

Fig. 5-3 Variation of the Fluence at the TF Magnets with the 1/b Shield Thickness.

Fig. 5-4 Fluence at the TF Magnets Versus Direct Cost

thus, considerably lower direct cost. For ITER-type devices, the penalty for employing an order of magnitude lower fluence limit is about \$65 M increase in the direct cost and ~ 25 cm increase in the major radius. Because of limited data-base for B-SS and FeB, they will not be considered further in the present analysis.

5.4 REFERENCE DESIGN

In the current design of ITER, where the i/b shield is 55 cm thick, the all PCA shield resulted in radiation effects that exceed the design limits for the TF coils. In order to meet the limits without increasing the size of the machine, it was decided to combine both PCA and W in the i/b shield so that the PCA could be placed in the front layers of the shield. This offers the advantage of lessening the severity of the decay heat problem without altering the dimension of the device. It was also found that several centimeters of Be behind the first wall of the i/b shield will enhance the tritium breeding capability of the reactor. Therefore, the i/b shield is configured in 3 main layers: Be layer, followed by PCA layer, then W layer. The thicknesses of the 3 layers were determined based on the effect the different materials have on both the damage in the magnets and the tritium breeding ratio (TBR).

The effects of replacing the front layers of W by Be or PCA on the fluence and TBR are demonstrated in Figs. 5-5 and 5-6. The TBR includes both the i/b and o/b breeding as calculated in the toroidal 1-D model. The origin of the abscissa represents the all W shield case. Of the 55 cm thick i/b shield, 41.8 cm is occupied by W. As shown in the figures, replacing the W by Be or PCA increases the damage in the magnets, with more pronounced effect for Be. The TBR varies appreciably when Be is used but does not vary strongly when PCA is used. This is because of the remarkably enhanced neutron multiplication in Be.

As illustrated by Fig. 5-5, there are many options for replacing the front layers of the W by Be, PCA or combination of Be and PCA. About 14 cm of Be or 30 cm of PCA can replace the W without exceeding the 10^{19} n/cm² fluence limit for the magnets. Many combinations of Be and PCA with variety of thicknesses are also possible. For example, Be/PCA thicknesses of 2/26, 5/20, and 8/13 cm are acceptable. Hence, the choice between the different options should be based on the effect on the TBR. Figure 5-6 shows that significant

Fig. 5-5 Effect of Replacing the Front W Layers of the 1/b Shield by Be or PCA on the Fluence at the TF Magnets

Fig. 5-6 Effect of Replacing the Front W Layers of the 1/b Shield by Be or PCA on the Tritium Breeding Ratio

enhancement in the TBR is achieved with 5 cm thick Be layer and thicker Be layers do not benefit the breeding much. Therefore, we selected the option of 5 cm front Be layer, followed by 18 cm PCA layer, then 18.7 cm W layer to be the reference design for the 55 cm thick i/b shield of ITER. The distribution of the coolant channels within the reference shield is discussed in the next section.

The above shielding analysis was performed using the $\text{H}_2\text{O}/\text{LiNO}_3$ aqueous solution as a coolant for the i/b shield. It is of interest to determine the radiation level at the magnets when the other coolants, which are employed in the different blanket design options, are used to cool the reference design of the i/b shield. These coolants are the $\text{H}_2\text{O}/\text{LiOH}$, H_2O , and He gas. The results are summarized in Table 5-2 and show that the radiation effects slightly increase when the LiOH replaces the LiNO_3 salt and the design limits are still satisfied. The water and He gas coolants result in excessive radiation effects at the magnets and different i/b shield design should be adopted. It should be mentioned that in the nonbreeding physics phase the aqueous blanket design might employ either pure water or borated water to cool the shield. Compared to the pure water case, the borated water will result in lower radiation effects at the magnets. Even if pure water is used in the physics phase (0.4 FPY) and the aqueous solution is used in the breeding technology phase, the end-of-life fluence and insulator dose as well as the nuclear heating in the magnet are still acceptable.

5.5 THERMAL HYDRAULICS

The inboard shield is divided into six circumferential zones of shielding materials with six circumferential coolant channels running vertically from the bottom to the top, a distance of 5 m. Coolant at 40°C enters through a header at the bottom and flows in a single pass to the top finally exiting through another header. The shield configuration is shown in Figure 5-7.

P L A S M A

Fig. 5-7 Inboard Shield Configuration

Table 5-2. Radiation Effects in Inner Legs of TF Magnets
for the Nominal ITER Baseline Design

Coolant	H ₂ O/ LiNO ₃	K ₂ O/ LiOH	H ₂ O	He
Peak Fast n Fluence (10 ¹⁹ n/cm ² @ 2.9 FPY)	0.93	0.97	1.0	50
Peak Dose to GFF Polyimide (10 ¹⁰ rads @ 2.9 FPY)	0.96	1.0	2.16	25.5
Peak dpa Rate in Cu Stabilizer (10 ⁻³ dpa/FPY)	2.8	2.9	3.1	89
Min. # of Anneals Needed for $\rho < 3$ nm	1	1	1	∞
Peak Nucl. Htg. in W.P. (mW/cm ³)	5.1	5.4	16.1	124
Nucl. Htg. in Straight Legs (kW):				
Winding Packs	16	16.5	34.3	646
Coil Cases	25.6	26.8	109.5	373
Total Nucl. Htg. in 16 TF Magnets (kW)				
Winding Packs	23	23.7	49	923
Coil Cases	36.6	38.3	156	532

The first wall composed of graphite tiles and PCA is cooled by a 0.8 cm thick coolant channel. The next four channels are 1.0 cm thick and are spaced for approximately the same power recovery. Finally the last channel is 8 cm wide. This coolant is needed as a moderator and also as a means of supplying coolant to the upper divertor zone. The temperature rise in the first five channels is 20°C and in the last, 4°C. Maximum operating pressure is the same as the outboard blanket, 0.27 MPa. The maximum graphite tile temperature facing the plasma is 161°C, the maximum coolant interface temperature is 80°C and the maximum material centerline temperature is 197°C.

As in the case of the outboard blanket, the inboard shield is capable of operating at $\pm 50\%$ of neutron wall loading at a coolant pressure of 0.41 MPa. Table 5-3 gives the thermal hydraulics parameters for 100% and 150% of power operation.

Table 5-3. Thermal Hydraulics Parameters - Inboard Shield

% of Nominal Power	100%	150%
Coolant Inlet/Outlet Temp., °C	40/60	40/60
Coolant Inlet/Outlet Pressure, MPa	0.27/0.17	0.42/0.2
Max./Min. Velocity, m/s	3.9/0.2	5.9/0.3
Mass Flow Rate, kg/s	1991	2987
Max./Min. FW Temp. (behind C tile), °C	152/132	191/171
Max./Min. Graphite Tile Temp., °C	161/141	204/184
Max./Min. PCA-Coolant Interface Temp., °C	75/50	77/52
Max./Min. Be-Coolant Interface Temp., °C	74/42	75/43
Max./Min. W-Coolant Interface Temp., °C	80/47	82/49
Max. PCA Centerline Temp., °C	197	260
Max. Be Centerline Temp., °C	100	114
Max. W Centerline temp., °C	93	104
Pumping Power, MW	0.24	0.8

5.6 CONCLUSIONS

The impact of the i/b shielding materials is analyzed with respect to magnet protection, reactor size, and cost. The use of steel shield is cost effective only for higher radiation limits than those adopted for the ITER study. An arrangement of the Be, PCA, and W materials within the i/b shield of the base case of ITER was proposed. The neutronics, safety, and thermal hydraulics requirements are all met and there appear to be no feasible problems in the aqueous cooled Be/PCA/W i/b shield design. For other blanket options which employ H₂O or He coolants, different i/b shield design should be adopted.

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CHAPTER 6

AQUEOUS LITHIUM SALT BLANKET

6.0

AQUEOUS LITHIUM SALT BLANKET

6.1 INTRODUCTION

The aqueous self-cooled breeding shield concept used in the TIBER-II design⁽¹⁾ is analyzed for possible implementation in the ITER design. This concept is based on dissolving small amounts of lithium compounds in the water coolant.⁽²⁾ This makes it possible to breed tritium in all the shield zones, and results in tritium self-sufficiency as well as enhanced magnet protection due to strong neutron absorption in lithium. This is of particular significance in a compact technology test reactor such as ITER where the space available for separate breeding blankets is limited due to the large fraction of the outboard region devoted for test modules and plasma heating penetrations and because of the negative impact on magnet protection of using separate breeding blankets in the thin inboard and divertor shield zones. In this chapter, we will describe briefly the blanket and shield design proposed for ITER. Detailed analysis of this concept is given in the TIBER-II report.⁽¹⁾

6.2 NUCLEAR ANALYSIS

Many Li compounds have high solubility limits in water allowing for significant tritium breeding. A major consideration in the choice of Li salt is the compatibility with structural materials and neutron multipliers. Other considerations include the tritium breeding and shielding performance, induced radioactivity, salt radiolysis, and impact on tritium recovery and containment. Our results show no significant difference in breeding potential when salts with high solubility are used close to their solubility limits. Among the different Li salts lithium nitrate and lithium hydroxide are identified as the primary candidates.⁽¹⁾ Preliminary corrosion experimental results⁽³⁾ indicate that while LiOH is more corrosive at high temperatures, both salts yield comparable corrosion rates at temperatures below 100°C. Both Li salts are analyzed here for use in ITER.

Figure 6.1 is a midplane cross section of a blanket segment. Thermal hydraulics considerations resulted in using a double first wall that insures uniform cooling and minimizes system pressure.⁽¹⁾ A 2 cm thick first wall with 0.5 cm thick PCA layers surrounding a 1 cm thick coolant zone is used in

Fig. 6-1 Midplane Cross Section of Blanket

the analysis. Single size balls of beryllium and PCA are used to minimize the pressure drop. An optimum ball diameter of 3 cm yields a minimum system pressure of 0.27 MPa. Using balls in the design makes it possible to accommodate the complicated ITER geometry with its numerous penetrations. A 10 μm PCA coating is used for the Be balls with the LiNO_3 solution. Due to the larger potential for corrosion of Be with the LiOH , solution a 250 μm PCA cladding is used. The composition used for the front Be zone is 61.13% Be (0.9 density factor), 2.12% PCA, and 36.75% LiNO_3 solution. For the LiOH case the composition is 58.19% Be, 5.06% PCA, and 35.63% aqueous solution. The Li is enriched to 90% ^6Li and the results are normalized to the peak neutron wall loading of 1.91 MW/m^2 and total reactor life of 2.9 full power years (FPY).

The effects of Li salt concentration and multiplier zone thickness on the outboard tritium breeding ratio (TBR) were analyzed. The design that maximizes the outboard tritium breeding ratio utilizes a 40 cm thick zone of Be balls followed by a 80 cm thick zone of steel balls. The salt concentration is 16 g/100 cm^3 for LiNO_3 and 5 g/100 cm^3 for LiOH . Table 6-1 lists the nuclear parameters for the outboard blanket and shield. While most of the overall TBR is obtained from the outboard blanket and shield, some additional breeding will occur in the aqueous solution used as coolant in other zones. In addition, the impact of neutron reflection from these zones on tritium breeding in the outboard region has to be considered. This is properly assessed by performing neutron transport calculations using a toroidal cylindrical model. Several toroidal cylindrical geometry calculations have been performed with different materials being used in the inboard region of the model to simulate the effect of the different inboard and divertor shield zones as well as penetrations. The results of these calculations were then corrected for the actual coverage fractions of the different breeding zones (55.65% outboard blanket, 16.98% divertor shield, and 18.6% inboard shield) to determine the overall TBR as illustrated in Fig. 6-2. This approach was previously used in the early stages of the design of the HWL reactor⁽⁴⁾ and TIBER-II⁽¹⁾ and found to agree with the three-dimensional results to within ~ 4%. The estimated TBR of 1.1 represents a conservative estimate since penetrations and plasma heating zones are treated as black absorbers. In addition, no credit is taken for breeding in the test modules. They will

Fig. 6-2 Estimation of ITER Overall TBR Using the Aqueous Blanket

Table 6-1. Nuclear Parameters for the Outboard Aqueous Blanket

Li Salt	LiNO ₃	LiOH
Salt concentration (g/100 cm ³)	16	5
Local TBR	1.506	1.442
Local energy multiplication	1.366	1.364
Power density at midplane (W/cm ³)		
First wall	15.20	15.85
FW coolant	24.97	24.48
Second wall	12.83	13.61
Front of Be zone	13.28	13.28
Average in Be zone	4.23	4.22
Front of SS zone	0.76	0.75
Average in SS zone	0.064	0.064
Peak dpa/FPY in first wall	16.86	16.75
Peak He appm/FPY in first wall	190.1	189.6
Peak fast neutron fluence in superconductor at 2.9 FPY (n/cm ²)	1.02 x 10 ¹⁶	1.12 x 10 ¹⁶
Peak insulator dose at 2.9 FPY (rads)	1.08 x 10 ⁷	1.21 x 10 ⁷
Peak winding pack power density (mW/cm ³)	3.81 x 10 ⁻³	4.24 x 10 ⁻³
Peak Cu dpa/FPY	2.88 x 10 ⁻⁶	3.2 x 10 ⁻⁶

contribute about 0.07 to the overall TBR assuming a local TBR of ~ 1.1. Following the same procedure the estimated overall TBR for the LiOH solution is 1.05 excluding test modules. The cost impact of using the LiNO₃ solution and Be in ITER is given in Table 6-2.

6.3 THERMAL HYDRAULICS

The outboard blanket is cooled with an aqueous salt solution of 16 g/100 cm³ LiNO₃. The coolant enters the reactor through a header at the bottom of the blanket and flows upwards in a single pass, a distance of 6.2 m, finally exiting the reactor through another header at the top. Coolant flows in three separate zones of the blanket: the first wall, the Be zone and the steel zone. The first wall consists of a double layer of steel with coolant in between. The front layer was taken to be 0.8 cm thick to make allowance for possible material sputtering and the rear layer was taken as 0.2 cm thick.

Table 6-2. Cost Impact of Using Li Salt and Be in ITER

	No Be No Li Salt	No Be Li Salt	Be Li Salt
Local TBR	0	0.676	1.506
Overall TBR	0	0.42	1.10
Local M	1.367	1.2	1.366
kg T ₂ purchased in technology phase	97	60	0
Cost of T ₂ purchased in technology phase (M\$)	970	600	0
Tonnes of Be needed	0	0	90
Be cost (M\$)	0	0	36
T ₂ extraction cost (M\$)	0	24	60
Cost of additional heat removal (M\$)	10	0	10
¹⁴ C waste disposal cost (M\$)	0	1	1
LiNO ₃ cost (M\$)	0	7	7
Total of relevant cost items (M\$)	980	632	114
Cost savings over reactor life (M\$)	0	348	866

The Be and steel zones consist of closely packed 3 cm diameter balls. An optimization study⁽¹⁾ has shown that this size balls gives the lowest pressure drop for the selected operating condition. Coolant at 40°C enters each of the zones and flows at different velocities. The maximum velocity at the first wall is 4.9 m/s. In the Be and the steel zones the average superficial velocity is 15.5 cm/s and 0.3 cm/s respectively. The coolant temperature rise in the first wall zone is 15°C and in the Be and steel zones, it is 35°C. Coolant pressure drop in the first wall and the Be zone is 0.12 MPa and much lower in the steel zone. The maximum coolant pressure is 0.27 MPa. Suppression of radiolysis may require a higher overpressure but this is yet to be determined.

