SUMMARY OF THE ISFNT WORKSHOP ON THE INTERNATIONAL THERMONUCLEAR EXPERIMENTAL REACTOR

MEETING SUMMARY

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Part of the International Symposium on Fusion Nuclear Technology held in Tokyo in April 1988 was devoted to a workshop on the basic tritium-producing blanket for the International Thermonuclear Experimental Reactor (ITER). The workshop participants addressed the key issues of the ITER blanket in several technical areas: neutronics, tritium, safety, materials, and mechanical/maintenance. In addition, special sessions were held on selection criteria and nuclear testing in ITER.

I. INTRODUCTION

An International Symposium on Fusion Nuclear Technology was held in Tokyo, Japan, April 10-19, 1988. With the exception of a smaller workshop held at the University of California-Los Angeles (UCLA) in 1983, this was the first international conference devoted specifically to fusion nuclear technology. The symposium was attended by ~300 scientists and engineers from 13 countries, and consisted of a conference (April 10-15, 1988) and a workshop (April 18-19, 1988). The conference had three major plenary sessions: (a) overviews of major fusion programs [European Communities (EC), Japan, United States, and USSR]; (b) reports on technical progress in the design of the next fusion engineering facility [Next European Torus (NET) in Europe, Fusion Experimental Reactor (FER) in Japan, OTR in the USSR, and Tokamak Ignition/Burn Experimental Reactor (TIBER) in the United States], as well as the new recent activity on the International Thermonuclear Experimental Reactor (ITER); (c) overviews of fusion nuclear technology programs in EC, Japan, United States, USSR, and Canada. About 260 contributed technical papers presented research progress on blanket technology, tritium processing systems, first-wall and high-heat flux components, modeling and experiments, neutronics, and system analysis related to fusion nuclear technology, including safety.

This report contains a summary of a 2-day workshop held at the University of Tokyo following the conference. The workshop was attended by ~70 international participants and was devoted to the tritium-producing blanket for ITER. Discussion and information exchange among the international technical experts focused on a number of key technical topics for the ITER blanket:

1. requirements on the performance of the blanket
2. technical features of candidate blanket concepts
3. key issues
4. major research and development (R&D) items required for selection (the workshop did not address the general R&D requirements)
5. important selection considerations.

These topics were addressed in plenary sessions and also in a number of parallel sessions devoted to specific technical disciplines. Although the workshop focused mainly on the ITER basic blanket, a session was devoted to nuclear testing in ITER.

The participants felt that the workshop provided a unique opportunity for discussion and information exchange among fusion nuclear researchers in the world program. They also felt that the workshop was timely since it provided important input to the new and important activity for the design of an international thermonuclear experimental reactor. Summaries of all the workshop sessions are presented in the rest of this summary.

II. SUMMARY OF BLANKET CONCEPTS

The driver blanket designs that have been developed and presented over the past few years are summarized in Table I. The main technical aspects of these concepts are summarized here.

II.A. Aqueous Lithium Salt

In this concept, a lithium salt is dissolved in the water coolant of the shield at the phase of ITER operation when tritium breeding is desired. There are two candidate salts: $LiNO_3$ and $LiOH$. $LiNO_3$ has been assumed in most studies to date because of its lower corrosion and higher solubility, but it also is more susceptible to radiolysis and produces ~4000 Ci of ^{14}C over a 3 $MW\cdot yr/m^2$ ITER fluence, compared to 10 to 50 Ci for LiOH (and other water-cooled and oxide ceramic blankets). The water coolant would operate at under 100°C and 0.3 to 1 MPa—the higher pressure

TABLE I

Types of Designs for the Basic Tritium-Producing Blanket

	Water-Cooled	Helium-Cooled	Self-Cooled
Aqueous salt	*	---	*
Solid breeder	*	*	---
LiPb	*	*	*
^3He	*	*	---
Helium/particulate	---	---	*

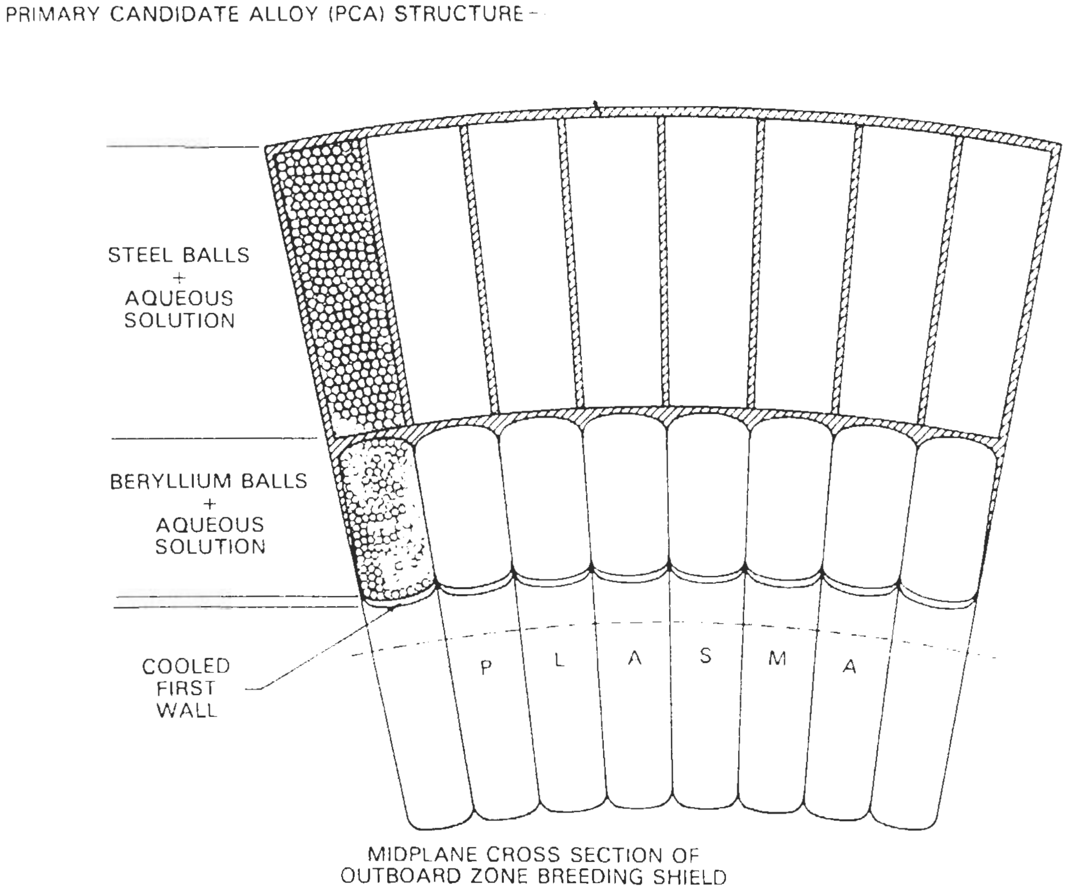


Fig. 2. The TIBER (University of Wisconsin) driver blanket concept. Steel and beryllium balls are used to simplify the overall fabrication. The lithium salt solution flows between the balls.

coolant is brought up to high temperature and pressure, and the LiPb is melted and circulated for tritium recovery. Figure 6 illustrates this concept.

II.D. Self-Cooled Liquid Metal

Liquid metals such as lithium or LiPb can also be used without a separate coolant. LiPb is better with respect to chemical reactivity. The magnetohydrodynamic (MHD) pressure drop is minimized by directing the coolant such that it flows slowly poloidally in large channels, and quickly (for good cooling) along the first wall in a toroidal direction. Electrical insulators have also been proposed. For ITER, such a liquid-metal-cooled blanket might be used for breeding only on the outboard side where the magnetic field is lower. To minimize possible chemical reactivity, it is desirable to avoid the use of water in other systems such as the divertor. An inert cover gas in the reactor hall would also be useful. Figure 7 illustrates the concept.

II.E. ^{3}He Blanket

Helium-3 has a higher (n,T) microscopic cross section than ^{6}Li by a factor of ~60. In this concept, ^{3}He is circulated through the blanket and multiplier in a manner analogous to the purge stream in a solid breeder blanket except that in this case there is no lithium ceramic. Adequate tritium breeding is achieved with use of a neutron multiplier, even with small amounts of ^{3}He in the blanket, typically 3 to 15 vol% at 0.1 to 5 MPa. If the circuit volume is minimized, the overall ^{3}He inventory for ITER has been estimated as 20 to 50 kg, with 0.5 kg/yr leakage and burnup of up to 10 kg/yr for 700-MW fusion power at 30% burn time. This much ^{3}He appears to be available in the post-2000 time frame from the decay of military tritium stockpiles. Both helium- and water-cooled versions have been studied. Figure 8 illustrates the water-cooled version of the concept.

II.F. Helium/Particulate Blanket

In this concept, small solid breeder particles are swept through the blanket entrained by the helium coolant. Tritium bred in the ceramic escapes into the coolant since the mean-free-path of the mega-electron-volt-tritons is larger than a typical particle size. As with other blankets with a beryllium multiplier, only would allow H_2 addition to suppress radiolysis, or keep radiolytic gases dissolved if they are produced. Radiolysis products would be recombined outside the blanket.

Tritium recovery from the water (at ~10 Ci/kg on average) is by conventional technology. The equipment would be physically similar to or smaller than that at the Ontario Hydro Darlington Tritium Removal Facility. The tritium inventory is reduced by minimizing the water volume to ~300 m^3 as in Canada Deuterium Uranium (CANDU) reactors. CANDU reactor leak-tightness and air driers are necessary. Separating the salt circuit from the water coolant is an option that would substantially reduce tritium losses and inventory.

Figures 1 and 2 illustrate the blanket concept.

II.B. Solid Ceramic Breeder

In this concept, a lithium-bearing ceramic is cooled by water or helium. Li_2O, $LiAlO_2$, and Li_4SiO_4 are reasonably well-characterized ceramics that could be used. Tritium is recovered by purging the ceramic continuously with helium containing 100- to 1000-ppm H_2. Tritium is recovered from this helium stream, as usually proposed, by oxidation and trapping on molecular sieves. The coolant conditions are 0.3 to 1 MPa and 60 to 90°C for water, and 1 to 5 MPa and 80 to 300°C for helium. The helium purge is 0.1 to 1 MPa, leaving the breeder at around 300°C.

The tritium release from the breeder is highly sensitive to the temperature. Generally, most ceramics must be kept between 400 and 800°C for adequate tritium release. This requires either a high-temperature coolant or the use of a thermal resistance region between the coolant and breeder. This thermal resistance may be provided by using a helium-filled gas gap of a few millimetres, with a layer of ceramic such as Al_2O_3, or (as in recent designs) by using the beryllium multiplier. This last approach also provides good mixing of the multiplier and solid breeder, which improves tritium breeding.

The breeder may be in sintered pellet or sphere-pac (pebble) form. Recent designs have tended to prefer pebbles because of the ease of filling of the blanket, no cracking in operation, and predictable temperatures. The breeder is usually designed to operate within only part of its allowable temperature window to allow operation at power levels other than the design value.

Figures 3, 4, and 5 illustrate the blanket concept.

II.C. Lithium Lead: Water or Helium Cooled

In this concept, the 17Li-83Pb eutectic is used for breeding and is cooled separately by water or helium. The LiPb is circulated slowly to the external tritium recovery system. Since 17Li-83Pb melts at 235°C, there are two possible operating modes. In the first, the coolant is also hot—water at 300°C and 10 to 15 MPa, for example. The LiPb is kept liquid and circulates slowly with a pressure drop of ~1 MPa in the reactor magnetic field. In the second operating mode, the coolant is cold during ITER operation, e.g., water at <100°C and <1 MPa. The LiPb is solid and the tritium accumulates. During ITER downtime, the

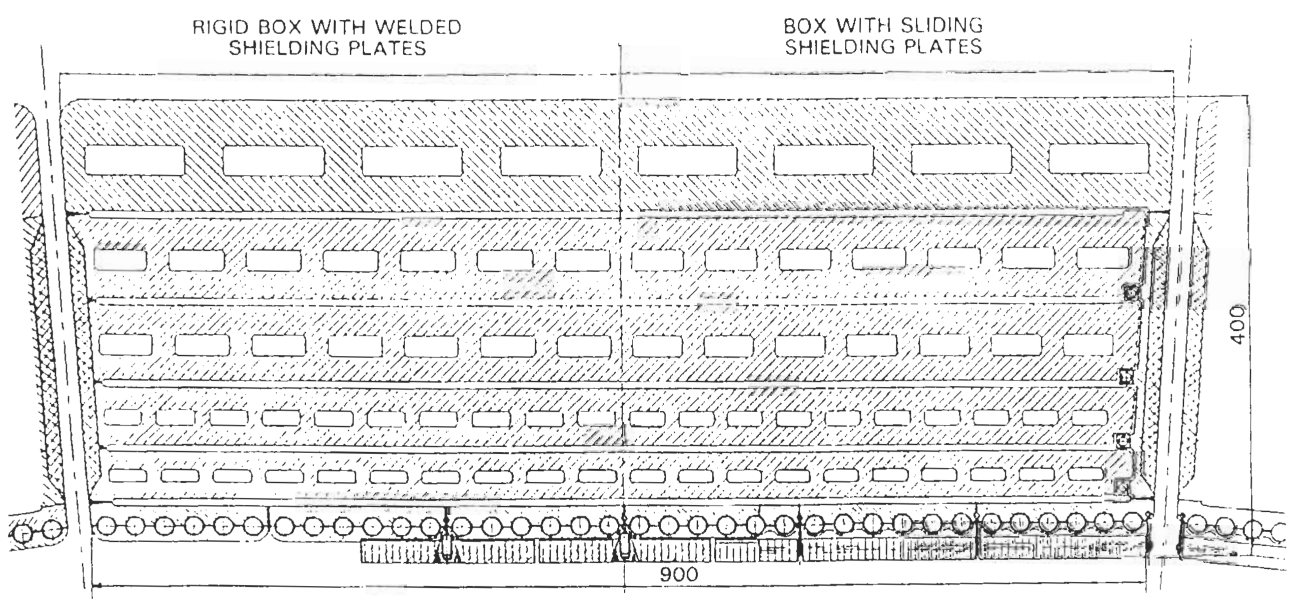
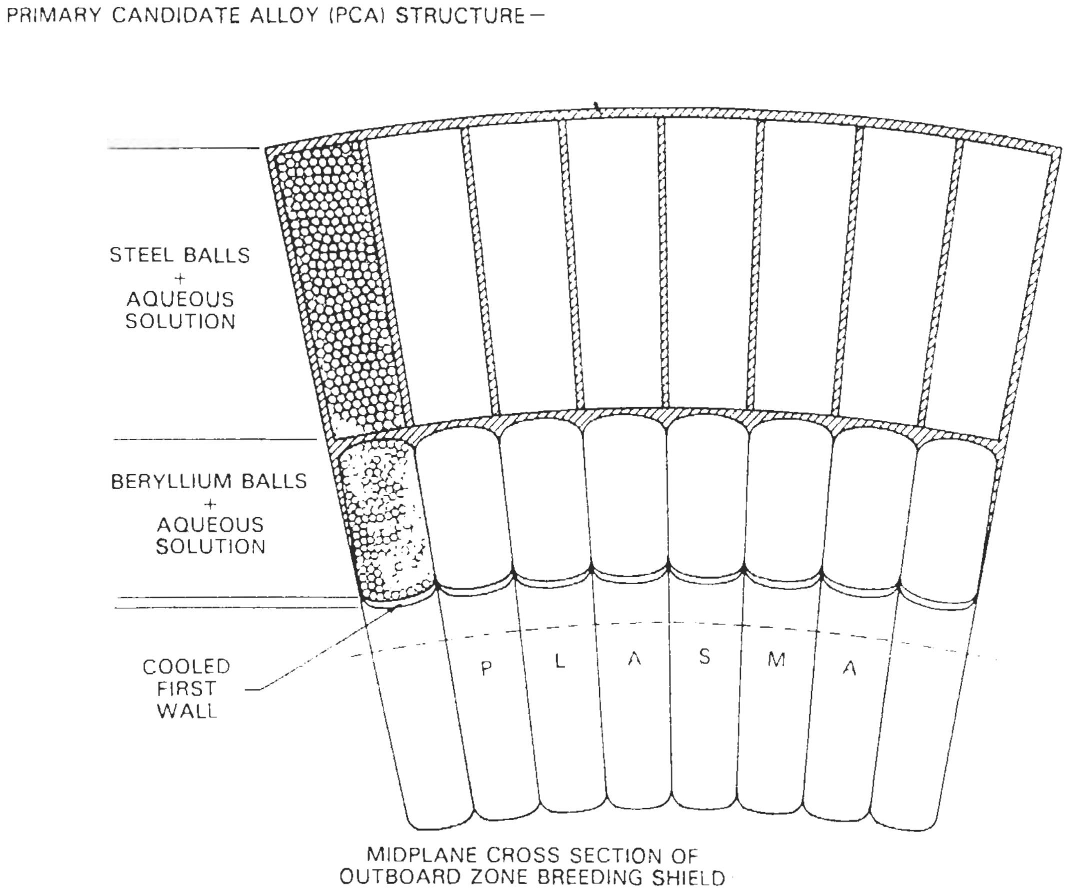