The highest temperature, 230°C occurs on the first wall facing the plasma at the top, and the thermal stress is 400 MPa. These values would be much lower if this layer can be less than 0.8 cm thick. The highest coolant interface temperature is 83°C and occurs in the first row of Be balls.

The blanket is designed to be capable of operating with a power variation of $\pm 50\%$ at a pressure of 0.42 MPa. Table 6-3 gives the thermal hydraulic parameters for operating at 100% and 150% of power.

6.4 ACTIVATION ANALYSIS

Detailed activation analysis was performed for the outboard and inboard blanket and shield simultaneously using the DKR-ICF code⁽⁵⁾ and the ACTL library.⁽⁶⁾ The options of using PCA or the low activation austenitic steel Tenelon were assessed. 2.9 FPY reactor operation was used in the calculations. The inboard activity is higher than the outboard up to ~ 5 days following shutdown and after 5 years. The total activity in the blanket and shield using Tenelon is higher than that obtained using PCA up to ~ 8 hours following shutdown but is much lower than the PCA activity after ~ 10 years. The total activity in the blanket and shield (excluding the coolant) at shutdown is 1560 MCi with Tenelon and 1130 MCi with PCA. The total biological hazard potential (BHP) at shutdown is $2.1 \times 10^8 \text{ km}^3$ of air with Tenelon and $2.3 \times 10^8 \text{ km}^3$ with PCA.

The waste disposal ratings (WDR) for class C low level waste (LLW) were calculated for the outboard and inboard blanket and shield using both the 10 CFR 61 limits⁽⁷⁾ and the limits calculated by Fetter et al.⁽⁸⁾ Table 6-4 lists the results with the major contributing radionuclides indicated. The WDR range given corresponds to the range given by Fetter for the limits. The outboard blanket and shield qualifies as class C LLW. If tenelon is used it can even qualify as class A waste. The inboard shield by itself qualifies as class C waste only if Tenelon is used. If PCA is used exclusively, then class C qualification for the inboard and outboard shields can be realized if the two are disposed of together. The total amount of ^{14}C produced in the coolant is 4650 Ci for LiNO_3 and 54 Ci for LiOH . Since 20% of the coolant will be processed per full power day for tritium recovery, ^{14}C can be removed at the

Table 6-3. Thermal Hydraulics Parameters - Outboard Blanket

% of Nominal Power	100%	150%
Coolant Inlet/Outlet Temp. (FW), °C	40/55	40/55
Coolant Inlet/Outlet Temp. (Be zone), °C	40/75	40/75
Coolant Inlet/Outlet Temp. (Fe zone), °C	40/75	40/75
Coolant Inlet/Outlet Pressure, MPa	0.27/0.14	0.42/0.16
Max./Min. Velocity in FW, m/s	4.9/3.9	7.4/5.8
Mass Flow Rate in FW*, kg/s	1269	1903
Mass Flow Rate in Be Zone*, kg/s	2082	3123
Mass Flow Rate in Fe Zone*, kg/s	83	125
Max./Min. Temp. FW Facing Plasma, °C	230/128	310/168
Max./Min. Temp. FW/Coolant Interface, °C	79/43	82/45
Max./Min. Temp. Be/Coolant Interface, °C	83/48	85/50
Max./Min. Temp. Fe/Coolant Interface, °C	81/46	82/47
Pumping Power, MW	0.43	1.42
FW Thermal Stress, MPa	400	600

*Test zones and penetrations excluded.

same rate leading to a coolant specific activity in the reactor of only 0.05 Ci/m³ for the LiNO₃ solution. Based on the 10CFR61 limit of 8 Ci/m³ of ¹⁴C for class C waste, the volume of low level operational waste to be disposed of annually is 50 m³ for LiNO₃ and 0.6 m³ for LiOH.

Figure 6-3 shows the effect of using different shield designs on integrated decay heat in the inboard shield. It is clear that the optimized Be/SS/W shield used in ITER results in decay heat very close to that in an all steel shield and much less than that in an all W shield used in the TIBER-II design. The inboard shield produces more decay heat than the outboard blanket and shield up to ~ 5 days and after ~ 5 years. The integrated decay heat values in one day after shutdown in the outboard region are 68 and 97 GJ for PCA and Tenelon, respectively. The corresponding values for the inboard shield are 86 and 141 GJ.

Fig. 6-3 Integrated Decay Heat in Inboard Shield

Table 6-4. Class C Waste Disposal Rating for
ITER Blanket/Shield (No Compactness)

	i/b: Be/PCA/W o/b: Be/PCA	i/b: Be/Tenelon/W o/b: Be/Tenelon
<u>Inboard Region</u>		
10CFR61 limits	5.96 (5.4 ^{94}Nb , 0.41 ^{63}Ni)	0.025 (0.018 ^{94}Nb , 0.004 ^{14}C)
Fetter limits	6.01 - 9.76 (5.4 ^{94}Nb , 0.42-4.2 ^{99}Tc)	0.0571 - 0.0581 (0.038 $^{108\text{m}}\text{Ag}$, 0.018 ^{94}Nb)
<u>Outboard Region</u>		
10CFR61 limits	0.514 (0.446 ^{94}Nb , 0.051 ^{63}Ni)	1.65×10^{-3} (0.001 ^{94}Nb , 4.6×10^{-4} ^{14}C)
Fetter limits	0.504 - 0.842 (0.446 ^{94}Nb , 0.038-0.38 ^{99}Tc)	4.4×10^{-3} - 4.53×10^{-3} (3.2×10^{-3} $^{108\text{m}}\text{Ag}$, 1×10^{-3} ^{94}Nb)
<u>Avg. over i/b and o/b regions</u>		
10CFR61 limits	1.179	4.35×10^{-3}
Fetter limits	1.176 - 1.926	0.011

6.5 LOFA/LOCA ANALYSIS

The outboard blanket and the inboard shield were analyzed with respect to loss of flow (LOFA) and loss of coolant (LOCA) accidents. In case of LOFA both inboard and outboard regions will have adequate natural circulation of the coolant with indefinite protection. The analysis has concentrated on LOCA, since this is the more critical case.

In the LOCA analysis we assume that the plasma stays on for 10 seconds after the instantaneous loss of all the coolant. The inboard and outboard sides are solved simultaneously assuming that the space between them is filled

with air at one atmosphere. We also assume that air at 1 atm. fills the spaces where there is normally coolant. The code ATHENA⁽⁹⁾ a one-dimensional thermal hydraulics code developed at EG&G, Idaho has been used to analyze LOCA. In the version used, there is no thermal radiation, however, a newer version of ATHENA which includes radiation will soon be available. The superconducting TF coils at the back of the inboard shield are used as heat sinks but the outboard blanket has an adiabatic boundary on its outer perimeter.

The outboard blanket is modelled as consisting of a first wall physically separated from the blanket, and a homogenized heat conducting zone of the Be and steel. The inboard shield is modelled as five heat conducting slabs separated by six cooling ducts running vertically between the slabs (see Fig. 5-7 in Section 5.5).

Figure 6-4 shows the outboard blanket temperature history following LOCA, assuming PCA structure and Be and PCA balls. It is expected that the first wall temperature, which reaches 625°C in 24 hours, will be lower when radiation is incorporated, as well as conduction along the connecting plates to the bulk blanket. The heat up in the outboard blanket is very low as anticipated.

Figure 6-5 is the temperature history of the inboard shield using Be/PCA/Tungsten materials. It can be seen that the first wall (Zone 10) temperature at 24 hours is 660°C and that of Be, 615°C. It is felt that 24 hours will give enough time to incorporate some active cooling and prevent further heatup of these zones.

6.6 KEY TECHNICAL ISSUES

Long term corrosion and the radiolytic decomposition of the aqueous solutions in contact with the shield materials is a critical issue for this blanket concept. While there is no experimental evidence to substantiate this concern, a relatively inexpensive (< 0.5 M\$) R&D experimental program will provide data in the next three years. The technology required for tritium recovery has been demonstrated on a large scale by Ontario Hydro. The time average tritium inventory in the reactor coolant loop of 400 g is less than that found in current CANDU reactors. This inventory can be reduced at little or no penalty in TBR by using the aqueous solution only in the front zones of

*RADIATION AND CONDUCTION ALONG CONNECTING
PLATES WILL REDUCE THIS TEMPERATURE

Figure 6-4. Outboard Blanket Temperature History Following LOCA Using Be/PCA Materials.

Figure 6-5. Inboard Shield Temperature History Following LOCA Using Be/PCA/W Materials.

the outboard blanket and inboard shield. Reducing the aqueous solution volume in the reactor by a factor of 8 leads to a reduction in TBR of only 5%. If credit is taken for tritium bred in the test modules an overall $TBR \geq 1.1$ can still be maintained. The relatively large amount of C-14 produced in $LiNO^3$ is a major concern. Since the radiolysis problem is less pronounced with LiOH and comparable corrosion rates are produced by both salts at low temperatures⁽³⁾, LiOH is proposed for use in ITER. As indicated before tritium self-sufficiency can still be achieved.

6.7 CONCEPT EVALUATION

The aqueous lithium salt blanket concept satisfies all the absolute requirements for ITER given in Section 2.2. This section evaluates the blanket based on the evaluation criteria prescribed in Section 2.2.

The total lifetime cost for the reactor utilizing this blanket concept was calculated to be \$5042 M. This is the lowest cost obtained using any of the proposed blanket concepts and a score of 3 is given for the cost criterion. It is estimated that intermediate R&D requirements are needed for this concept with data available from small to modest facilities built within 3 years. A score of 2 is given for R&D requirement. Due to the design simplicity, single coolant system, and ability to accommodate geometrical complications and penetrations, this concept scores 3 for the design and fabrication complexity criterion. This design provides the flexibility of changing from the nonbreeding physics phase to the breeding technology phase by simply adding the Li salt to the water coolant and hence scores 3 for compatibility with phased operation. A time between 1 and 3 months is required for changeout of a single blanket module leading to a score of 2 for maintenance. Since an overall $TBR \geq 1.1$ is achievable the concept scores 3 for tritium breeding. The technology for tritium recovery from heavy water has been demonstrated by Ontario Hydro. This concept scores 2-3 for tritium extraction.

Because of the relatively large tritium inventory, this concept scores 1 for accidental tritium release. It scores 2-3 for accidental activated product release and 2-3 for routine tritium release. Because of the potential for water interaction with the liquid lithium used in the test modules a score of 1-2 is given for chemical and thermal reaction potential. The LOCA analysis indicates that half the melting point for Be can be reached in 1-7 days

implying a score of 2. A score of 2 is given for waste disposal rating with operational waste resulting from C-14 production not being a concern with LiOH. Since LiOH cannot be used at high temperatures typical of power reactors due to excessive corrosion, a score of 1 is given for power reactor relevance.

In summary, a low cost, simple, flexible, and low risk state of the art tritium breeding concept has been developed for ITER. This design could be constructed and implemented in the next 10 years with no major increase in the current R&D levels.

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CHAPTER 7

WATER-COOLED SOLID-BREEDER BLANKET

7.0

WATER-COOLED SOLID-BREEDER BLANKET

7.1 GENERAL DESCRIPTION

A water-cooled, solid-breeder blanket concept has been developed for ITER using a low-pressure coolant system. The blanket is designed to produce the necessary tritium required for the operation of ITER and to operate at commercial power reactor conditions as much as possible. The other consideration is to minimize the impact on the design and operation of ITER.

Because of the ITER test plan and the desire to reduce the reactor size, a large fraction of the fusion neutrons (less than 0.5) is not available for tritium breeding. This mandates the use of beryllium to multiply the fusion neutrons for tritium production. Beryllium is the only non-fissionable material with adequate neutron multiplication for ITER conditions⁽¹⁻³⁾. In order to utilize these neutrons, the solid breeder has to contain a large concentration of the lithium-6 isotope. A 90% lithium-6 enrichment is used with any of the solid breeders (Li_2O , LiAlO_2 , or Li_4SiO_4). The use of high lithium-6 enrichment has several advantages. It reduces the solid breeder volume required in the blanket. This reduction results in a lower tritium inventory and low blanket cost. Also, the high lithium-6 enrichment insures the same breeding ratio at the beginning and end of life for the blanket (4.7 MW-y/m² peak fluence). The design analysis in this chapter are based on the use of Li_2O breeder.

For satisfactory tritium inventory, the solid breeder temperature has to be within a specific temperature range. The minimum temperature of this range for any of the solid breeders under consideration is about 350 to 400°C which does not match the desired operating temperature range for a low-pressure water coolant system. The adopted design approach is to locate the beryllium multiplier between the solid breeder and the water coolant to raise the breeder temperature to the required temperature range without relying on an adjustable gap conductance between the solid breeder and the steel clad. This design approach increases the blanket reliability and simplifies the design configuration. This arrangement is also desirable from the neutronics viewpoint to maximize the tritium breeding ratio. The use of the beryllium zone between the water coolant and the solid tritium breeder increases the

number of tritium barriers which reduces the tritium permeation from the solid breeder to the water coolant. Carbon is used at the back of the blanket because of its low absorption cross section as a reflector and moderator. In addition, it operates as a thermal insulator between the cold steel shield and the solid breeder.

The above discussion implies that a multilayer configuration is well suited for this blanket concept. Also, it reduces fabrication costs and simplifies the design. Two or three thin layers of solid breeder (1.8 to 3 cm total thickness) with beryllium layers (20 to 27 cm total thickness) in between are used. In fact, the solid breeder layers are inserted in the beryllium material. The use of thin layers of solid breeders results in 100 to 200°C temperature variation in the breeder material which is a small fraction of the solid breeder temperature range required for satisfactory performance. This permits an increase in the neutron wall loading up to 40 to 60% from the nominal value without exceeding the maximum allowable temperature of any material.

The one-dimensional tritium breeding ratios are 1.7 and 1.9 for the two and three breeder layers configurations, respectively. The energy multiplication factor is about 1.6 for these configurations. The tritium purge system design shows a total tritium inventory of 1.7 g in the Li_2O breeder and the beryllium multiplier. The maximum partial tritium pressures are the same in the breeder and the multiplier to eliminate the tritium permeation between these materials. The total pumping power required for the helium purge system is 3.5 kW. The radioactivity analysis shows the blanket materials satisfies class C for waste disposal.

7.2 CONCEPT DESIGN

Several blanket arrangements were analyzed to determine the tritium breeding potential based on one-dimensional calculations. The tritium breeding ratio was calculated as a function of the beryllium zone thickness in front of the Li_2O breeder. Table 7-1 gives the blanket parameters considered for this calculation. The one-dimensional discrete ordinates code ONEDANT⁽⁴⁾ was used to perform the transport calculations with a P_5 approximation for the scattering cross sections and an S_8 angular quadrature set. A 67-coupled group nuclear data library (46-neutron and 21-gamma) based on ENDF/B-IV was

Table 7-1. Blanket Parameters for the One Beryllium Zone Configuration

Zone Description	Zone Thickness, cm	Zone Composition, Volume %
First Wall	1	50% H ₂ O, 50% Type 316 SS
Multiplier	x	100% Be (0.8 DF)
Second Wall	0.5	50% H ₂ O, 50% Type 316 SS
Breeder	40 ^a	5% H ₂ O, 10% Type 316 SS, 5% He, 80% Li ₂ O ^a (0.8 DF)
Shield	100	20% H ₂ O, 80% Type 316 SS

^a A 90% lithium-6 enrichment.

Table 7-2. Tritium Breeding Ratio and Energy Generated Per Fusion Neutron for the One Beryllium Zone Configuration

Beryllium Zone Thickness, cm	Tritium Breeding Ratio	Energy Generated MeV/DTn
2.5	1.155	18.9
5.0	1.293	20.0
7.5	1.367	21.0
10.0	1.396	22.0
12.5	1.392	22.9
15.0	1.367	23.7
17.5	1.327	24.5
20.0	1.278	25.1

employed for these calculations. Vitamin-C⁽⁵⁾ and MACKLIB-IV⁽⁶⁾ libraries were used to obtain this library. The results from this calculation are shown in Table 7-2. The maximum TBR is ~ 1.4 at a 10 cm beryllium zone thickness.

Blanket configurations with two beryllium zones were considered to improve the tritium breeding ratio. Table 7-3 gives a sample of the configurations as a function of the first beryllium zone thickness with a tritium breeding ratio > 2.0. The thermal hydraulics analysis of this configuration suggested the use of thin breeder layers to control the temperature distribution within the required limits. Design configurations with three breeder layers and high tritium breeding ratios were developed⁽⁷⁾. Table 7-4 gives the configuration for two of these designs. The total breeder zone thickness is 6 cm. Again, the thermal hydraulics analysis suggested that each of the last two coolant panels be located within the preceding beryllium zone or that different gap conductance values be used at the breeder interfaces with the steel clad.

Another simpler configuration has been developed based on two breeder layers (1.8 cm total thickness) as shown in Table 7-5. This configuration has a 1.7 tritium breeding ratio with 18-cm beryllium. The first wall/blanket/shield design and optimization system (BSDOS)⁽⁸⁾ was employed to carry out the neutronics and thermal hydraulics analyses in an integrated manner where the beryllium zone thicknesses were adjusted for satisfactory breeder temperatures. The analysis was performed for a 1.0 MW/m^2 neutron wall loading. The breeder temperatures were adjusted at the bottom of the breeder temperature range for satisfactory tritium recovery. This configuration can accommodate a 40% increase in the neutron wall without exceeding the maximum allowable temperature of any material as shown in Table 7-5. If an extra margin in this configuration with respect to the changes in the neutron wall loading is required, the reduction of the breeder zone thickness by 1 mm increases the margin by ~ 10%.

The mechanical design of the last configuration is shown in Figs. 7-1 and 7-2. A whole sector is shown in Fig. 7-1 where the reactor is divided into 16 sectors. On the outboard, the sector consists of two sections. Each section has its coolant and tritium purge lines, and it can be removed separately from the reactor. The water coolant flows in the toroidal direction inside the sector as shown in Fig. 7-2. Two poloidal manifolds are integrated in the

Table 7-3. Tritium Breeding Ratio and Energy Generation Per Fusion Neutron for Two Beryllium Zone

Zone Thickness, cm						
First Wall ^a	Beryllium Multiplier ^b	Li ₂ O Breeder ^c	Be Multiplier ^b	Li ₂ O Breeder ^c	TBR	Energy MeV/DTn
1	1	1	20	35	2.039	22.0
1	2	1	20	35	2.047	22.2
1	3	1	20	35	2.048	22.4
1	4	1	20	35	2.044	22.6
1	5	1	20	35	2.034	22.8

^a First wall consists of 50% H₂O, 50% Type 316 SS.

^b 100% Be (0.8 DF)

^c 5% H₂O, 10% Type 316 SS, 5% He, 80% Li₂O, (0.8 DF)

Table 7-4. Dimensions and Tritium Breeding Ratios for Three Breeder Layers Configuration

Zone Material	Configuration A Zone Thickness, cm	Configuration B Zone Thickness, cm
First Wall (50% H ₂ O, 50% Type 316 SS)	1.0	1.0
Be (0.8 DF)	4.0	4.0
Type 316 SS	0.1	0.1
Li ₂ O (0.8 DF)	1.0	1.0
Type 316 SS	0.1	0.1
Be (0.8 DF)	15.0	20.0
Coolant (50% H ₂ O, 50% Type 316 SS)	0.5	0.5
Li ₂ O (0.8 DF)	2.0	2.0
Type 316 SS	0.1	0.1
Be (0.8 DF)	5.0	10.0
Coolant (50% H ₂ O, 50% Type 316 SS)	0.5	0.5
Li ₂ O (0.8 DF)	3.0	3.0
Type 316 SS	0.1	0.1
Carbon	30.0	30.0
Tritium Breeding Ratio	1.889	1.979

Figure 7-1 Water-Cooled Solid Breeder Blanket Section

Figure 7-2 Water-Cooled Solid-Breeder Blanket

shield at the back of the sector where the water manifolds are utilized for shielding purposes. In the inboard section, the water flows only in the poloidal direction as shown in Fig. 7-1. Coolant panels or water tubes can be used to cool the blanket. The water panel can be viewed as a row of mechanically connected tubes for extra structure support. The outboard shield is integrated with the blanket, and it is designed to achieve a 2.5 mrem/h dose one day after shutdown in the reactor hall.

The radioactivity analysis for this blanket shows that it is qualified for a class C waste at the end of life based on the Nuclear Regulatory Commission regulations and $3 \text{ MW}\cdot\text{y}/\text{m}^2$ average fluence. The total carbon-14 production is 0.69 and 0.99 g in the Li_2O breeder and the water coolant, respectively from $^{17}\text{O}(\text{n},\text{He})^{14}\text{C}$ reaction. The corresponding ^{14}C concentration in these materials are 0.79 and 0.12 curies/ m^3 which qualify them a class A waste based on ^{14}C only.

The inboard shield configuration used for the water-cooled solid-breeder blanket has type 316 SS, H_2O , TiH_2 , and lead for shielding. The total shield thickness is 55 cm to fit the current design configuration. The zone thicknesses in the shield were optimized to satisfy the design guidelines. The compositions are not optimized yet. The shield performance is also shown in Table 7-6.

The tritium inventory formalism presented in the ITER Nuclear Design Guidelines was used to calculate tritium inventory in the Li_2O breeder. Input to these models and correlations was: tritium generation rate (G) = 84.6 g/ $\text{cm}^3\cdot\text{y}$, grain size (r_g) = 10 μm , temperature profile (T) = 1073-1600 η^2 , K where $0 \leq \eta \leq \frac{1}{2}$, H_2O partial pressure ($P_{\text{H}_2\text{O}}$) = 27 Pa; HTO partial pressure ($P_{\text{H}_2\text{O}}$) = 7 Pa, and the total breeder mass (M_b) = $5.2 \times 10^6 \text{ g}$. The assumptions and results are shown in Table 7-7. Under conditions of protium swamping, the major contribution to the inventory is due to solubility. The overall uncertainty in these calculations is taken to be a factor of 2. While the solubility is known to much better accuracy than this, the overall estimate includes uncertainties in the purge flow analysis which determines the partial pressures of H_2O and HTO . With regard to diffusion and surface desorption, even a three-order-of magnitude uncertainty in these would only increase the calculated inventory from 1.3 g to 8.2 g in the Li_2O .

Table 7-7 also summarizes the results of the Be calculations using the

Table 7-5. Blanket Parameters for the Two Breeder Layers Configuration

Zone	Material	Thickness, cm	Min.-Max. Temp., °C	
			1 MW/m ²	1.4 MW/m ²
First Wall	Type 316 SS	0.5	76-194	79-203
Coolant	H ₂ O	0.3		
Back Wall	Type 316 SS	0.2	91-163	111-208
Multiplier	Be (0.8 DF)	5.0	157-388	201-558
Clad	Type 316 SS	0.1	386-400	555-572
Breeder	Li ₂ O ^a (0.8 DF)	0.8	397-521	569-782
Clad	Type 316 SS	0.1	426-437	627-636
Multiplier	Be (0.8 DF)	7.5	133-427	168-625
Coolant Channel	Type 316 SS	0.2	80-139	95-177
Coolant	H ₂ O	0.2		
Coolant Channel	Type 316 SS	0.2	72-121	84-151
Multiplier	Be (0.8 DF)	10.0	115-390	143-565
Clad	Type 316 SS	0.1	389-398	564-574
Breeder	Li ₂ O ^a (0.8 DF)	1.0	397-518	573-795
Clad	Type 316 SS	0.1	526-528	792-795
Reflector	C	30.0	74-525	87-792
Shield	Steel Shield	88.7		

^a A 90% lithium-6 enrichment.

Table 7-6. Steel Inboard Shield for the Water-Cooled Solid-Breeder Blanket

Zone	Thickness cm	Composition Fraction by Volume
<u>Shield Arrangement</u>		
FW	1	0.7 Type 316 SS, 0.3 H ₂ O
Steel	34	0.9 Type 316 SS, 0.1 H ₂ O
TiH ₂	17	0.05 Type 316 SS, 0.95 TiH ₂
Pb	3	Pb
Total	55	
<u>Shield Performance^a</u>		
Maximum dose in the insulator material ^b , rads		1.0 x 10 ¹⁰
Maximum fast fluence in the superconductor ^b , n/cm ²		9.8 x 10 ¹⁸
Total heat load in the TF coils, kw		51.9
Maximum dpa in the copper stabilizer ^b dpa		7.9 x 10 ⁻³

^a Values are calculated at the end-of-life.

^b No credit is taken for the 2-cm carbon tile.

methodology established in the ITER Nuclear Design Guidelines. The primary uncertainty in these calculations is in the "diffusive" inventory. Using the data from Jones and Gibson (based on arc-cast Be outgassed in vacuum at 900°C) for the temperature range of 400-900°C, the calculated diffusive inventory is very small. The problem comes in extrapolating their results to lower temperatures. It has been observed that Be specimens (containing 1-3 wt.% BeO) which have been bombarded by deuterons release tritium very slowly from room temperature to about 350°C. As the two data sets do not overlap in temperature, it is not clear how to compare them and if there is any contradiction. Thus, a second calculation was performed in which the fraction of multiplier mass below 350°C was assumed to retain all of the tritium generated. This led to the 400g estimate in the footnote of Table 7-7. While this may be very pessimistic, it is adopted at this time until longer-time data for higher-purity Be samples are generated to clarify the tritium release characteristics of Be at $T < 350^{\circ}\text{C}$.

From tritium inventory considerations, it is desirable to introduce protium in the purge stream in quantities large enough to swamp the amount of tritium generated. This protium swamping effectively minimizes the amount of tritium adsorbed on the surfaces and dissolved in the bulk of the breeder (or multiplier). A ratio of protium flow rate to tritium generation rate of ~ 20 has been chosen to minimize tritium inventory in the breeder. The combined effect of this "protium swamping" and the stainless steel cladding means that the tritium in the gas phase is mostly in the HT molecular form.

With tritium in the HT form, there is a greater potential for tritium permeation through the breeder cladding. However, as stainless steel clad Be plates surround the breeder plates and the Be purge design is such that the HT partial pressures in the breeder purge and the Be purge are about the same, there is very little driving force for tritium leakage out of the breeder plates. Also, at the Be/H₂O boundary, the stainless steel cladding temperature is quite low, yielding a very small leakage rate from the Be plates to the coolant.

With regard to system pressure and pressure drop, the permeability of the breeder was designed to be high enough to allow the purge inlet pressure to be no higher than 0.202 MPa (2 atm) at the inlet for a 0.101 MPa (1 atm) purge outlet pressure. This minimizes the amount of structure one would need to

Table 7-7 Tritium Inventory

Parameter	Li ₂ O Breeder	Be Multiplier
Mass, 10 ⁶ g	5.2	63.0
Tritium generation rate, g/day	84.6	1.3
T _{min} , °C	400	100
T _{Max} , °C	800	600
Grain diameter, μm	20	20
Maximum P _{HTO} , Pa	7	-
Maximum P _{HT} , Pa	63	63
Inventory, g (wppm)		
Diffusion	2.9 x 10 ⁻³ (5.6 x 10 ⁻⁴)	1.6 x 10 ⁻⁴ (2.5 x 10 ⁻⁶)*
Desorption	4.0 x 10 ⁻³ (7.6 x 10 ⁻⁴)	
Solubility	1.31 (0.252)	0.254 (4.0 x 10 ⁻³)
Gas Phase	9.9 x 10 ⁻³ (1.9 x 10 ⁻³)	0.144 (2.4 x 10 ⁻³)
Total	1.3 (0.26)	0.40 (6.4 x 10 ⁻³)

* These calculations assume relatively pure (<< 1 wt.% BeO) Be. For conditions of ≥ 1 wt.% BeO, as much as 400g of tritium might be trapped in the Be.

contain higher pressure systems and also minimizes the pumping power. For monolithic 80% dense breeder microstructures, the permeability is much too low. However, by introducing cylindrical holes in the breeder of ~ 0.4 mm in diameter every cm^2 of breeder cross-section, the permeability can be raised to the desired value consistent with a pressure drop of 0.1 MPa (1 atm). The contribution of these holes to the overall breeder porosity is negligible ($\ll 1\%$). Also, the holes can be either fabricated along the centerline of the breeder or as semi-circular notches at the breeder/clad interfaces. From a fabrication and in-reactor, thermal-stability viewpoint, it would be preferable to have semi-circular notches at the breeder/clad interface.

Table 7-8 summarizes the design and operating conditions for the helium purge system for both Li_2O breeder and Be multiplier. The blanket is composed of 189 1m x 1m modules. While the breeder requires purge holes because of its high tritium generation rate, the Be multiplier with its low generation rate does not require such holes. For each module, a single inlet tube is required for the breeder layers and the multiplier layers.