Fig. 1. The NET steel/water shielding blanket concept. The water (25 vol%) could have a lithium salt added to the water coolant during the technology testing phase. Dimensions are in millimetres.



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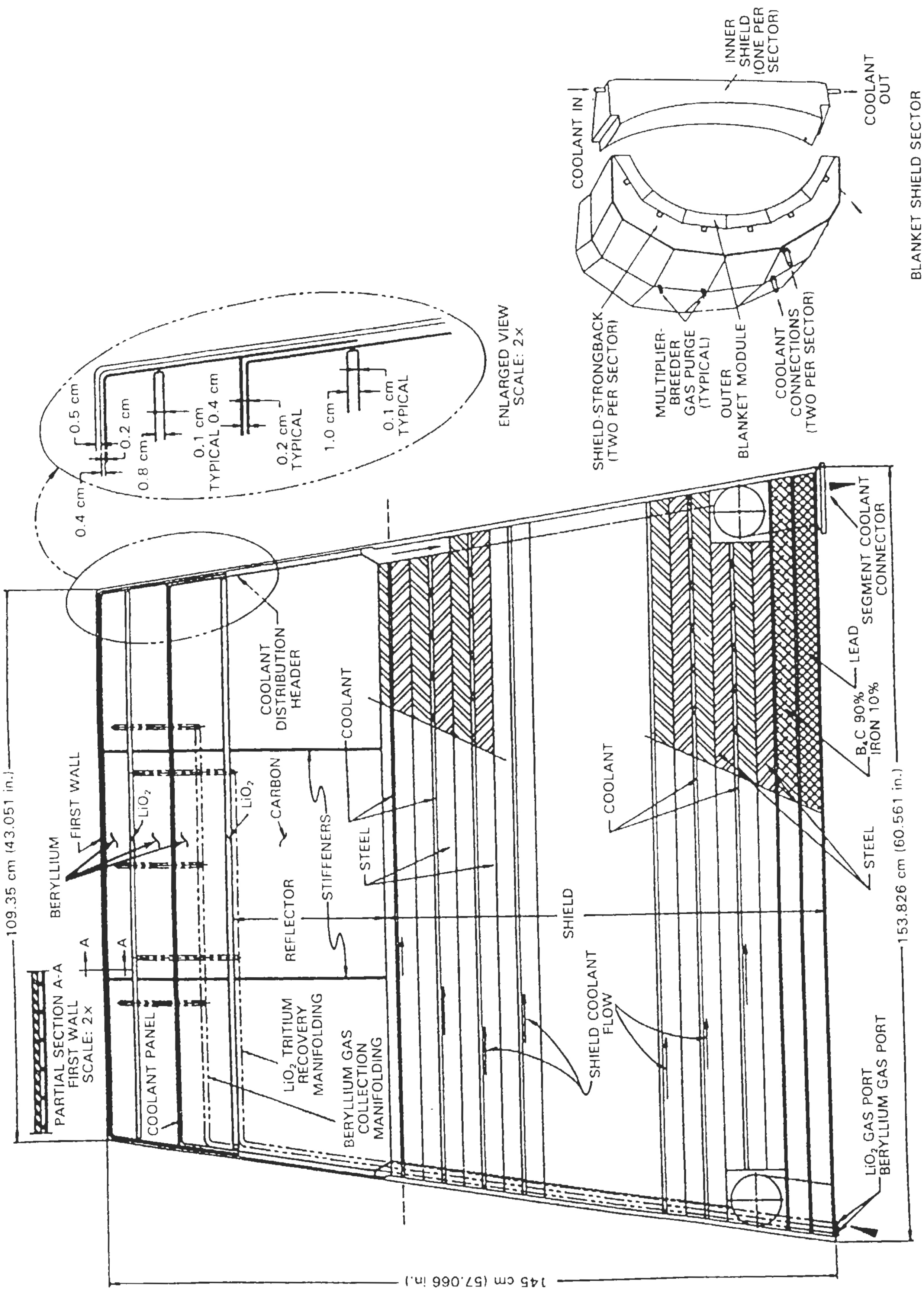

Fig. 3. The Argonne National Laboratory (ANL) water-cooled ceramic blanket concept. The Li_2O breeder is contained within two 1-cm-thick regions and separated from the coolant by a thick beryllium plate. Nonstructural cladding prevents Li_2O-beryllium reactions.

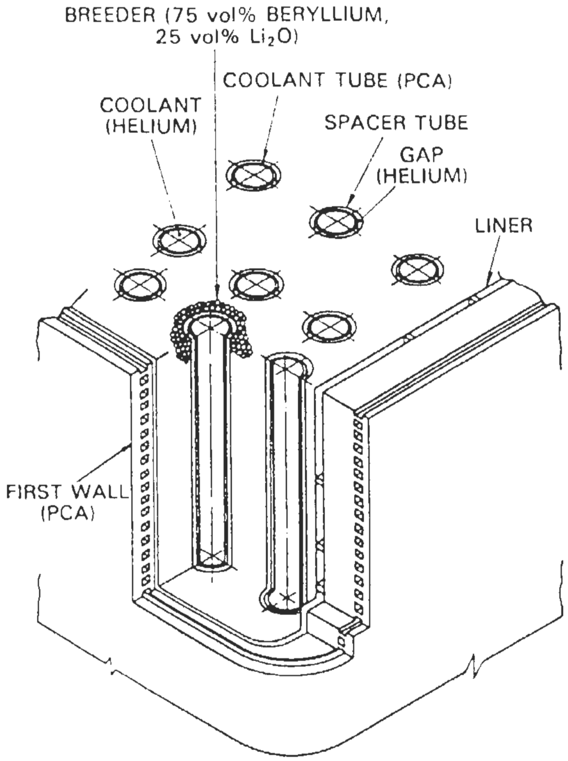

Fig. 4a. The Japan Atomic Energy Research Institute (JAERI) helium-cooled ceramic breeder concept. The coolant flows through tubes within the breeder. Thermal insulation is provided by a gas gap.

a small percentage of particles is required—about 1.5 vol%—for good breeding. The addition of these particles increases the effective heat capacity of the coolant, so the flow rate, and thus the pressure drop, is reduced relative to pure helium. There is some erosion data base from experiments conducted in the United Kingdom in the 1960s as part of their gas-cooled reactor program. Reasonable conditions are 3-MPa helium at 50 to 300°C and 8 to 10 m/s, with 1- to 10-μm particles on average. The piping geometry must be simple to avoid settling or erosion, and the particles can be separated from the helium coolant outside the blanket to avoid damaging the heat exchanger or pumps. Figure 9 illustrates the concept.