The relationship between pressure drop (ΔP in MPa) and helium flow rate is:

$$\Delta P = \dot{n}RT\mu l / (PK_e) \quad (1)$$

where $\dot{n}$ = He molar flow rate per unit breeder cross-sectional area (moles/ $\text{m}^2 \cdot \text{s}$), R = Gas constant ($\text{MPa} \cdot \text{m}^3 / \text{mole} \cdot \text{K}$), T = Temperature (K), μ = He viscosity ($\text{MPa} \cdot \text{s}$), l = Flow length (m), P = Average He pressure (MPa), K_e = Permeability (m^2). The desired pressure drop is ≤ 0.10 MPa, and the desired system pressure is ≤ 0.15 MPa. The helium flow rate is determined by the maximum allowable tritium partial pressure in the purge and the tritium generation rate to be 0.210 moles/ $\text{m}^2 \cdot \text{s}$ for the breeder and 4.76×10^{-4} moles/ $\text{m}^2 \cdot \text{s}$ for the Be multiplier. Solving Equation 1 for the desired permeabilities gives $K_e = 4.1 \times 10^{-12} \text{ m}^2$ for the breeder and $K_e = 4.1 \times 10^{-14} \text{ m}^2$ for the Be multiplier.

The pumping power (W_p in MW) is determined from the equation

$$W_p = A_b \dot{n}RT \ln(P_i/P_o) \quad (2)$$

$$\approx A_b \dot{n}^2 R^2 T^2 \mu l / P_o^2 K_e \text{ for } \Delta P/P_o \leq 1$$

Table 7-8. Summary of Design and Operating
Conditions for Purge Flow for 189
(1 m x 1 m) Blanket Modules

Parameter	Li ₂ O Breeder	Be Multiplier
Tritium generation rate, g/day	84.6	1.3
Average temperature, °C	600	400
Grain diameter, μm	20	20
Pore diameter, μm	4.8	4.8
Density, % TD	0.8	0.8
Purge Holes:		
Diameter, mm	0.36	-
#/cm ²	1	-
Permeability, 10 ⁻¹² m ²	4.1	4.1 x 10 ⁻²
Helium pressures, MPa		
Inlet	0.202	0.123
Outlet	0.101	0.101
Hydrogen pressures, Pa		
Inlet	692	662
Outlet	630	630
Maximum tritium pressures, Pa		
HT	63	63
HTO	7	-

where

A_b = total breeder cross-section area perpendicular to flow (m^2)

P_i, P_o = inlet, outlet pressures

Table 7-9 summarizes the pumping powers required to purge the breeder and multiplier regions of the blanket, along with pressure drops and flow rates. Notice that the pumping powers are quite small because of the relatively low flow rates required and the relatively high breeder permeability designed into the system.

The disruption vaporization and melting of graphite and stainless steel first wall candidates are calculated for ITER design using A*Thermal Code. The disruption time for the first wall exceptional case is assumed to be 3 ms according to the design guidelines. The peak energy density for the pulsed inductive ignition and the steady-state current drive designs are about 380 J/cm^2 and 460 J/cm^2 respectively. Table 7-10 shows the amount vaporized and melted from both graphite and stainless steel materials for several energy densities. Interpretation of these results provide estimates of material erosion rates and consequently the expected reactor component lifetime from the disruption view point. The evaporative erosion rate for stainless steel is roughly about 50 micron/disruption for the pulsed operation and about 60 micron/disruption for the steady state design. With vapor shielding effect, these erosion rates are reduced to about 12 and 17 micron/disruption respectively. The number of exceptional disruptions on the outboard wall during the entire lifetime of the reactor depends on the number of cycles for both physics and engineering phases, and it is not fully determined yet. This number is assumed in this study to be about 30 exceptional disruptions for the outboard wall. The erosion losses are then calculated to be 1.5 mm for stainless steel and about 0.6 mm for carbon. If vapor shielding proved to be effective, these thicknesses may be reduced to .4 mm for steel and .1 mm for carbon. Thus, it is recommended that the outboard stainless steel wall thickness should be greater than 3 mm based on this disruption analysis. The inboard wall may not use bare stainless steel because of the excessive erosion rates. This calculation does not assume any losses from the melt layer formed

Table 7-9 Purge Flow Pressure Drops, Flow Rates,
and Pumping Power

Parameter	Li ₂ O Breeder	Be Multiplier
Helium pressure		
Average, MPa	0.152	0.112
Drop, MPa	0.101	0.022
Helium flow rate per module		
moles/s	3.78×10^{-3}	1.07×10^{-4}
cm ³ /s	135	4.9
Hydrogen flow rate per module		
moles/s	2.58×10^{-5}	1.55×10^{-7}
Helium velocity in a 1 cm diameter pipe, cm/s	43	15
Total pumping power, kw	3.45	0.022

Table 7-10
Calculated Disruption Erosion During Lifetime for ITER First Wall
Candidate Materials

(disruption time = 3 ms)

Energy Density (J/cm ²)	Material		Carbon	
	Stainless Steel			
	Evaporation* (μm)	Melting* (μm)	Evaporation* (μm)	Melting** (μm)
100	.02/.02	40/40	.003/.003	-----
200	9.30/.60	90/70	2.80/.22	-----
300	28.70/4.5	110/90	10.00/1.40	-----
400	51.00/12	110/100	18.50/4.10	-----
450	62.70/16.5	120/100	22.80/5.80	-----
500	74.60/21.2	120/100	27.20/7.60	-----

*Without/with vapor shielding.

**Carbon does not melt.

on the stainless steel surface. If the melt layer, or part of it, is to be lost during the disruption the erosion rate can be much higher than this value.

7.3 KEY TECHNICAL ISSUES

The water-cooled, solid-breeder blanket developed for ITER makes use of current technology and data bases. However, several key technical issues require further investigations to provide the missing data and to insure a satisfactory blanket performance. The issues for this class of blanket are the following:

- Lithium-6 burnup and irradiation effects impact on the properties of the solid breeder materials are not completely known. During operation, the middle section of the solid breeder plate will have a compressive stress and a high temperature which result in changes in the microstructure of the solid breeder materials. The main concerns are the changes in the tritium release characteristics and the thermal conductivity of the solid breeders. Also, the change in the gap conductance at the breeder-to-structure interface is another important factor for the blanket performance.
- Irradiation data base for austenitic steel at a low temperature (50 to 350°C) at an appropriate He to dpa ratio need to be determined. This issue applies to all blanket and shield options for ITER.
- Stress corrosion cracking of austenitic steel in water blanket options under fusion environment needs to be assessed; especially, the design solutions developed to mitigate stress corrosion in light water reactor materials which involve: a) stringent water chemistry control, b) modification of residual stress distributions in weldments, and c) more resistant materials, e.g., type 316 NG, need to be assessed for fusion conditions.
- The ability of the blanket to accommodate a large change in the neutron wall loading greater than 50% of the nominal value needs to be addressed if ITER design requires this feature. The main concern is the maximum temperature of the breeder materials will exceed the design temperature

range during the transient period.

7.4 CONCEPT EVALUATION

The blanket evaluation process used fourteen criteria to judge the different blanket options for ITER. A brief statement is given in this section to provide comments on the main issue for each criterion.

- The cost of the water-cooled solid-breeder blanket option is dominated by the beryllium multiplier cost (18 cm layer). The cost of the breeder is about 10% of the beryllium multiplier. The other cost item is the shield cost which is mainly steel. The blanket scores 3 because it has a small impact on the overall reactor cost, less than 5%.
- The R&D requirement for this concept is related to the predictability of the solid breeders performance in a fusion environment which is a current activity in the fusion community. The results from this activity are expected to provide adequate input for ITER design process. The blanket scores 2 according to the evaluation criteria.
- Solid breeder fabrication has been demonstrated, and current industrial capabilities are adequate for ITER requirements. The blanket scores 2 based on the number of fluid systems.
- The total tritium inventory is less than 2 g which reduces the concern about accidental tritium release from ITER blanket system. A score of 2/3 is given to the blanket.
- The energy source in the blanket after LOCA or LOFA conditions is very small. The concern about activated material releases are dominated by the inboard shield, not the blanket. The blanket scores 2-3 in this category.
- The blanket system is designed to be well below the design guidelines for the routine tritium release. The blanket scores 3 in the routine tritium release category.
- The use of the water-cooled solid-breeder blanket in the physics phase of

ITER will cause the blanket to operate at very low temperature (neutron wall loading and neutron fluence are very small values) which will cause the generated tritium to accumulate in the blanket. This causes the blanket to score 1-2 in the category of compatibility with ITER mission.

- No special requirements to operate or maintain the blanket are required. The blanket scores 2 based on the 1 to 3 months required to change a blanket sector.
- The blanket produces adequate tritium for ITER operation. The blanket scores 3 in the tritium breeding category.
- The test modules may contain a liquid metal which requires extra caution in the design. The blanket scores 1-2 because of the safety concerns related to lithium-water reactivity.
- The decay heat in all the blanket options for ITER is dominated by the inboard section of the reactor. The blanket scores 2 because of the steel decay heat in the inboard section at LOCA conditions.
- The tritium extraction from the helium purge flow uses current technology. The blanket scores 2 for tritium extraction because a tritium extraction at ITER conditions does not exist.
- The waste disposal class for the blanket is class C which results in a score of 2 for the blanket.
- The blanket uses a low temperature water-coolant which is not relevant to power reactors. A score of 1-2 is given for power reactor relevance category.

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CHAPTER 8

HELIUM-COOLED SOLID BREEDER BLANKET FOR ITER

CHAPTER 8

HELIUM-COOLED SOLID BREEDER BLANKET FOR ITER

8.1 GENERAL DESCRIPTION

A helium-cooled solid breeder blanket has been proposed for ITER with the idea of developing a low-risk blanket option with reactor relevant materials and configuration. Some of the more attractive features associated with this proposed blanket option are: 1) There is a significant design margin since the wall loading ($1\text{--}2\text{ MW/m}^2$) and fluence (3 MW-yr/m^2) of ITER are substantially lower than the wall loading (up to 5 MW/m^2) and fluence (up to 20 MW-yr/m^2) of commercial reactor designs and only part of the allowable solid breeder temperature window needs to be used. 2) There is an expanding data base available from the number of past and on-going solid breeder experiments carried out internationally. It is expected that data will be available from these experiments in fission reactor for the fluence expected in ITER. 3) The solid breeder can be designed to operate at high temperature, providing valuable information applicable to reactor conditions, while the helium is kept at a moderate temperature and pressure for safety and reliability. 4) Since helium is a gas, the pressure increase for any temperature increase is minor compared to a saturated liquid. Thus, the helium can be run so as to optimize the structure temperature, and any long term power variation can be accommodated by adjusting the rate of flow and, hence, temperature of the helium without incurring any substantial pressure penalty. 5) Safety is expected to be a key factor for ITER. In this respect, the use of helium, an inert gas precluding any chemical reaction and corrosion, in combination with a low activation solid breeder, such as Li_4SiO_4 and Li_2O , is highly appealing.

Some of the key technical areas addressed in this work for the helium-cooled solid-breeder design include the achievable TBR, the coolant pressure requirement, the provision for producing the required temperature drop between the high temperature breeder and the lower temperature coolant and the helium containment and manifolding. The results of the design and analysis efforts indicate that all requirements can be met in a low-risk, reactor relevant blanket with many attractive features.

8.2 CONCEPT DESIGN

The design is based on previous studies^(1,2) and consists of a number of independent canisters, lying side by side in the poloidal direction, as shown in Fig. 8-1 for the outboard of the reference ITER design. A rod bundle configuration inside each canister is chosen for the design calculations. In this configuration, the solid breeder and multiplier

8 - 1 . Cross-Section of ITER Showing Canister Layout

are in well-defined volumes to enhance the predictability of their properties. They can also be placed in the same rods or in different adjacent rods, creating a near-homogeneous mixture which increases the neutronics performance. In addition, such a configuration provides the flexibility of varying individual rod sizes and compositions in different rows to allow for the exponential decrease in heat generation in the radial direction while maintaining the solid breeder at an acceptable temperature. Finally, a rod bundle configuration provides reasonable heat transfer and pressure drop and has been well studied in heat exchanger designs. Note that the rods are staggered to reduce radiation streaming in the main blanket region. However, the effects of radiation streaming in the plenum and manifolding must be accounted for in the design of the shield and manifolds.

Figure 8-2 shows a typical canister with the solid breeder and/or multiplier rods. In the design, the canister width is 33 cm and its length is 1.1m. The reason for using these small canisters is to maximize the coverage of available space on the outboard of the ITER reactor, where space allowances are required for test modules and penetrations. A single main helium flow is shown in this design to reduce the manifolding complexity. The flow comes in along the side wall to provide for the cooling of the first wall and then enters the canister at the first wall and flows over the rod bundle before exiting the canister at the back. A porous mesh is provided along the side wall helium channel so that a stagnant layer of helium inside the mesh will act as an insulator between the cooler side wall flow and the hotter rod-bundle flow. Based on simple initial stress calculations, both the side walls and end walls are 5 mm thick with thin ribs located about 10 cm apart providing the support for each canister to be structurally independent.

For simplicity, a single rod configuration is shown inside the canister. However, the canister could easily be divided into two or three regions with different rod configurations in each, as required by neutronics or other consideration. The smaller rods are Be rods which are clad to preserve the integrity of the Be and to provide a barrier against contamination of the main He flow from tritium produced in the Be. The larger rods consist of a clad solid breeder inner cylinder with a clad sphere-pac Be/He annular region on the outside, as shown in Fig.8-3.

The use of Be particles and He in the thermal resistance gap between the SB and He is proposed because it offers several advantages, including the following: 1) The thermal resistance between the solid breeder material and coolant can be selected by properly tailoring the bed characteristics (porosity, type and size of particles, type of gas). 2)

8 - 2 . Helium-Cooled Solid Breeder Blanket Canister for ITER

8-3. Solid Breeder/Multiplier Rod Configuration

The increased gap size, as compared to pure helium, allows for better uniformity and predictability of the thermal resistance. 3) The use of beryllium in a near-homogeneous mixture enhances the tritium breeding capabilities of the design. 4) Increased control over the solid breeder temperature (through both passive and active means) permits operation over a wider range of power densities and transient conditions without violating temperature constraints. 5) Swelling can be partially accommodated by expansion into the bed void fraction. 6) Tritium generated in the beryllium multiplier is more easily removed, perhaps as a part of the basic solid breeder purge system. 7) An additional purged barrier is provided against tritium permeation to the coolant. The effects of the combination of SB, Be and structural volume fractions in this configuration on TBR are addressed below.

A complete list of design parameters is given in Table 1 at the end of this section. Some of the more important parameters are discussed in more details in Sections 8.2.1 to 8.2.7.

8.2.1 Neutronics

A series of 1-D calculations using toroidal and poloidal geometry models were first performed to discuss the impact of design options on tritium breeding. A 3-D calculation using the Monte Carlo code was performed for the final design to provide a more accurate evaluation of the net TBR.