III. NEUTRONICS

The issues discussed were divided into four main categories: tritium breeding capability, shielding performance, activation, and R&D needs.

The blanket concepts proposed for the ITER basic tritium breeding blanket were reviewed. It was concluded that overall tritium self-sufficiency can be achieved without tritium breeding in the inboard region provided that a sufficient amount of neutron multiplier is used in any of the proposed blankets. It was pointed out that for the 17Li-83Pb blanket concept to provide adequate tritium breeding without an inboard blanket, a separate beryllium multiplier is

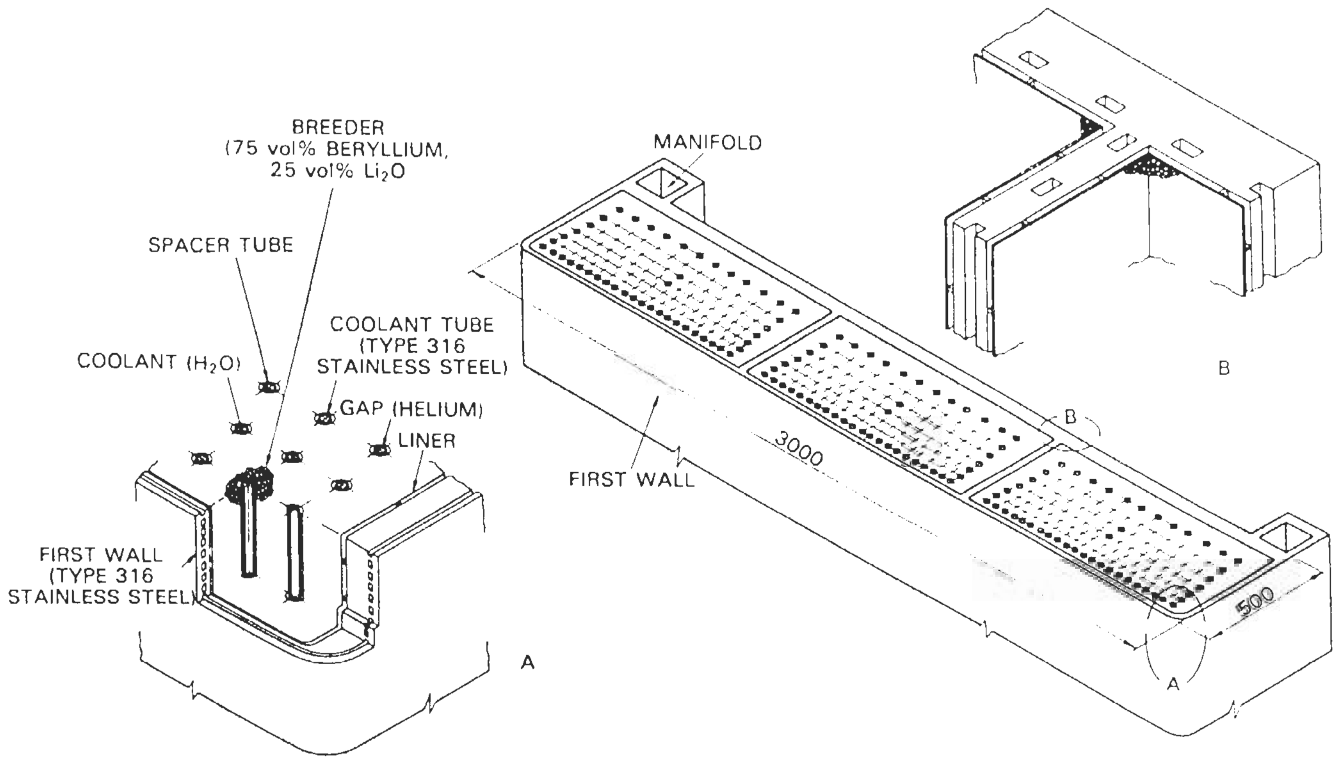

Fig. 4b. The JAERI water-cooled ceramic breeder concept.

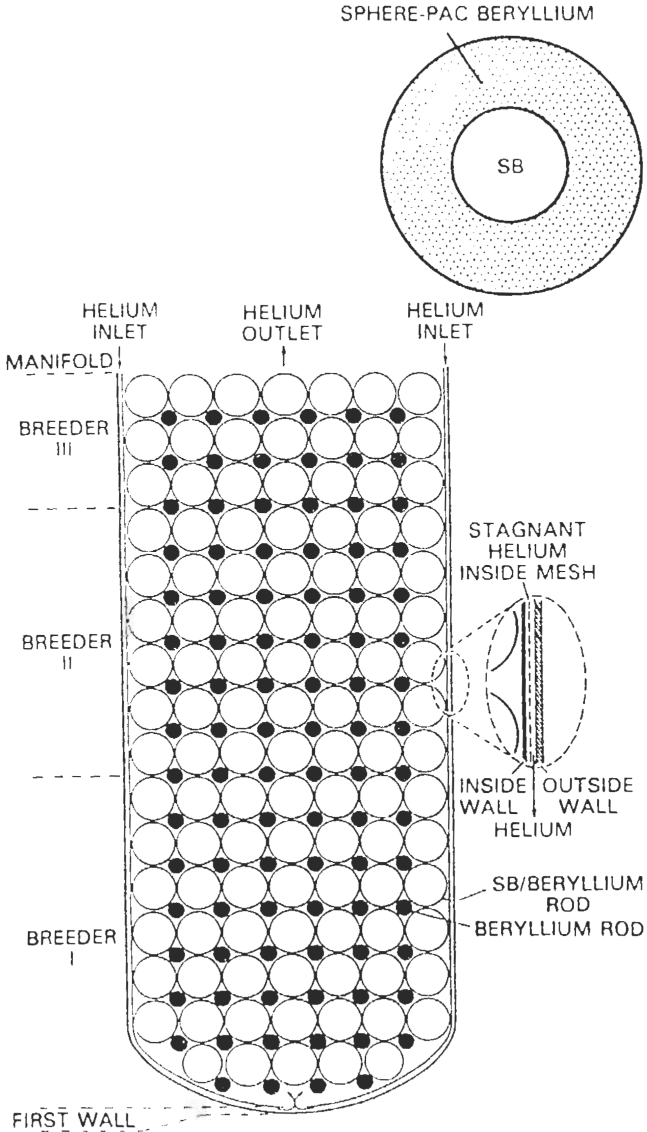

Fig. 5. The UCLA helium-cooled ceramic blanket, with the breeder and multiplier inside tubes and the coolant flowing around them. Thermal gradient is provided by the use of beryllium pebbles.

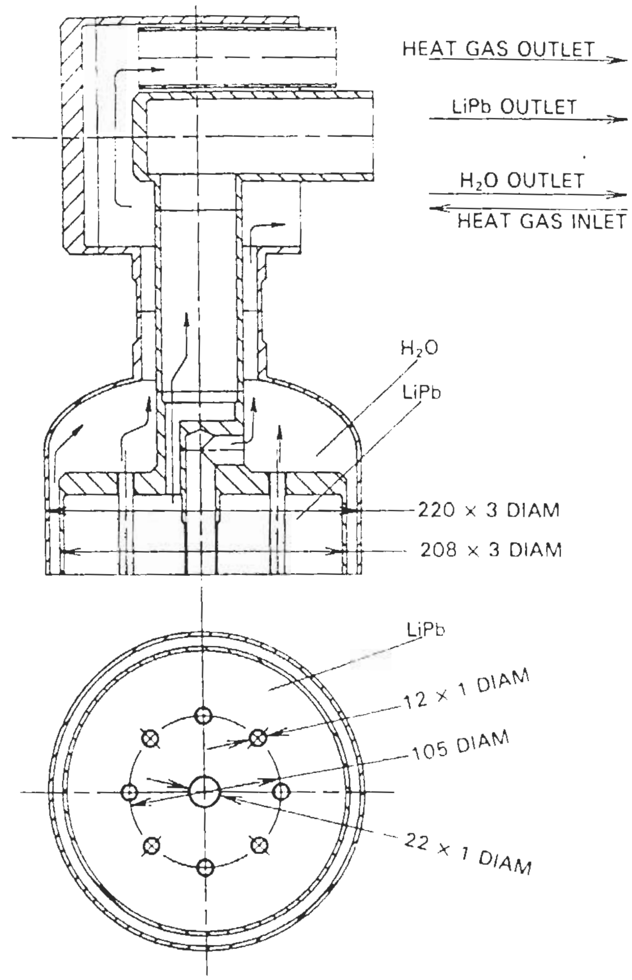

Fig. 6. Representative water-cooled LiPb concepts developed in the USSR. Dimensions are given in millimetres.

needed. The aqueous blanket concept has a higher tritium breeding potential as it allows for inboard tritium breeding with no additional complexity or machine cost penalties and with enhanced magnet shielding performance. This can lead to requiring a lower amount of multiplier in the aqueous blanket to achieve overall tritium self-sufficiency. Beryllium is considered to be a more effective neutron multiplier than lead. While using beryllium as a multiplier ensures overall tritium self-sufficiency, using lead as a multiplier makes overall tritium self-sufficiency questionable for most of the proposed blanket concepts. The breeding capability is enhanced when a mixture of beryllium and breeder material is used when compared with a separate layered configuration. The degradation in tritium breeding due to lithium burnup, particularly in solid breeders for the ITER fluence of 3 $MW \cdot yr/m^2$, is not a major concern. Penalties in tritium breeding are expected from thick first-wall, passive control coils and protective tiles on the outboard first wall. The reduction in tritium breeding ratio (TBR) depends on the thickness and material used. If protective tiles have to be used in the outboard region, beryllium is the preferred tile material from the neutronics point of view, because it enhances rather than degrades the TBR. The material choice for the inboard first-wall protective tile is expected to have an insignificant impact on TBR.

The magnet radiation limits proposed by the different parties were reviewed and found to vary by up to a factor of 5. Since the limits have a strong impact on the magnet shield design, it is essential to agree on them very early in the ITER design. A specialists' meeting among magnet, material, and neutronics experts is highly recommended to resolve this issue.

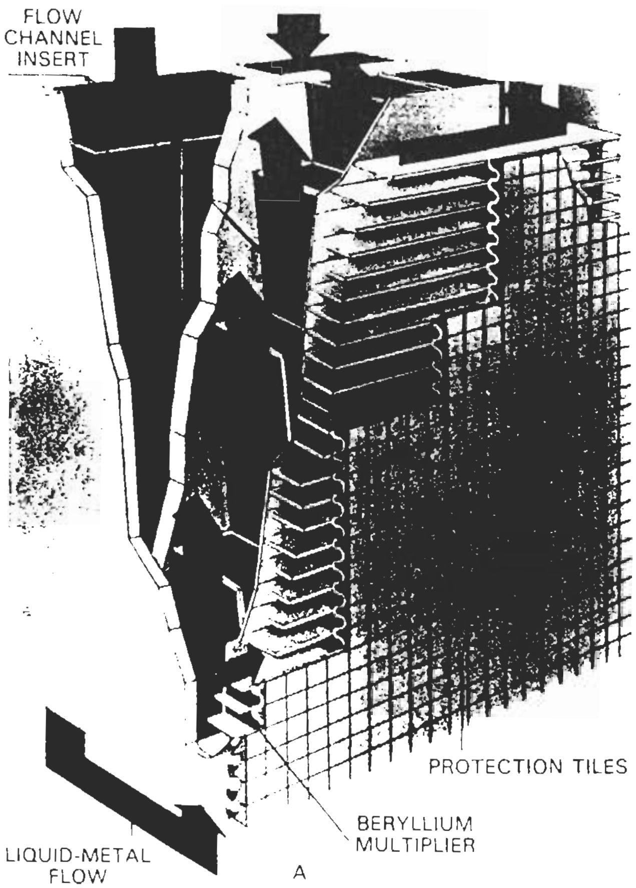

Fig. 7. The Kernforschungszentrum Karlsruhe self-cooled liquid-metal blanket, illustrating the poloidal/toroidal flow path used to minimize pressure drop while providing good first-wall cooling.

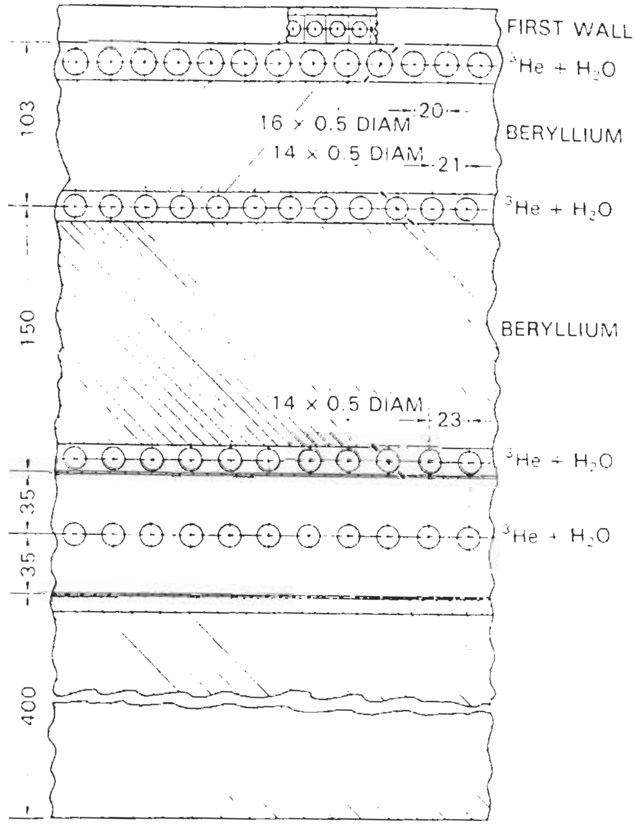

Fig. 8. Water/beryllium version of the ^{3}He blanket concept developed by the Kurchatov Institute. The ^{3}He flows through tubes in the water channels at 0.1 MPa. The water coolant is at <100°C and <0.7 MPa. Dimensions are in millimetres.

Design margins should be allowed to account for streaming in shield gaps, nuclear data uncertainties, and uncertainties in geometric and calculational modeling. Careful design of the blanket and shield is required to minimize streaming through assembly gaps and coolant tubes. The shield design should also have adequate margin to accommodate possible enhanced plasma performance. Due to the poor shielding properties of helium gas, the helium-cooled blankets and shields have lower shielding efficiency compared to the water-cooled concepts. However, helium has safety advantages over water. The choice of inboard shield material impacts the required shield thickness and machine size and cost. A trade-off study is required to determine the best inboard shield design.

The activation issues were discussed in a joint meeting with the safety group. It was recommended that the design should minimize elements that produce large decay heat (e.g., tungsten and tantalum) in regions with high neutron flux. This is essential to yield a design that depends on inherent and passive methods to remove the decay heat in case of a loss-of-flow accident or a loss-of-coolant accident (LOCA). It was pointed out that using LiOH instead of $LiNO_3$ in the aqueous blanket is preferred from the neutronics point of view to avoid the production of large amounts of ^{14}C.

The waste management philosophy for ITER was discussed. Some participants emphasized satisfying the conditions for shallow land burial. However, most of the participants indicated that while this requirement is important for commercial fusion reactors, it is not essential for an experimental reactor like ITER. This issue needs to be addressed by the ITER team. Information needed to define what is meant by low-activation structural materials is not sufficient at the present time. However, an effort should be made to reduce the amount of long-lived radionuclides produced without compromising the safe operation of the machine. A primary effort should aim at reducing the impurities that produce long-term activity, such as silver and niobium.