The materials considered for the neutronics calculations are Li_4SiO_4 (based on the discussion in Ref.(3), Be and PCA as breeder, multiplier and structure respectively. Figure 8-4 shows the outboard arrangement and the corresponding material volume fractions are shown in Table 8-1. The breeder region was initially divided into three neutronically homogeneous zones. The first zone (Breeder 1), which is the closest to the first wall, has the maximum Be to SB ratio of the three zones (85/15). The second breeding zone (Breeder II) has a Be/SB mix ratio of 30/70, while the last zone (Breeder III) consist of SB rods only. The change in Be/SB mix in the rod is achieved by decreasing the amount of Be in the Be/He region and its thickness also to satisfy the thermal conductance requirement. For these initial calculations the TIBER water-cooled inboard shield configuration was adopted, with a total thickness of 55 cm. The design and optimization of a helium-cooled inboard shield should be done later as a trade-off between the attractive safety feature of having an inert gas and the expected cost penalty of requiring more shielding material to account for the helium voids. For the present, since the inboard shield will require a separate cooling circuit for any blanket, it is reasonable

Table 8-1

List of Major Parameters for Helium Cooled
Solid Breeder ITER Blanket Design (outboard only)

Materials

Coolant:	Helium
Breeder:	Li ₄ SiO ₄
⁶ Li Enrichment:	90%
Neutron Multiplier:	Be
Structure:	PCA, 25% CW
Reflector:	Carbon
Shield	Stainless steel

Configuration & Dimension

Breeder/Multiplier rods in canister

Canister:	Length	1.1 to 1.3m
	Width	0.33m
	Number	480
Rod:	Diameter	0.041m
	Spacing Between Rods	0.002m
	Number per Canister	106
First Wall:	Thickness	0.005m
	Channel Width	0.003m
	Second Wall Thickness	0.002m

Design Parameters

Coolant:	Inlet/Outlet Temperature	50°C/300°C
	Inlet/Outlet Pressure	1.5MPa/1.4MPa
Breeder:	Minimum/Maximum Temperature	325°/475°C
Structure:	Maximum First Wall Temperature	275°C
	Minimum/Maximum Temperature at:	
	Coolant Interface	50°C/300°C
	Breeder Interface	325°C/475°C
	Multiplier Interface	150°C/475°C

Tritium Breeding Ratio:

1-D Poloidal Model with 100% coverage	1.6
1-D Toroidal Model with breeding in outboard only	1.3
Net 3-D including all space losses	1.0

Table 8-1 (Cont'd.)

Maximum Nuclear Heating Rates (based on 1-D, homogenized-region calculations)

Breeder	16W/cm ³
Coolant	0
Multiplier	16W/cm ³
Structure (First Wall)	9W/cm ³
Energy per Fusion Neutron:	19.7 MeV

First Wall/Blanket Description:

Total Thickness	1.1 m
Manifold Thickness	0.1 m
Coolant Total Pressure Drop	0.1 MPa
Coolant Pumping Power	18 MW
Coolant Velocity (Max/Ave):	
First Wall	62/47 m/s
Canister	31/20 m/s
Coolant Flow Rate	1.1 x 10 ⁶ kg/h

Tritium Removal

Method:	Proven method for tritium processing from helium		
Steady-state Blanket Inventory:	Diffusion:	0.4g	
	Solubility:	1g	
	Surface Adsorption:	0.1g	

Purge Gas Parameters:

Gas Material	Helium
Inlet/Outlet Temperature	~400°C
Pressure	14 MPa
Mass Flow Rate	590 kg/h
Average tritium production period	0.5g/FPY
HT partial pressure	10 Pa

Heat Transport System

Helium Side

Pressure Drop	0.1 MPa
Pumping Power	2.2 MW

Water Side

Maximum Pressure	0.1 MPa
Inlet/Outlet Temperature	20°C/40°C
Tritium Barriers	Oxide layer probably

LOCA

Method to Accommodate	Initial calculations indicate passive method but needs verification
-----------------------	---

Table 8-1 (Cont'd.)

Maximum Material Temperature

Structure	725°C
Breeder	725°C
Multiplier	725°C

Power Variation Capability

Method	Passive for power increase He flow rate reduction for power decrease
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Minimum/Maximum Change from Average Conditions 50%/50% at least

Compositions and Dimensions for Neutronics Calculations

<u>Region</u>	<u>Thickness (m)</u>	<u>Structure</u>	<u>Volume Fraction</u>		<u>Helium</u>	<u>Shield Material</u>
			<u>Breeder</u> (80% TD)	<u>Multiplier</u> (90% TD)		
First Wall	0.02	0.4			0.6	
First Breeder	0.5	0.09	0.07	0.46	0.38	
Second Breeder	0.1	0.16	0.37	0.14	0.32	
					0.91	
Plenum	0.1	0.09				
Reflector	0.1	0.05			0.1	0.85 C
Shield	0.3	0.4			0.1	0.5 B ₄ C

Nuclear Responses in the TF Coils (@ 2.5 FPY)

Maximum Fast Neutron Fluence in the Superconductor	$2.8 \times 10^{18} \text{ n/cm}^2$
Maximum Atomic Displacement in Copper Stabilizer	$8.63 \times 10^{-4} \text{ dpa/FPY}$
Maximum Dose to Electrical Insulator Polyimide	$3.55 \times 10^9 \text{ rad}$
Maximum Nuclear Heating in the Superconductor	$1.36 \times 10^{-3} \text{ W/cm}^3$
Nuclear Heating in the Superconductor at Midplane	$0.26 \times 10^{-3} \text{ W/cm}^3$
Maximum Nuclear Heating in the Magnet Case at Midplane	$4.72 \times 10^{-3} \text{ W/cm}^3$

Max. Local After Heat (W/cm³)

		<u>First Wall</u>	<u>Breeder</u>
1 h after	Shutdown	0.1474	0.04947
1 Day after	Shutdown	0.1437	0.0482
1 Month after	Shutdown	0.1206	0.0404

Mass of Materials

Breeder	$2.8 \times 10^4 \text{ Kg}$
Multiplier	$8.8 \times 10^4 \text{ Kg}$
Structure	$1.4 \times 10^5 \text{ Kg}$
Shield	$3.4 \times 10^5 \text{ Kg}$

to assume that the inboard shield, away from the lithium carrying test modules in the outboard, can be water-cooled.

Fig. 8-5 shows the tritium breeding ratio (TBR) as a function of the first breeder region thickness for a constant total breeder thickness of 60 cm and a 90% ^6Li enrichment. These results were obtained from 1-D neutronics calculations using a toroidal arrangement, with breeding in the outboard only. The TBR increases as the thickness of Breeder I is increased, until saturation is reached at a thickness of about 50 cm. More importantly, the results illustrate the effect of the inboard shield configuration on the outboard TBR. The three cases shown correspond to: 1) the water-cooled TIBER design; 2) a modified version of this design with the first 2.8 cm of water replaced by 1.4 cm of water and an additional 1.4 cm of Be and 3) another modified version where the first 2.8 cm of water is replaced by 2.8 cm of helium. It is interesting to observe the effect of progressively removing the water and, in effect, increasing the neutron reflection to the outboard. The maximum TBR of 1.35 is obtained for the case where helium is substituted. This is another important factor to be considered for the comparison between a helium-cooled to a water-cooled inboard, particularly when considering that 3-4 cm of helium might be all that is needed to cool the inboard.

It is not clear exactly how to derive a net TBR from the 1-D value obtained using the toroidal arrangement. The problem is exacerbated by the uncertainties introduced by loss of coverage due to test modules and penetrations and also by the fact that breeding can be done in the top and bottom parts of the inboard and in the divertor regions, particularly if a single null poloidal divertor design is assumed. As a result, the 3-D MCNP neutronics code was run assuming homogeneous regions for the case which maximizes the coverage, with breeding on the outboard, top flat region (assuming a single null divertor design), top corner and lower inner corner, but excluding the space limited 4m high inboard shield. The configuration is shown in Fig. 8-6. Notice that there is much more space in the top corner than in the inboard region and there is no difficulty in providing adequate shielding. Ease or difficulty of maintenance will depend on the actual configuration and maintenance scheme selected for ITER. For example, the commonly used scheme of removing the sector in order to replace a module will make the outboard and top corners comparable from maintenance standpoint. There is also adequate space in the top and bottom region that can be used for manifolding.

The TBR results are summarized in Table 8-2. They are quite conservative since the neutronics calculations assume a uniform source and no plasma MHD shift both of

8-5. TBR in Outboard Calculated Using a Toroidal Arrangement as a Function of Breeder I Thickness for a Constant Total Breeder Thickness of 60 cm.

8-6. Configuration for 3-D Neutronics Calculations

which would increase the outboard contribution to the TBR. The results are encouraging and show flexibility as a number of different regions can be included depending on the TBR requirements. A TBR of ~ 1.27 can be obtained without breeding in the space-limited inboard region. Void space for penetrations is a modest penalty since there is no substantial loss of neutrons, and space required for test modules might also not be a penalty if even partial breeding from the modules is taken into account. However, even if an assumption is made that out of the 217 m^2 of the outboard, 29 m^2 is not available for breeding and that another 3% coverage is lost due to plenum and structural requirements at the canister end walls, an overall net breeding ratio in excess of unity is obtainable. The margin in excess of unity can be used to simplify the design, for example, by eliminating breeding in the access-limited lower corner region.

Table 8-2

TBR results from 3-D Neutronics Calculations for ITER He Cooled Solid Breeder Blanket

Region Contribution to TBR:	<u>TBR</u>
Outboard Blanket	1.01
Top Flat Region	0.082
Top Corner	0.064
Inboard Shield (no breeding)	0.0
Lower Inner Corner	0.067
Lower Divertor	0.06
Total TBR	1.27

The above analysis and discussion indicates that even by being conservative with this initial configuration, we can comfortably design for at least a 3-D net TBR of unity even when including space loss penalties and without breeding in the space-limited inboard region. The flexibility of the configuration allows a trade-off between increasing the number of breeding regions to meet higher TBR requirements and increasing the manifolding and maintenance complexity for less accessible regions.

8.2.2 Thermal-hydraulics

Fig.8-7 shows the canister helium pressure drop, ΔP , as a function of the inlet pressure, $P(\text{in})$, for different He temperature rises, ΔT_{He} , for the case with 4 cm tube and 2 mm spacing. Even if the pressure drop is doubled or tripled to account for the He flow outside the canister (as our calculations indicate), a wide range of $P(\text{in})$ and ΔT_{He}

8-7. Canister He Pressure Drop as a Function of the Inlet Pressure for Different Temperature Rises

choices is available. As shown in Fig.8-8, the corresponding pumping power requirement is less than 20 MW for all cases with $P(\text{in}) \geq 10 \text{ atm}$ and $\Delta T_{\text{He}} \geq 250\text{K}$.

A $P(\text{in})$ of 1.5 MPa and a ΔT_{He} of 250K are reasonable design choices; the required pumping power for the complete helium circuit, including the piping and heat exchanger is about 22 MW. The low stress caused by the pressure will facilitate the design of canister structural configuration.

8.2.3 Power Variation Allowance

This blanket is very flexible in allowing for operation at 50% below or above the design power level. Both passive and active ways can be employed to allow for power variation. The passive way makes use of the large SB temperature window whereas the active way makes use of the helium temperature which can be varied by changing the inlet temperature and/or flow rate without any significant pressure penalty.

Fig. 8-9 illustrates the flexibility of the design in this regard for a canister located at the highest wall loading location (1.9 MW/m^2). The figure shows the maximum solid breeder temperature in the first row as a function of the thickness, r , of a uniform Breeder region 1. The minimum temperature of the solid breeder at r is 600 K. Any increase in the power level is allowed by letting the solid breeder temperature in the first row rise to the maximum allowable temperature, 1000 K. Thus, for a 20 cm thick Breeder region 1, as an example, the maximum SB breeder temperature in the first row is 760 K and the allowable increase in power is 50%. There is a trade off here between the allowable power variation and the number of uniform regions per canister and the number of canister configurations also, since the wall loading varies poloidally. Our calculations indicate that, to allow for a 50% increase in power, 2 or 3 different canister regions and 3 or 4 different canister configurations are required. Note that, any decrease in power can be easily accommodated with no apparent penalty by reducing the helium coolant flow rate, thereby increasing its temperature.

8.2.4 Reliability

The reliability of the blanket is often thought to decrease as the number of rods increases. The overall consequence of failure, though, does not necessarily follow the same trend since the reduced consequences of failure per rod can outweigh the increased probability of rod failure in designs with a larger number of smaller rods. To address this issue, the probability of achieving a given level of consequences was investigated in

8-8. Canister He Pumping Power as a Function of the Inlet Pressure for Different Temperature Rises.

8-9. Maximum Solid Breeder Temperature and Power Level Allowed

different designs. The calculations performed assume that tritium contamination is one of the most serious consequences of rod failure.

Fig. 8-10 shows the probability of tritium release for different solid breeder volumes from which tritium escapes due to weld failure. Three cases are considered corresponding to rod diameters of 28, 40 and 56 mm. Note that the number of rods for a given canister varies inversely with the square of the rod diameter. The total length of welds depends both on the size of rods and the number of rods. A probability of 0.005 is arbitrarily assumed for a 1 mm crack per m of weld. For this case, depending on whether the critical amount of tritium release is less than or greater than the content of $1.5 \times 10^{-6} \text{m}^3$ of solid breeder, the most reliable design is with the smaller or greater number of rods respectively. The 0.005 probability for a 1 mm crack per m of weld used here does not affect the conclusion as a different value will merely shift the point at which more rods are more reliable than fewer rods.

In summary, the rod design is very robust against failures. It provides multiple barriers against contamination. With the inert coolant, the consequences of failure are not catastrophic, at worst leading to contamination of the main coolant. The rod design also dilutes the source term, and increasing the number of rods can actually decrease the probability of exceeding contamination limits.

8.2.5 LOCA

Figs. 8-11 and 8-12 summarize the initial calculations done for LOCA for the first wall and the Li_4SiO_4 solid breeder in the first row of rods. Under adiabatic conditions, in time both the first wall structure and solid breeder would exceed their maximum allowable temperatures resulting in loss of investment. If radiation to the cooled inboard is assumed the first wall temperature stays under half its melting point of about 1700 K. Including the effect of free convection slightly increases the first wall temperature but only retards the time for the solid breeder to reach its maximum allowable temperature of 1000 K (based on sintering).

A possible solution would be to connect a helium cover gas in the building to a heat exchanger to create a draft which could be enough to increase convection and keep the solid breeder rods and first wall at acceptable temperature levels, with no loss of investment. However, key issues must be addressed, such as whether it is feasible or desired to keep the cover gas at 1.5 MPa or whether the cover gas could be pressurized only when a LOCA occurs. Further analysis is required in this area.

8-10. Probability of Releasing Tritium from more than a Given SB Volume, V , for Different Rod Diameters (i.e. different number of rods).

8-11. First Wall Temperature Following a LOCA.

8-12. Solid Breeder Temperature Following a LOCA.