The immediate neutronics R&D needs for ITER were discussed. The effort of the International Atomic

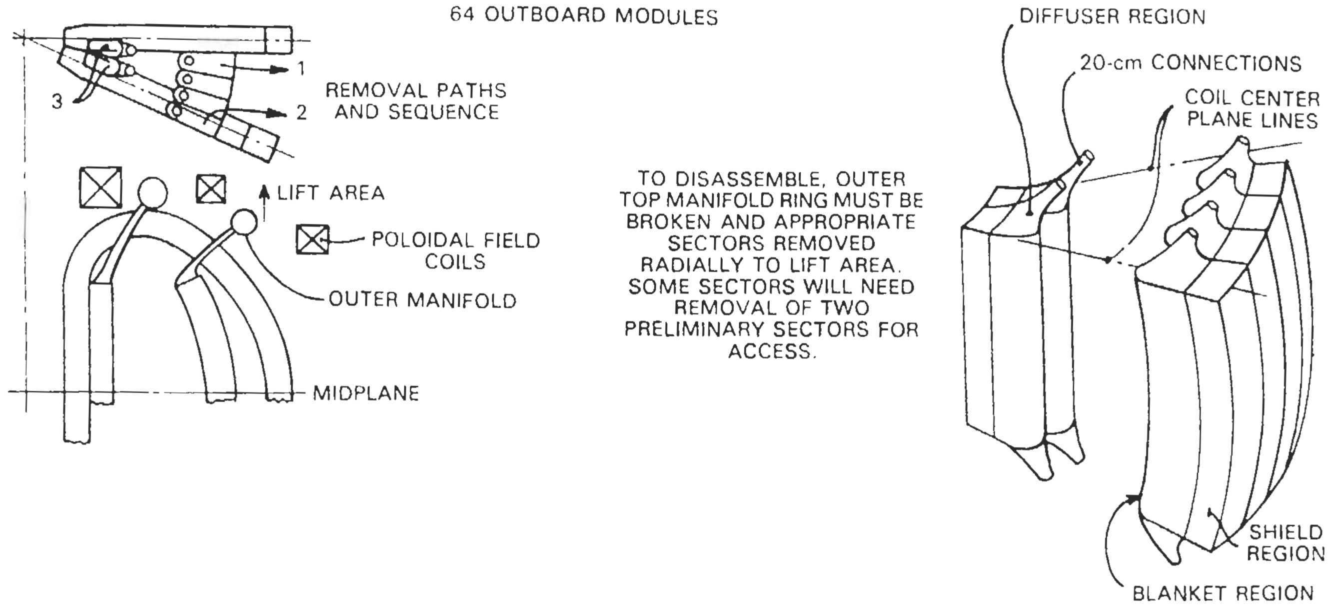

Fig. 9. General Atomics/ANL helium/particulate blanket concept. The coolant containing particulates enters at the top and flows down and out the bottom through channels in the beryllium multiplier.

Energy Agency to develop a joint international data library that can be used in ITER was brought to the attention of the participants. This effort was endorsed by the participants in this meeting. In the meantime, benchmark calculations are recommended to compare the capabilities of the parties involved in ITER and to determine the expected uncertainties in the calculated major nuclear parameters. It was pointed out that most of the uncertainty in the calculated TBR is related to uncertainties in the beryllium data. Beryllium multiplication integral experiments as well as experiments that determine the energy spectrum and angular distribution of neutrons produced by the Be(n,2n) reaction are required. The present kerma factor data are not adequate for accurate determination of nuclear heating required by the design. An effort to verify experimentally nuclear heating data and calculations is essential. The need to improve the uncertainty in iron and tungsten data was emphasized. Since these materials are likely to be used as the main inboard shield material, a significant increase in machine cost is required to allow for the uncertainties in their cross-section data.

IV. TRITIUM

The tritium aspects were considered under three headings:

1. tritium retention in the blanket (inventory)
2. tritium permeation and leakage from the blanket
3. tritium technology appropriate for tritium recovery from the blanket.

In order to focus the discussion, the list of concepts presented in Sec. II was used as reference. It was not felt appropriate to rank the concepts based on tritium issues, since

1. tritium aspects represent only a small point of the potential ranking criteria
2. the perceived severity of the issues is strongly design dependent, and the workshop was asked to assess *classes* of concepts rather than specific designs.

Therefore, attention was focused on identifying and quantifying the tritium-related issues as they related to new data needs.

Safety and economic aspects were not considered explicitly. However, safety concerns are reflected in a number of specific data/design issues. It was pointed out that economic aspects should reflect the cost to provide containment, monitoring, and other service facilities. These facility costs may be as large as the actual blanket recovery system costs.

The following issues were identified in the workshop as those requiring further study to provide data for design. Designs that take the results of these investigations into account are required before the relative merits of the various blanket options can be reliably assessed for ITER. In most cases, basic experiments or work with small "pilot" assemblies are required. In others, literature review and reinterpretation of existing data may be sufficient. Below, the issues are listed by blanket concept.

IV.A. Aqueous Salt Blanket

1. impurity control in the circulating solution
2. radionuclide (e.g, ^{14}C in CO_2 for nitrate-salt) carryover to tritium extraction process
3. hydrogen overpressure required for radiolysis control
4. cost scaling for tritium extraction at rates corresponding to a TBR of 1 for ITER
5. electromagnetic effects on ion separation and corrosion.

IV.B. Lithium-Ceramics Blankets

1. correlation of hydrogen addition and tritium inventory in breeder material
2. exchange of tritiated water with water in molecular sieve used in tritium extraction dryer beds
3. effect of "preswamping" of molecular sieve on capture efficiency for "pilot-scale" dryer beds
4. lifetime of proposed permeation barriers under blanket conditions of radiation, thermal cycling, and reducing atmosphere
5. feasibility of temperature control in ceramic blankets that use low-temperature helium or water
6. irradiation effects on tritium retention in breeder
7. tritium extraction demonstration with larger scale irradiation experiment representative of ITER-like breeder module
8. breeder compatibility with beryllium multiplier
9. tritium retention in beryllium (common for all beryllium multiplier blanket designs)
10. effect of batch regeneration on tritium retention in solid breeder.

IV.C. LiPb Eutectic Blanket

1. effect of hydrogen addition to helium sweep-gas on tritium recovery efficiency
2. demonstration of an ITER-relevant tritium extraction process
3. evaluation of variations in solubility and diffusivity data for tritium in the breeder material
4. behavior of tritium under "remelt" conditions following irradiation in the solid state.

IV.D. ^{3}He Blanket

1. effect of tritium implantation into beryllium-oxide layer during formation on inventory and recovery
2. confirmation of leakage and blowdown-frequency assumptions for the breeder circuit.

IV.E. Lithium-Ceramic Particulate Blanket

1. particulate agglomeration and size distribution under blanket conditions
2. impact on mechanical components such as valves, pumps, and molecular-sieve dryers (pore blocking)
3. availability of technology for separation of fine particles from the flow stream
4. tritium distribution between particles, coolant/sweep-gas, and structural/breeding material.

V. SAFETY

The objective of the Safety Working Group was to examine the ITER blanket concepts to determine the relative safety and environmental advantages or disadvantages and also to identify any key unresolved issues and specific data requirements. Much discussion was devoted to potential accidents because the group felt that this subject is critical and has not been discussed among the various design groups. The following topics were considered.

V.A. Plasma Disruptions

For the very high plasma currents considered for ITER, major plasma disruptions could cause severe forces due to induced currents. To protect against these forces, the first wall may contain much more structural material than considered in the present designs. This could affect the breeding ratio. Also, in conducting coolants such as LiPb and perhaps aqueous salt, transient MHD-induced pressures may be high. To evaluate forces for complex blanket geometries, a three-dimensional eddy current model must be incorporated into a plasma disruption code.

V.B. Station Blackout/Loss of Coolant/ Loss of Coolant Flow

The design should employ inherent/passive features to ensure the machine can survive a loss of power leading to the loss of coolant flow, as well as a LOCA. Water may have an advantage over helium as a coolant since for water, natural convection could cool the blanket during a loss of pumping power. Also, dual independent flow circuits such as proposed for NET may be desirable in reducing the probability of a total loss of coolant. The separate aqueous salt and water coolant design may have an advantage in that either circuit could remove the afterheat.

If materials are selected that minimize afterheat in the blanket and inboard shield, then afterheat could potentially be removed by conduction. If this is possible, then helium cooling would not pose a concern for afterheat removal and water systems could be simplified. More analysis is needed on each concept and the inboard shield to determine if afterheat can be removed by conduction during a LOCA. Also, analysis on the effectiveness and costs associated with approaches for afterheat removal discussed above should be performed early in the design process.

V.C. Breaks Releasing Coolant into the Torus

A coolant release into the torus could result in overpressurization of the vacuum vessel and, in the case of water, production of hydrogen by reaction with hot graphite. Overpressurization could also damage the blanket modules. Helium coolants may have sufficient pressure that some protection would be required. Water coolants are at a disadvantage since hydrogen generation from graphite reactions could result in an explosion if the accident was accompanied by air inlet or caused air inlet. Designers should look at bonded graphite tiles to see if temperatures can be maintained at sufficiently low levels to remove this concern. Potentially serious consequences of reactivity of released water with hot liquid metals, e.g., in test modules, were also pointed out.

V.D. Breaks Releasing Coolant Inside the Blanket Module

Because of the complexity of the blanket designs, a coolant break inside the module may have a high probability of occurrence. Because of this high probability, the consequences of this event must be kept small, such as a replacement of the module. The NET water-cooled LiPb design appears to be acceptable since the cooling tubes are enclosed in a canister that can withstand the pressure of a break. It may be possible to design a water-cooled ceramic breeder to operate at a sufficiently low pressure such that the module wall could contain the pressure from a coolant tube break. If the aqueous salt concept requires 1 MPa to reduce radiolysis, then design of the module wall to contain this pressure may be difficult.

V.E. Breaks Releasing Coolant Outside the Vacuum Vessel

Breaks releasing coolants outside the plasma vacuum boundary could occur due to a failure or to an external initiator such as a serious event that could also result in loss of building confinement. Release of tritium and activation products was considered.

As a general guideline, it was considered that any accidental tritium release should involve a dose at the site boundary lower than the public evacuation limit. Typical inventories associated with this type of tritium release are 150 to 200 g. In the case of ceramic breeders, where inventories of up to several hundred grams are expected, it seems that this guideline would not be too difficult to satisfy. There is some concern in the case of lithium aqueous solutions where larger inventories are expected. In aqueous salt designs, solutions have to be envisaged to limit water boiling in case of accidents where afterheat is a concern.

Another important concern is the possible mobilization of structural material elements, in particular those of components near the plasma. In this respect, tungsten was judged to pose problems owing to its high afterheat value at shutdown. Concerning the breeder and neutron multiplier, the afterheat and activation levels at shutdown appear low compared to values for structural materials. However, some concern was expressed concerning the possible mobilization of some of the lead elements (in particular, ^{210}Po), and of beryllium, owing to its larger volatilization rate as compared to steel. For beryllium, the problem of toxicity is more important to address than that of radioactivity. The question of impurities in the breeder materials could also be a concern.

V.F. Waste Disposal Problems

Designs should minimize the volume and activity level of wastes. For the United States, meeting low-activation wastes in ITER is very important to promote fusion in general. For the other participants the problem of low-activation waste was not considered as an absolute requirement for the current design studies. A study should be done to define common criteria and determine what has to be done in terms of material selection and procedures to meet these requirements. In general, it was noted that the main problems associated with waste disposal are from structural materials and their impurities. One exception is represented by ^{14}C produced in the nitrate salts.

VI. MATERIALS

The materials group considered eight classes of materials that present important issues for the various ITER blanket concepts that have been proposed:

1. austenitic steel structure
2. "low-activation" austenitic alloys
3. beryllium multiplier
4. lead multiplier
5. 83Pb-17Li breeder
6. ceramic breeders
7. aqueous salt breeders
8. ^3He breeder.

The discussions were dominated by the structural materials and beryllium issues. The properties of ceramic breeder materials, e.g., Li_2O, Li_4SiO_4, $LiAlO_2$, and Li_2ZrO_3, were discussed; however, aspects associated with tritium recovery were also discussed in the group on tritium, as indicated in Sec. IV. Although several issues were raised for most of the materials classes considered, the following sections highlight only the most important conclusions for each type of material.

VI.A. Austenitic Steel Structure

Austenitic steels are proposed as the structure in virtually every ITER blanket; therefore, issues associated with this material are particularly important. The most important issues for the austenitic steel structure include (a) aqueous stress corrosion ("pure" water and aqueous salt), (b) low-temperature fracture toughness of irradiated material, (c) preferred thermomechanical treatment (TMT), and (d) joining of carbon tiles to structure.

Aqueous stress corrosion cracking (SCC) is considered a high-priority feasibility issue for both the pure water-cooled and aqueous salt concepts. The issue for the water coolant is particularly important since water-cooled austenitic steel systems are generally proposed for auxiliary systems, e.g., heating, current drive, etc. In addition to the serious problems encountered in the fission reactor industry, the fusion environment produces high hydrogen transmutation rates, higher radiation fluences, and electrolysis and radiolysis effects, all of which are expected to exacerbate the SCC. The SCC should be less severe at low temperatures (<100°C); however, weldments and crevices are expected to be particularly sensitive to SCC. The SCC of austenitic steels in candidate aqueous salts, e.g., $LiNO_3$ and LiOH, is expected to be even more severe than in pure water. Although no data have been reported in the literature for the salt concentrations suggested, scoping experiments have been initiated.

Although radiation-induced swelling of austenitic steels is not regarded as a serious problem at temperatures below ~350°C, loss of ductility or fracture toughness at temperatures <300°C is a major concern. Significant effects are expected at moderate fluences (<10 dpa) and weldments are of particular concern.

The preference of cold-worked (CW) versus solution-annealed TMT is controversial. The CW material provides a significant strength advantage; however, high-temperature joining processes tend to anneal the structure. Joining of protective carbon tiles to a steel structure was raised as a difficult problem.

Compatibility of austenitic steel with beryllium was not considered serious at temperatures below ~600°C.

Candidate low-activation austenitic steels include manganese-stabilized steels and possibly Type 304 steel. These alloys are expected to have properties similar to the conventional steels. Some participants questioned whether an adequate data base for these alloys can be provided in a time frame consistent with ITER.

VI.B. Beryllium Issues

Beryllium is used extensively in nearly all high breeding blankets, and, therefore, its performance limitations are high priority. Most of the issues identified are constraints and not feasibility issues since alternate design solutions generally exist. Dominant issues relate to irradiation performance and chemical performance.

The temperature and fluence limits for extensive swelling are highly uncertain but are expected to be important constraints. The mechanical integrity under irradiation is important for unclad concepts. Fragmentation may lead to coolant or purge flow constriction. Alternate methods of fabrication that may alleviate these problems should be investigated.

Beryllium oxidizes readily at extremely low oxygen potentials. The kinetics of the oxidation process, particularly for mixtures of beryllium with lithium ceramics, may have important implications with regard to tritium inventory in beryllium.

VI.C. Lead Neutron Multiplier

It was concluded that there are no serious materials issues for the lead neutron multiplier. It is recognized that melting occurs if the temperature exceeds 327°C; however, this is a design issue and not a materials properties issue.

VI.D. 83Pb-17Li

Three primary issues were identified for the 83Pb-17Li eutectic alloy, which has been proposed as a tritium breeding materials in a few blanket concepts:

1. H_2O compatibility in event of a leak that would lead to oxidation of the lithium and possible safety consequences

2. segregation of the eutectic in low-temperature regions or during solidification

3. corrosion of the austenitic steel structure.

These are not considered serious feasibility issues; however, further R&D is required to validate PbLi blanket concepts.

VI.E. Ceramic Breeder Materials

Several materials constraints were identified for the candidate ceramic breeder materials, but no feasibility issues were defined. Primary candidate ceramic breeder materials include Li_2O, $LiAlO_2$, Li_4SiO_4, and Li_2ZrO_3.