8.2.6 Helium leakage to Plasma

It is very difficult to accurately predict the helium leakage to the plasma, but some observations can be made. Extrapolation of the data from Fort St. Vrain HTGR⁽⁴⁾ indicate that the leakage rate through welds to the plasma is only about 1% of the helium generation rate in the plasmas based on an estimated 4700 m total weld length exposed to plasma. However, estimates assuming cracks through the whole thickness of the first wall, could increase the above number substantially. The difficulty of accurate calculations is exacerbated by the fact that the number and size of cracks escaping inspection are not known and that mechanisms such as crack growth must be considered. Including this consideration in the design process can help alleviate the problem. For example, crack propagation is more problematic in welds and the number of welds in contact with the plasma can be minimized. Note also that any reasonable amount of helium leaking to the scrape-off region would be swept away in the divertor and would probably provide a beneficial effect by cooling the plasma edge.

8.2.7 Tritium Inventory

The steady state tritium inventory in the Li_4SiO_4 breeder is estimated to be very low, 1.5g. Radiation effects can increase this inventory. However, even if a factor of 100 increase results from radiation effects, the tritium inventory would still be significantly smaller than in other candidate concepts. The concern regarding tritium accumulation in Be is lessened in this design since the Be is also purged. However, the effect of a BeO layer and of its thickness on the tritium inventory inside the Be spheres should be considered as it could freeze much of the tritium inside the Be.

8.3 TECHNICAL ISSUES

A key issue has been the predictability of the gap conductance between clad and breeder. A novel idea has been proposed, whereby a mix of Be particles and helium is used in the gap, increasing the thermal conductance and allowing for larger gap dimensions. In this concept, the design is much more forgiving, and compared to earlier gas-filled gaps, there is a significant improvement in the reproducibility and predictability of the gap conductance. Such a concept, however, needs to be verified experimentally, first in a simple laboratory experiment and then in a more prototypical environment where irradiation effects would be included.

Irradiation effects on solid breeder material properties, in particular those affecting the tritium transport, need also to be addressed. However, the estimated

tritium inventory in the whole blanket is low (about 1.5 g) and uncertainties of up to a factor of 100 can be tolerated.

Helium containment and manifolding are other issues. It is difficult to accurately predict the helium leakage to the plasma and further effort is required in this area to find out the extent of the problem, and, if required, to develop acceptable solutions. The difficulty associated with helium manifolding is linked with the typically large piping dimensions. This difficulty can be minimized by optimizing the configuration arrangement and maximizing the use of the outboard for breeding while excluding breeding in the space-limited inboard.

8.4 CONCEPT EVALUATION

The He-Cooled SB blanket looks very attractive. It meets all the absolute ITER blanket requirements shown in Section 2.2. It also fares very well in the preliminary ITER blanket and shield evaluation (as tabulated in the summary section of this report) done by the ITER Nuclear Group based on the criteria discussed in Section 2.2. Among the designs that have received the largest effort and are best understood, namely the aqueous salt, the water-cooled SB and the helium-cooled SB, this concept appears to receive a most favorable ranking.

As shown in the table of the summary section, the impact of the blanket cost on the overall reactor cost is about the same for all blanket concepts. The R&D requirements prior to ITER are also shown as being the same for the different concepts. It must be said here, though, that there is an increasing data base for solid breeders and that only a modest acceleration of pace from the base program is required to meet the ITER R&D requirements. The He-cooled SB design ranks about the same as the other designs in terms of design and fabrication complexity. The only exception is the aqueous salt blanket which has the advantage over all the other blankets in this category.

The He-cooled SB design ranks at the top from the safety standpoint. It is very attractive with regards to accidental and routine tritium release and activated product release. In fact, the design is quite robust regarding the consequences of any rod failure. First tritium permeation to the main flow is greatly reduced by having a purged clad Be/He annular region around the SB. Second, even if a rod failure occurs, the worse consequence is a minor tritium contamination of the main flow. Our calculations indicate that to reach the tritium concentration limit prescribed for water coolant, about 200 rods would need to fail. In addition, helium is inert and there is no concern regarding chemical and thermal reactions, such as between water and liquid lithium from a test

module. It also fares well based on waste disposal rating. Activation is minimized in the breeder by choosing Li_4SiO_4 or Li_2O as the solid breeder material.

This design is at a disadvantage in terms of the ease of change between phases. It is conceivable, however, that instead of changing the whole blanket area, the blanket canisters could be placed from the start and the tritium left to accumulate in the breeder over the low fluence of the physics phase. No blanket change would then be required at the end of the physics phase.

This concept has a ranking comparable to the other concepts in terms of maintenance, tritium breeding and decay heat response. Tritium extraction from helium is also being done routinely for experiments and extending the extraction to the scale of an ITER blanket is believed to be straightforward.

The helium-cooled solid breeder concept has another key advantage in that both the materials and configuration chosen for this concept are based on main line reactor relevant blankets. This is important particularly when looking at the fusion program in perspective. ITER is only a step in fusion development. A key part of ITER's mission is blanket testing. The cost of the testing program can be considerably reduced and the research value of the ITER project can be augmented if a reactor-relevant blanket such as the helium-cooled SB concept is used. Furthermore, there will be another fusion facility such as a demonstration reactor. The R&D required after ITER would be minimized by using a reactor relevant blanket in ITER which would provide information for the design of the next fusion facility.

In summary, the helium-cooled solid breeder is a low risk reactor relevant concept that meets all the absolute requirements for an ITER blanket. In addition, it is particularly attractive in terms of overall safety which will be a key factor for ITER.

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CHAPTER 9

HELIUM-COOLED LITHIUM COMPOUND SUSPENSION BLANKET

9.0

HELIUM-COOLED LITHIUM COMPOUND SUSPENSION BLANKET

9.1 GENERAL DESCRIPTION

A simple tritium breeding blanket has been developed for ITER. The blanket uses a dilute suspension of fine solid-breeder particles (Li_2O , LiAlO_2 , or Li_4SiO_4) in a carrier helium gas as the tritium breeder stream. This stream is processed outside the reactor for tritium recovery. Also, it can serve as the coolant for the blanket (self-cooled) or a separate coolant can be used. The self-cooled option is considered for the current analysis to simplify the reactor configuration. The blanket consists of a beryllium multiplier and carbon/steel reflector. A steel clad is used for all materials (Be and C) in the multi-layer configuration. The carbon reflector has the advantage of reducing the beryllium thickness used in the blanket for a specific tritium breeding ratio.

The current design uses helium as a coolant and carrier for the fine solid-breeder particles. The breeder particle size has to exceed a few microns (≥ 2 microns) to avoid sticking problems on cold surfaces. The helium gas pressure should be in the range of 20-30 atmospheres to carry the solid breeder particles through the blanket and the heat exchanger loop. The solid breeder concentration in the helium stream is 1 to 5 volume percent.

A high lithium-6 enrichment is required to produce a high tritium breeding ratio and to reduce the breeder concentration in the helium gas. At a lithium-6 enrichment of 90%, the local tritium breeding ratio for a multi-layer configuration is 2.0 based on a one-dimensional poloidal model. The total thickness of the helium stream is only 4 cm out of the 50 cm total blanket thickness. These dimensions can be changed to adjust the helium velocity or solid breeder concentration in the helium gas without significant impact on the blanket dimensions or the tritium breeding performance.

One version of the blanket module configuration is a lobe design, with the helium coolant flowing in the poloidal direction from the bottom to the top of the reactor. This configuration has a tritium breeding ratio of 1.6. At coolant inlet and outlet temperatures of 50 and 200°C, respectively, the maximum structural temperature at the first wall is 417°C. Tritium recovery is performed outside the reactor by diverting ~ 1% of the coolant to the

tritium recovery system. The estimated tritium inventory is less than 1 g in the helium stream.

This blanket concept has an important advantage which allows blanket operation to be switched between nonbreeding and breeding modes without hardware changes in the reactor. Also, blanket performance can be adjusted during reactor operation by changing the breeder concentration, lithium-6 enrichment, helium pressure, or helium velocity. The addition of μ -sized solid particles in the helium gas improves its heat transfer and transport properties. Similar gas-suspension coolants were proposed for fission reactors in the 1960's where models and experimental data were generated.

9.2 CONCEPT DESIGN

Two blanket configurations are considered for ITER. The first configuration is a simple multi-layer design with four helium purge streams. The first wall/blanket/shield design and optimization system (BSDOS)⁽¹⁾ was employed for this configuration to maximize the tritium breeding ratio and limit the total blanket thickness to 50 cm. The three beryllium zone thicknesses are allowed to vary. Table 9-1 gives the dimensions and the compositions of the different zones as calculated by BSDOS. The one-dimensional tritium breeding ratio of this configuration is 2.034.

The second configuration shown in Fig. 9-1 uses grooved beryllium plates with Li_4SiO_4 as the breeder particles. The helium gas at 30 atmospheres pressure carries the dilute particles and circulates around the blanket and heat exchanger loop. The solid breeder concentration in the helium stream is 1.5 volume percent. At a lithium-6 enrichment of 60%, the one-dimensional poloidal model gives a 1.6 tritium breeding ratio for the configuration shown in Table 9-2. The corresponding blanket energy multiplication is 1.52.

The sticking of the fine breeder particles on the wall surfaces is a potential concern for such systems. However, previous work on gas suspension coolants provides the following conclusions:

- Fine particles do not stick on hotter surfaces.

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Fig. 9-1 Li-Particulate Design Configuration

Fig. 9-2 Blanket Module for Li-Particulate Design

Table 9-1
Helium-Cooled Lithium Compound Suspension Blanket with Multi-Layer
Configuration^(a)

Zone	Composition, Thickness, cm	Volume Fraction
First Wall	0.7	Steel
Helium Breeder	1.0	0.05 Li ₂ O, 0.95 He
Back Wall	0.3	Steel
Multiplier	13.8	Be (0.8 DF)
Coolant Channel	0.2	Steel
Helium Breeder	1.0	0.05 Li ₂ O, 0.95 He
Coolant Channel	0.2	Steel
Multiplier	13.9	Be (0.8 DF)
Coolant Channel	0.2	Steel
Helium Breeder	1.0	0.5 Li ₂ O, 0.95 He
Coolant Channel	0.2	Steel
Multiplier	16.0	Be (0.8 DF)
Coolant Channel	0.2	Steel
Helium Breeder	1.0	0.5 Li ₂ , 0.95 He
Coolant Channel	0.2	Steel

(a) Li₂O has a 90% lithium-6 enrichment, Type 316 Stainless Steel is the structure material.

Table 9-2
Helium-Cooled Lithium Compound Suspension Blanket with Grooved Beryllium
Configuration^(a)

Zone	Thickness, cm	Composition, Volume Fraction
First Wall	0.6	Type 316 SS
Multiplier	1.0	0.5 Be, 0.5 Coolant
Breeder-1	27.0	0.37 Be, 0.37 Coolant, 0.047 Type 316 SS, 0.213 Void
Breeder-2	20.0	0.74 Be, 0.185 Coolant, 0.075 Type 316 SS
Hot Shield	10.0	0.815 Type 316 SS, 0.185 Coolant
Shield	30.0	Type 316 SS

(a) Coolant is He with 1.5 volume % Li₄SiO₄ with a 60% lithium-6 enrichment.

- Fine particles do not stick on colder surfaces if certain operating conditions are satisfied. For the current configuration with the following operating conditions: helium pressure = 20 atm, helium velocity = 3 m/s, solid volume fraction = 0.05, and helium temperature = 200°C, the particle size has to exceed 2×10^{-4} cm to avoid sticking.
- The conditions which particle sticking may occur have been theoretically and experimentally verified.

The other factor which effects the choice of the particle size is erosion of the steel walls which can be avoided by operating below the erosion threshold. Experimental results on stainless steel show a threshold at 10 m/s for a 40-micron particle size. This threshold is energy sensitive. It allows a higher velocity with smaller particle size as long as the particle energy is constant.

There is also a possible concern about magnetic field effects on the solid breeder particles behavior in the helium coolant. The concern is that the magnetic field may separate solid breeder particles from the helium coolant by driving the particles to the walls. Although high magnetic field strength can polarize even weakly magnetic materials, only a highly nonuniform field can exert strong magnetic forces on dipolar bodies. The magnetic field gradient through the tubes carrying the helium-solid lithium compound is too small to produce any significant forces on the particles. These magnetic forces, however, may cause a spatially nonuniform distribution of particles in helium flow. On the other hand, the particles should not stick to the walls. It is also important to know that the helium flow is in the turbulent regime and the fluid drag forces should be higher than both gravitational and magnetic forces. However, there may exist some parts in the reactor where the gradient of the magnetic field is relatively high. Then, analysis is needed to evaluate the impact of such effects.

There is a general concern about helium containment which applies to any helium-cooled concept. The flow of gases through orifices and cracks can be characterized by either viscous flow or molecular flow with a transition region exists between these two types of flow. Viscous type of flow will

dominate the gas flow at high pressures, while molecular type of flow will dominate the flow at low pressures. In reactor conditions where there is a large pressure difference between the coolant and the vacuum chamber. All flow conditions can exist and the actual leak rate must then be determined experimentally.

The only experimental data that is directly applicable for comparing theoretical prediction of the gas leak rate is the one given in Ref. 2. Based only on this limited data it is recommended to use molecular flow equations to predict the leak rate of helium through cracks in a vacuum first wall. This conclusion is contrary to a previous study in which it was recommended to use viscous flow to predict leak rates in fusion reactors based on the same experimental data^(2,3).

It is assumed in this study that the critical amount of helium gas leakage should be in the order of helium generation rate from the thermonuclear reaction. This in turn will determine the maximum crack size or the maximum number of cracks corresponding to this critical leakage rate.

Table 9-3 shows the helium leakage rates as a function of crack dimensions and the helium coolant pressure for a 1 mm thickness wall. These leakage rates are given in the percentage of the helium generation rate for a 700 MW fusion reactor operating at 300°C first wall temperature. Cracks of sizes 100-200 microns in thin structures may be difficult to detect by conventional nondestructive techniques. It can be seen that the leakage through 10-50 cracks of sizes 100-200 micron is equivalent to the helium generation rate for a reactor with helium coolant at 50 atm. Cracks existing in weldments or cracks created during operation (fatigue) in the complex structure could also have a serious impact on the successful operations of a helium-cooled reactor.

The analysis given above is only based on the few experimental data available. More experiments are needed to confirm the analytical predictions. If, however, other experimental data show that the helium leakage rate is controlled by viscous flow rather than molecular flow, the leakage rate can be even much higher than those predicted in this study.

Table 9-3
Helium Leakage Rate Per Crack (% of the Plasma Helium Generation Rate)

Crack Width/ Crack Opening (μm)	Pressure (atm)		
	1	5	50
400/40	2.18	10.9	108.8
200/20	.27	1.36	13.6
100/10	.03	.17	1.7
10/10	.002	.012	.12

9.3 KEY TECHNICAL ISSUES

The helium-cooled lithium compound suspension blanket makes use of current technology and data bases. However, some technical issues require further investigations to provide the missing data and to insure a satisfactory blanket performance. Helium leakage through micro-cracks and weldments is a main concern for this concept. Also, the erosion threshold of the type 316 stainless steel requires an experimental confirmation to insure a satisfactory blanket performance.

For the external coolant circuit, it is also important to avoid excessive erosion on external loop walls and circulating machinery. A preliminary design has been made to circulate the particles and gas mixture by using a centrifugal compressor. It was found that the blade tip speed is much over the 61 m/sec speed limit. Based on this observation, a loop design was developed that uses a multi-stage cyclone to separate the particles after they leave the blanket and reintroduce them into the circuit after the gas is transported by a blower. This will allow continuous gas and particle flow and reduce the amount of solid particles impinging on the turbo machinery.

9.4 CONCEPT EVALUATION

The blanket evaluation process used fourteen criteria to judge the different blanket options for ITER. A brief statement is given in this section to provide comments on the main issue for each criterion.

- The cost of this option is dominated by the beryllium multiplier and the steel shield. The blanket scores 3 because it has less than a 5% impact on the overall reactor cost.
- The R&D requirements for this concept are related to helium containment and steel erosion which can be satisfied with small facilities and low cost in less than three years. The blanket scores 2 according to the evaluation criteria.
- The blanket scores 2 in the design and fabrication complexity because it is relatively simple but requires special attention for the helium flow geometry.
- The total tritium inventory is less than 1 g in the helium stream which reduces the concern about accidental tritium release. The blanket scores 2/3 in this category.
- The energy source in the blanket for LOCA or LOFA conditions is very small. The concern about activated material release is dominated by the inboard shield, not the blanket. The blanket scores 2/3 in this category.
- The tritium system is designed to be well below the design guidelines for the routine tritium release. The blanket scores 3 in this category.
- The blanket has the capability to switch between nonbreeding and breeding mode without reactor changes which causes the blanket to score 3 in the compatibility-with-ITER-phased-operation criteria.
- The blanket scores 2 in the maintenance category because of the 1 to 3 months required to change a blanket sector.
- The blanket has a very high net tritium breeding ratio, it scores 3.
- The blanket has no potential for chemical or thermal reaction with other materials, it scores 3.

- The decay heat in all the blanket options is dominated by the inboard section of the reactor. The blanket scores 2 because of the steel decay heat in the inboard section at LOCA conditions.
- The tritium extraction from the helium purge flow uses current technology. The blanket scores 2 for tritium extraction because a tritium extraction at ITER conditions requires a demonstration.
- The waste disposal class for the blanket is Class C, which causes it to have a score of 2.
- The blanket uses low temperature helium which is not relevant to power reactors. A score of 1/2 is given for the power reactor relevance category.

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CHAPTER 10

HELIUM-COOLED, ^3He BLANKET

10.0 HELIUM-COOLED, ^3He BLANKET

10.1 GENERAL DESCRIPTION

In this approach terrestrial supplies of ^3He , rather than lithium, are used for tritium breeding. In order to assess the potential of the ^3He blanket, a reference configuration was adopted based on minor modifications to the modular helium-cooled blanket concepts considered in the Blanket Comparison and Selection Study (see Fig. 10-1). The chosen configuration assumes a stainless steel for structure and cladding and beryllium for neutron multiplication. The beryllium can be in the form of plates or rods. In order to minimize ^3He inventories, and ^3He and T leakage rates, the ^3He is placed separate from the main helium (^4He) coolant, flowing through the beryllium in a manner similar to the helium purge through a solid breeder. This arrangement provides good tritium-breeding and also allows for removal of the Be-bred tritium. The ^3He circulates only fast enough to recover the tritium. Both the breeding and cooling loops operate at about 5 MPa and over a temperature range of ~ 80-300°C.

The ^3He blanket offers an attractive option for tritium breeding in ITER. The concept exhibits good tritium breeding potential, low tritium inventories, and low tritium leakage rates. Moreover, the concept retains the desirable safety and operational features inherent in gas-cooled blankets; while, at the same time, eliminating concerns associated with lithium-breeder concepts, such as chemical energy release (liquid breeder) and stringent requirements on temperature control (solid breeder). There is no need to mechanically access the blanket in order to control breeding; the same blanket hardware can provide shielding or breeding controlled only by out-of-core addition of ^3He . The blanket performance is not adversely affected by substantial variations in power level. Finally, although the concept is not extrapolatable to power reactors (assuming we are limited to terrestrial supplies of ^3He), it will provide reactor relevant information on helium-cooled blankets and helium purge streams.

(A). Li_2O

(B.) Liquid Lithium

(C). LiAlO_2/Be

Figure 10-1 Modular Helium-Cooled Blanket Concepts Considered in the BCSS

10.2 CONCEPT DESIGN

Table 10-1 summarizes the concept performance parameters. Key aspects of the design are discussed in the following sections.

10.2.1 Tritium Breeding

The tritium breeding potential of the ^3He blanket configuration was assessed using the ONEDANT one-dimensional discrete ordinates transport code. Material cross-sections were produced using the TRANSX-CTR code and the MATXS5 cross-section library, with a P_3/S_8 approximation. The one-dimensional cylindrical geometry and zone material compositions are given in Table 10-2.

Table 10-2
ONEDANT Geometry and Zone Compositions for Tritium Breeding Calculations

Zone	Thickness (cm)	Composition
1 (first wall)	1.3	17.5% HT-9, 9.3% ^4He
2 (inlet plenum)	3.0	14.1% HT-9, 47.5% ^4He
3 (breeder)	4.0	21.7% HT-9, 11.1% ^4He , 61% Be + ^3He
4 (breeder)	61.7	22.2% HT-9, 11.4% ^4He , 66.3% Be + ^3He
5 (outlet plenum)	15.0	19.3% HT-9, 80.7% ^4He

Tritium breeding was examined varying the relative compositions of beryllium and ^3He in zones 3 and 4. The total volume of beryllium and ^3He was kept constant. The results are displayed in Fig. 10-2, indicating that tritium breeding ratios above 1.5 are attainable when the ^3He volume is about 10% of the beryllium plus ^3He volume. This corresponds to an overall ^3He volume fraction in the blanket of 6%. It is noted that the reference configuration was not optimized for tritium breeding and, therefore, some improvements in breeding potential are to be expected.

Table 10-1
ITER First Wall/Blanket/Shield Performance Parameters List

First Wall/Blanket Shield Concept: ^3He Blanket

General Description:

	<u>Inboard Form</u>	<u>Outboard Board</u>
- Materials		
Blanket		
Coolant (^4He)	Gas	Gas
Breeder (^3He)	Gas	Gas
^6Li Enrichment (NA)		
Neutron Multiplier (Beryllium)	Metal	Metal
Structure (PCA)	Metal	Metal
Reflector		
Shield		
Shield(s) (FeB or W)	Metal	Metal
Structure (PCA)	Metal	Metal
Coolant (He)	Gas	Gas

(Material form: metal, powder, coating, cladding, etc.)

- Design Configuration (see figure)

Design Parameters:

- Coolant	
Inlet/Outlet Temperature, C/C	80/305
Inlet/Outlet Pressure, MPa/MPa	5/4.9
- Breeder	
Minimum/Maximum Temperature, C/C	80*/325
Maximum Pressure, MPa	5
- Structure	
Maximum FW Temperature, C	200
Minimum/Maximum Temperature at:	
Coolant Interface, C/C	80/305
Breeder Interface, C/C	100/325
Multiplier Interface, C/C	100/325
Reflector Interface, C/C	TBD
Shield Interface (all Matls.), C/C	TBD
First Wall (PCA)	
Thickness, mm	3
Temperature Drop, C	50
Thermal Stress Factor	0.3 (190 MPa)
Pressure Stress Factor	0.3 (125 MPa)

* In the tritium recovery system the breeder stream temperature would be substantially lower.

Table 10-1
ITER First Wall/Blanket/Shield Performance Parameters List
(continued)

	<u>Inboard Form</u>	<u>Outboard Board</u>
- Tritium Breeding Ratio		
1D-Poloidal Model with 100% Coverage		1.55
1D-Toroidal Model with 100% Coverage		
Net 3D with all Geometrical Details		
(estimation)		
(3D Calculations after April 1988)		
- Maximum Nuclear Heating Rates, W/cm ³		
(@ 1.91 MW/m ²)		
Breeder		6.1
Coolant		
Neutron Multiplier		14.1
Structure		20.4
Reflector		
Shield Materials		
- Energy per Fusion Neutron, MeV		TBD
Inboard Shield		(Will be lower than with
Outboard Blanket		Li-based blanket since
Outboard Shield		Q in ³ He is ~ 0.76 MeV
Shield Materials		compared to ~ 4.8 MeV
		in ⁶ Li)
First Wall/Blanket/Shield Description		
Total Thickness (including manifolds), m		
Manifold Thickness, m		
Coolant Total Pressure Drop, MPa		
Coolant Pumping Power, MW		
Coolant Velocity, m/s		
Coolant Flow Rate, kg/h		
- Inboard Section		
Total Thickness (including manifolds), m		
Manifold Thickness, m		
Coolant Total Pressure Drop, MPa		
Coolant Pumping Power, MW		
Coolant Velocity, m/s		
Coolant Flow Rate, kg/h		
- Full Coverage First Wall/Blanket		
Total Thickness (including manifolds), m		0.85
Manifold Thickness, m		0.15
Coolant Total Pressure Drop, MPa		0.04
Coolant Pumping Power, MW		5
Coolant Velocity, m/s		30
Coolant Flow Rate, kg/s		510

Table 10-1
ITER First Wall/Blanket/Shield Performance Parameters List
(continued)

	<u>Inboard Form</u>	<u>Outboard Board</u>
- Outboard Shield		
Total Thickness (including manifolds), m		
Manifold Thickness, m		
Coolant Total Pressure Drop, MPa		
Coolant Pumping Power, MW		
Coolant Velocity, m/s		
Coolant Flow Rate, kg/s		
Tritium Removal:		
Method: Oxidizers/Mole Sieve Beds/Electrolysis		
Average Blanket Tritium Inventory (including primary loop), g		46/1000*
Maximum Blanket Tritium Inventory (including primary loop), g		Same as above
Purge Gas Parameters (if applicable)		
Gas Material		^3He
Gas Volume, m^3		15
Inlet/Outlet Temperature, C/C		100/325
Pressure, MPa		5
Mass Flow Rate, kg/s		1.05
D/T Swamping Ratio		5/1
FW/B/S Steady-State Tritium Losses (including permeation & leakage from primary loop), Ci/d		10 (w/o oxide film)
Releasable Blanket Tritium Inventory (including primary loop), g		1-46
Tritium Removal System, g		
Heat Transport System:		
Maximum Pressure, MPa		5
Inlet/Outlet Temperature, C/C		80/305
Tritium Barriers		None Assumed
Volume, m^3		380
Methods to Accommodate LOCA: Thermal Inertia, ^3He Circuit		
Maximum Material Temperature for LOCA: 800-1400°C/Adiabatic Conditions		
Condition C		
Structure		
Breeder		(Should be comparable
Multiplier		to or lower than
Reflector		SB/He blanket.)
Shield(s)		

* If tritium diffuses out of Be, lower number obtains. The tritium inventory in the ^3He stream is ~ 1 g.

Table 10-1
ITER First Wall/Blanket/Shield Performance Parameters List
(continued)

	<u>Inboard Form</u>	<u>Outboard Board</u>
Power Variation Capability:		
Minimum/Maximum Change from the Average Neutron Wall Loading, %	- 100%/+ 100% (FW Thermal Stress Limited)	
First/Wall/Blanket/Shield Compositions and Dimensions Used for Neutronics, Thermal, and Structural Analysis:		
Nuclear Responses in the TF Coils		
Maximum Fast Neutron Fluence in the Superconductor, n/cm ²	TBD	
Maximum Induced Resistivity in Copper Stabilizer, ohm · cm	(Will require 5-7 cm additional shielding compared to water-cooled shields.)	
Total Number of Anneals		
Maximum Atomic Displacement in Copper Stabilizer, dpa		
Maximum Dose to Electrical Insulator, rad		
Maximum Nuclear Heating in the Winding Pack, W/cm ³		
Nuclear Heating in the Magnet Case at Midplane, W/cm ³		
Total Nuclear Heating in TF Coils, W		
Radiation Dose in the Reactor Hall at: mrem/h		
Shutdown	TBD	
1D after Shutdown	(Should be comparable to or lower than SB/He blanket.)	
1W after Shutdown		
1M after Shutdown		
Radiation Dose at Module: mrem/h		
Shutdown	~ 3 × 10 ⁶	
1D after Shutdown	~ 10 ⁶	
1W after Shutdown	~ 3 × 10 ⁵	
Integrated Total after Heat at: GJ		
1H after Shutdown - 38		
1D after Shutdown - 600	(Should be comparable to or lower than SB/He blanket.)	
1W after Shutdown - 900		
Radioactive Waste for Each Material		
Material	Activity Ci/cm ³	Mass, kg
	1 yr 30 yr 100 yr	Volume, m ³
(Provide for each material the total radioactivity and the dominant radioactive isotopes.)		

TBD (No C-14)

Table 10-1
ITER First Wall/Blanket/Shield Performance Parameters List
(continued)

	<u>Inboard Form</u>	<u>Outboard Board</u>
Stored Chemical Energy in FW/B/S, GJ		Carbon Tiles - 10-37 Beryllium - 130
Maintenance Information		
Changeout Time for Blanket Module, D ₂		3
Surface Area of the Blanket Module m ²		0.3-0.6
Time to Switch from Non-Breeding to Breeding Operation		~ 8 hrs
Cost Information		
Cost of FW/B/S, M\$		150*
Cost of Tritium Recovery System including ³ He Purge, M\$		10-15
Cost of Heat Transport System, M\$		60
Cost of Changes in Reactor Size (if required), M\$		5-7 cm more Inboard Shield
Cost of Special Maintenance Equipment, M\$		TBD
Waste Water Detritiation, M\$		10-15
Air Detritiation, M\$		Same as SB/He Blanket
Cost of ³ He Makeup Over Life of Machine, M\$		50-60

Note: Divertor and test sections are excluded from this list.

* Includes 35M\$ for ³He and 80 M\$ for Be
(600 MT of blanket and 1400 MT of shield)
(120 M\$ blanket and 30 M\$ shield)

Figure 10-2 Tritium breeding ratio versus ^3He volume % in blanket

10.2.2 Tritium Recovery and Control

For a 600 MW (fusion) machine with a tritium breeding ratio of unity, the tritium production rate of the blanket is 91 g T/full power day, which must be recovered from the ^3He carrier gas. The tritium pressure is set by tritium leakage through the clad breeder region (i.e., Be + ^3He) into the main coolant and then across the heat exchanger. Due to the low temperatures of this blanket, a 10 Ci/d loss rate across the heat exchanger is easily met without assuming oxide barriers by, for example, a DT partial pressure of 2 Pa and a D_2 partial pressure of 25 Pa in the main coolant. The tritium loss from the breeder across the cladding and into the main coolant is similarly expected to be low because of the low temperatures and because the breeder helium is at a slightly lower pressure than the coolant helium. Assuming partial pressures of 5 Pa DT/25 Pa D_2 in the ^3He breeder gas will lead to overall tritium permeation rates of about 10 Ci/d, less if oxide barriers are present. Direct loss of tritium through leaks at valves and flanges will be very small at these tritium concentrations. Assuming 2% of the circuit volume leaks per year, and 230 m^3 coolant volume and 15 m^3 ^3He breeder gas volume, a 5 Pa DT partial pressure leads to tritium losses of 160 Ci/yr and 10 Ci/yr, respectively.

The total tritium inventory in the breeder gas is 0.06 g, and in the main coolant 0.9 g. In the event of an accidental release of this entire inventory, the maximum off-site dose would be only 50 mrem. The total tritium dissolved in beryllium is about 0.5 g. The 1% of tritium directly produced in the beryllium leads to 45 g T inventory for a 1 cm thick Be layer, but up to 1 kg T by the end of ITER operation if a BeO layer forms over much of the beryllium and inhibits diffusion.

The tritium recovery system must remove 91 g T/d from a 5 MPa helium carrier gas, where the tritium concentration is about 5 Pa DT/25 Pa D₂. The required helium gas flow rate is then 350 mol/s, or about 0.3 m³/s at 5 MPa. This flow rate is comparable to the purge flow rate in solid breeder blankets. The most proven approach for recovering tritium from such helium flows is by adsorption onto molecular sieve beds. The tritium recovery system would also need oxidizers to convert the inlet DT into the water form, and electrolyzers to separate back the oxygen once the DTO has been trapped on the mole sieves and separated from the helium. A schematic of the tritium recovery system is shown in Fig. 10-3, along with the relevant flow rates of helium and hydrogen species. The installed cost for the tritium recovery system is about 10-15 M\$ U.S.⁽¹⁾.

Figure 10-3 Tritium recovery system

10.2.3 Helium Inventory and Leakage

The ^3He breeder gas circuit must maintain enough flow to extract the tritium and maintain the DT partial pressure at under 5 Pa. This requires a flow rate of $0.35 \text{ kmol } ^3\text{He/s}$. Based on an average density of 3.3 kg/m^3 (250°C , 4.9 MPa) and flow rate of 10 m/s , the total flow area is 0.032 m^2 . This gas flows through the blanket modules where it occupies about 5% by volume, through sector plena into a 10 m diameter ring header at the top of the reactor, and then through a pipe into the tritium recovery system. From here, it is pumped back into another ring header around the machine, and again into the blanket modules. The ring headers are center fed, so each half-ring carries only half the flow. The ring headers are center fed, so each half-ring carries only half the flow. The dimensions assumed for distance and flow areas, and the corresponding volumes, are summarized in Table 10-3. The path lengths are based on draft facility layouts for NET and U.S. ITER^(2,3). The total ^3He circuit volume is about 15 m^3 , requiring $50 \text{ kg } ^3\text{He}$. This inventory would be approximately 25% lower if only outboard breeding is assumed. Moreover, the reference configuration is based on an operating pressure of 5 MPa. This pressure could be reduced substantially for the associated reduction in structural content. A reduction in structure would lower the ^3He inventory requirements. Thus, through optimization the ^3He inventory might be reduced to about 10-20 kG. In the analysis carried out here a 50 kG inventory of helium is adopted for conservatism.

The dimensions of the ^4He coolant circuit were also estimated. For a helium flow velocity of 50 m/s carrying 600 MWth, the minimum helium volume is about 230 m^3 . The pressure drop is about 130 kPa, for a pumping power of 17 MWe (80% efficient). An average velocity of 30 m/s would increase the circuit volume to 380 m^3 , but decrease the pumping power to about 5 MWe.

The leakage rate of helium from the coolant circuit and from the breeder circuit is estimated as between 1 and 5%/yr. Higher rates have been reported in some industrial helium systems where leak-tightness was not required. The lower rates were assumed in BCSS⁽⁴⁾. For the ^3He blanket, we assume that the ^3He breeder circuit has a 1%/yr loss rate due to its small size and assuming welded, leak-tight construction. This corresponds to a ^3He loss rate of about 0.5 kg/yr.

Table 10-3
³He Breeder Gas Circuit Volume

Section	Dimensions	Number	Total Volume (m ³)
Inboard blanket	150 m ² × 0.3 m × 5%	-	2.3
Outboard blanket	250 m ² × 0.6 m × 5%	-	7.5
Module plena	10 m × 0.0013 m ²	24	0.3
Inlet/exit ring headers	15.7 m × 0.016 m ²	4	1.0
Pipe to/from headers	30 m × 0.032 m ²	2	1.9
Tritium recovery system	---	1	1.0
Pumps/valves (2 × 100%)	0.4 m ID × 0.2 m × 70%	2	0.1
Miscellaneous	---	-	0.5

10.2.4 ³He Resource Implications

The ITER ³He requirements include an inventory of 50 kg and a yearly makeup rate of 8.5 kg/yr (based on 25% availability/yr and a 0.5 kg/yr leakage). Wittenberg et al⁽⁵⁾ recently estimated terrestrial resources of ³He, showing ³He reserves that could be available in the year 2000, a time-frame consistent with the ITER schedule. Based on these estimates, tritium decay in U.S. weapons is the only potential source of ³He which could satisfy ITER needs. One would also expect that the USSR has a comparable source of ³He associated with its store of thermonuclear weapons. Thus, the ultimate viability of the ³He blanket is tied to a political issue: would the U.S. and the USSR agree to make their weapons-based ³He resources available to the

project. Clearly any opinion on the likelihood of such an agreement is speculative. Nevertheless, it is worth noting that the current international climate seems well suited to a proposal of this type.

10.3 KEY TECHNICAL ISSUES

In general the technical issues associated with the helium-cooled ^3He blanket are similar to those associated with the helium-cooled, solid breeder blanket. However, it is emphasized that issues arising from the use of a solid breeder are eliminated with the ^3He breeder - such issues include temperature control of the solid breeder and tritium inventory in the solid breeder. An issue unique to the ^3He blanket is the potential for implanting the bred tritium in the beryllium multiplier. Thus, the rate which tritium diffuses out of beryllium is an important technical issue for this concept.

While the technical features of the ^3He blanket option are extremely attractive, the ability to guarantee the required ^3He resources (50 kg inventory and 8.5 kg/yr makeup) emerges as the major issue associated with the ultimate viability of this concept. Tritium decay in weapons is the only potential source of ^3He which could satisfy ITER needs. Perhaps the initiation of the ITER project combined with the eventual ratification of the INF (Intermediate Nuclear Forces) treaty could provide the catalyst for a U.S./USSR agreement to supply the ^3He requirements for the proposed blanket in ITER.

10.4 CONCEPT EVALUATION

Overall Reactor Cost - 3. There could be a small cost penalty if helium was used to cool the inboard shield.

R&D Requirement - 2. Similar to the He/SB concept except no R&D required for the breeding medium.

Design and Fabrication Complexity - 2. Comparable to the He/SB concept except only multiplier must be fabricated and no concern about temperature control of the breeder relative to tritium release.

Accidental Tritium Release - 2/3. Comparable to the He/SB concept except tritium inventory in solid breeder is eliminated.

Accidental Activated Product Release - 2/3. Comparable to the He/SB concept.

Routine Tritium Release - 3. Comparable to the He/SB concept.

Compatibility with ITER Operations - 3. Introduction of ^3He is a simple operation.

Maintenance - 2. Comparable to the He/SB concept.

Tritium Breeding - 3. Design not yet optimized for TBR.

Chemical and Thermal Reaction Potential - 3. Comparable to the He/SB concept except no lithium compound present.

Decay Heat Response - 2. Dominated by structural material.

Tritium Extraction - 2. Comparable to the He/SB concept.

Waste Disposal Rating - 2. Comparable to the He/SB concept but no C-14 production.

Power Reactor Relevance. Will provide information on helium coolants and purge streams.

ACKNOWLEDGEMENT

This work was supported by the Department of Energy, Office of Fusion Energy and by the Canadian Fusion Fuels Technology Project.

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CHAPTER 11

NUCLEAR RESPONSES IN THE PLASMA STABILIZATION ELEMENTS

11.0

NUCLEAR RESPONSE IN THE PLASMA STABILIZATION ELEMENTS

11.1 INTRODUCTION

An assessment⁽¹⁾ was carried out to determine the nuclear response in the plasma stabilization elements. The assessments were performed in a parametric fashion with regard to the element location relative to the first wall and material choices, to define and compare the nuclear responses in these elements. The main nuclear responses are neutron fluence and radiation dose in the insulating material, induced resistivity and atomic displacement in the conductor material, nuclear heating, and life analysis for the elements. These elements consist of normal conductors where both copper and aluminum materials with either MgAl_2O_4 or MgO insulating material were investigated.

11.2 GEOMETRICAL MODEL AND CALCULATIONAL PROCEDURE

In this assessment, the stabilization elements were located in the outboard breeding blanket. The position of the plasma stabilization element relative to the first wall was varied from 0 (at the back of the first wall) to 48 cm (at the outer surface of the blanket). The radial width of the element is 5 cm including coolant and structural material. Both copper and aluminum conductor materials with either MgAl_2O_3 or MgO insulating material are included in this assessment. These insulators are the preferred ceramics for the high irradiation environment. The element composition is 80% conductor material (Cu or Al), 5% water coolant, 10% type 316 steel, and 5% insulating material by volume.

11.3 RESULTS AND DISCUSSIONS

The calculations were performed for four element types corresponding to the following combinations of conductor and insulator: Cu- MgAl_2O_4 , Cu- MgO , Al- MgAl_2O_4 , and Al- MgO . The element position relative to the back surface of the first wall was varied in steps as follows: 0, 8, 18, 28, 38, and 48 cm. All the results are normalized to a 1.3 MW/m^2 neutron wall loading or $3 \text{ MW}\cdot\text{y/m}^2$ DT neutron fluence at the end of life.

The results from the calculations are given in Table 11-1. The examination of these results leads to the following main observations:

- a) There is essentially no changes in the calculated nuclear responses when MgO is substituted for MgAl_2O_4 as an insulating material for the same conductor material. However, the experimental data base on swelling for both materials shows that MgAl_2O_4 exhibits less swelling for the same conditions.
- b) The maximum insulator dose in the stabilization elements with aluminum conductor is 1 to 10% smaller than the copper case for both insulator materials.
- c) The maximum fast fluence in the aluminum conductor is 6 to 7% less than the copper conductor due to the difference in the scattering cross sections of both materials for both insulators.
- d) The nuclear heating in the aluminum element is 50 to 60% smaller than the copper case due to the difference in the gamma heating. This difference is reflecting the difference between the gamma kerma factors of aluminum and copper.
- e) The atomic displacement rate in the aluminum material is 10 to 30% higher than the copper material.
- f) The insulator dose changes from 2.9×10^{13} to 5.6×10^{11} rads across the 48 cm blanket zone. These lifetime dose levels require the use of a ceramic insulator.
- g) The fast neutron fluence ($E > 0.1$ MeV) behind the first wall is $\sim 3 \times 10^{22}$ n/cm².
- h) Irradiation experiments indicate that the swelling in spinel (MgAl_2O_4) remains below 1% at 2×10^{22} n/cm² ($E > 0.1$ MeV) for temperatures up to 925 K.
- i) The swelling and the change in the electrical resistivity are the key issues for the conductor materials. For copper conductor at the back of

the first wall, the swelling is greater than 5% and the electrical resistivity increase is ~ 130%. However, copper swelling can be suppressed by alloying but the electrical resistivity will increase. For aluminum conductor at the same location, swelling is 1 to 2.5% and electrical resistivity increase is very small.

- j) The choice between copper and aluminum depends on the element location and operating temperature. At the first wall both Al and Cu can be used. Copper will be the choice if it can be located outside the blanket.

REFERENCE

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Table 11-1. Nuclear Responses the Plasma Stabilization Elements Normalized to 1.3 MW/m² Neutron Wall Loading and 3 MW·y/m² DT Neutron Fluence at the First Wall

	Cu-MgAl ₂ O ₄	Cu-MgO	Al-MgAl ₂ O ₄	Al-MgO
A. Fast Neutron Fluence (E > 0.1 MeV), n/cm ²				
Distance From First Wall, cm				
0	2.77 + 22 ^a	2.77 + 22	2.59 + 22	2.58 + 22
8	1.86 + 22	1.86 + 22	1.73 + 22	1.72 + 22
18	7.52 + 21	7.52 + 21	7.02 + 21	7.02 + 21
28	3.23 + 21	3.23 + 21	3.00 + 21	3.00 + 21
38	1.45 + 21	1.45 + 21	1.36 + 21	1.36 + 21
48	7.02 + 21	7.02 + 21	7.10 + 21	7.10 + 21
B. Maximum Insulator Dose, rads				
Distance From First Wall, cm				
0	2.82 + 13	2.89 + 13	2.81 + 13	2.88 + 13
8	1.04 + 13	1.05 + 13	8.81 + 12	8.90 + 12
18	4.48 + 12	4.57 + 12	3.93 + 12	4.03 + 12
28	2.23 + 12	2.28 + 12	1.99 + 12	2.04 + 12
38	1.11 + 12	1.14 + 12	1.02 + 12	1.05 + 12
48	5.68 + 11	5.82 + 11	5.58 + 11	5.72 + 11
C. Maximum Nuclear Heating, W/cm ³				
Distance From First Wall, cm				
0	2.04 + 1	2.05 + 1	1.06 + 1	1.07 + 1
8	9.02 + 0	9.02 + 0	3.38 + 0	3.39 + 0
18	3.93 + 0	3.94 + 0	1.54 + 0	1.54 + 0
28	2.08 + 0	2.08 + 0	7.98 - 1	7.98 - 1
38	1.08 + 0	1.08 + 0	4.16 - 1	4.17 - 1
48	5.70 - 1	5.70 - 1	2.33 - 1	2.33 - 1
D. Maximum Atomic Displacement, dpa				
Distance From First Wall, cm				
0	3.63 + 1	3.62 + 1	4.01 + 1	4.00 + 1
8	1.62 + 1	1.62 + 1	2.14 + 1	2.14 + 1
18	6.74 + 0	6.74 + 0	8.72 + 0	8.72 + 0
28	3.01 + 0	3.02 + 0	3.86 + 0	3.86 + 0
38	1.40 + 0	1.40 + 0	1.78 + 0	1.78 + 0
48	6.74 - 1	6.74 - 1	8.72 - 1	8.72 - 1

^a 2.77 + 22 reads 2.77 × 10⁺²²

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