1. Mechanical integrity and radiation-induced swelling may pose serious constraints. These properties are sensitive to fabrication method and limit the fluence and/or burnup.

2. Tritium transport/release at low temperature (<400°C) is of particular importance for the ITER blankets.

3. Chemical stability is a primary concern at high temperature, particularly for Li_2O and Li_4SiO_4.

4. Leakage of H_2O into the high-temperature ceramic breeder regions can have serious consequences.

VI.F. Aqueous Salts

The two lithium salts of primary interest are $LiNO_3$ and LiOH. The major materials issues for these salts involve various aspects of corrosion and stress corrosion of austenitic steels. Radiolysis is expected to be more critical for $LiNO_3$ than for LiOH; however, LiOH is more corrosive than $LiNO_3$. Severe SCC of Type 316L stainless steel is observed upon exposure to LiOH even without the additional effects of a fusion environment. However, SCC was not observed at 100°C. The sensitivity of austenitic steels to SCC must be determined in order to validate the aqueous salt concepts.

VI.G. ^3He Breeders

Two materials-related issues were identified for the innovative ^3He breeder concept that has been recently proposed. Tritium trapping/retention in the steel wall of the ^3He containment could be an important issue at low temperatures. At high pressure ^3He containment/leakage is a major concern.

VII. MECHANICAL/MAINTENANCE

The Mechanical/Maintenance Group reviewed the proposals with respect to the mechanical, maintenance, and reliability aspects. The available details of the blanket configuration were limited and the review focused on (a) the criteria by which to assess the blanket concepts; (b) the key mechanical, maintenance, and reliability issues; and (c) the R&D required to resolve the issues that will lead to the selection of a concept in the time frame of ITER.

VII.A. Criteria to Evaluate Blanket Design

The important features that blankets should exhibit in terms of mechanics, maintenance, and reliability were addressed separately, as follows (criteria that distinguish between proposed blanket concepts are marked by an asterisk):

VII.A.1. Mechanical Engineering Criteria for Assessment

1. **Coolant circulation.* The design of the blanket should be such that the coolant is able to circulate by natural convection.

2. **Materials.* The materials used for construction should be easily machined, formed, and welded.

3. *Design complexity.* The design should be such that it can be readily analyzed, many manufactured, and tested.

4. *Factors of safety.* Available knowledge of the materials and operating condition should be sufficient to allow low factors of safety.

5. **Pressure.* Operating pressure should be as low as possible.

6. **Temperature.* The design should avoid temperature gradients within the structure, absolute temperature should be as low as possible, and ΔT during operation should be minimized.

7. *Dimensional changes.* The design should be tolerant to changes in dimensions during operation.

8. *Segmentation.* The configuration of the machine should minimize the number of modules.

9. *Operational flexibility.* Blanket design should allow changes in the operating requirements such as may occur during the physics phase.

VII.A.2. Remote Maintenance Criteria for Assessments

1. *Maintenance time.* The design of the blanket should minimize the maintenance time, which includes inspection, installation of maintenance equipment, maintenance tasks, transportation, installation of components, and postmaintenance inspection.

2. *Structural attachment.* Attachment of the blankets to the machine should enable straightline assembly (preferably from above). Fasteners should be readily accessible. The attachment device should include aids for assembly and alignment.

3. *Weight.* The importance of component weight is dependent on the approach taken for assembly. Where horizontal assembly is necessary, it is very important to minimize weight. Where assembly is vertical, weight is not particularly important, providing it is within the capability of the handling equipment.

4. *Fit-up tolerances.* The fit-up tolerances should be as wide as possible using alignment aids to achieve the final accuracy.

5. **Feed connections.* The number of connections to the blanket should be minimized. The connections should be standardized and be compatible with the

overall machine. They should be well-proven under the required operating conditions.

6. *Preparation for maintenance.* The preparation for maintenance, which includes introducing systems to prevent the spread of contamination, or the introduction of a temporary coolant supply, must be simple and reliable.

7. *Postmaintenance inspection.* The blanket design must allow for extensive postmaintenance inspection.

VII.A.3. Reliability Criteria for Assessment

1. *Connections.* The number of feed connections to the blanket should be minimized.

2. *Fault detection.* The blanket should be such that a failure can be located and, if possible, isolated.

3. *Materials.* The blanket materials should be compatible in order to avoid corrosion.

4. *Welds.* Welds should be placed in locations where there is some protection from radiation damage. The total number and length of welds should be minimized. They should be located so as to facilitate postweld inspection.

5. *Cooling circuits.* There should be dual cooling circuits, if possible.

6. *Internal failure.* The blanket must be tolerant to internal rupture.

7. *Off-normal events.* The design must be tolerant to off-normal events, such as disruptions.

VII.B. Key Issues

With the exception of the LiPb NET blanket concepts, there was insufficient detailed information available to make a serious trade-off between the various proposed blanket concepts. Therefore, the key issue from the mechanical, maintenance, and reliability standpoint is to increase the level of information available. In particular, more attention must be given to (a) defining the physical configuration and design detail; (b) analysis, including stress, thermal, and fluid dynamics; (c) fabrication and inspection techniques; and (d) structural attachment to the machine.

VIII. SELECTION CRITERIA

The selection of a tritium breeding (driver) blanket for ITER will involve judgments of both qualitative and quantitative natures. Selection will likely be a two-phase process. During the definition phase of ITER, the roughly 12 options currently envisioned will be assessed, and two, perhaps three, options will be identified for further study. During the ITER design phase, a reference blanket, perhaps with a backup option, will be selected. In the selection process, it is anticipated that a set of selection criteria will be adopted as a bases for comparing blanket options. The Blanket Selection Criteria Group discussed possible criteria and evolved the set shown in Table II. The criteria given in Table II are divided into two groups, reflecting differences in relative importance. Thus, the "first-priority" criteria are judged to be more important in the selection process than the "second-priority" criteria. No quantitative weights were assigned to the two groups of criteria. The implication and scope of each criterion are discussed briefly in the following sections.

VIII.A. Reliability

There should be a high level of confidence that the blanket will neither compromise machine operation nor place the machine at risk. In evaluating blanket reliability, the following issues should be considered:

1. the number of welds required
2. the operating temperatures, pressures, and stresses
3. the simplicity of the configuration from engineering and fabrication perspectives
4. the consequence of failures within the blanket.

VIII.B. R&D

The blanket concept should present a modest level of unresolved issues such that the required R&D could be carried out consistent with the machine schedule and at reasonable cost. In assessing blanket R&D requirements, the following areas should be considered:

1. the materials data base
2. tritium recovery and control technology
3. manufacturing feasibility
4. the need for component testing.

TABLE II

Proposed Blanket Selection Criteria

First Priority
• Reliability • R&D • Safety perception
Second Priority
• Impact on machine design and cost • Robustness • Tritium breeding potential • Reactor relevance

VIII.C. Safety Perception

The blanket should not only satisfy engineered safety requirements but also should support public acceptance of fusion as a safe energy option. In determining safety criteria, the following issues should be considered:

1. the desirability and need for achieving passive safety in design
2. the establishment of public and occupational maximum exposure levels
3. the implications of evacuation plan requirements
4. the approach to waste disposal.

VIII.D. Impact on Machine Design and Cost

The blanket should impose only minor modifications to the basic machine size, configuration, and maintenance approach. The issues to be examined in this evaluation should include the following:

1. the impact of coolant choice
2. the consequence of interfacing the blanket with other machine components
3. the compatibility of the blanket with the test modules and the testing program.

VIII.E. Robustness

The blanket should be able to accommodate off-normal conditions (design margin) and variations in operating environment (design flexibility). In assessing blanket robustness, the following variations and situations should be examined:

1. operating conditions varying by ±50%
2. the impact of expected poloidal and toroidal variations in the environment
3. the ability of the blanket to operate with leakages
4. the consequence of plasma disruption.

VIII.F. Tritium Breeding Potential

Some credit should be assigned to blanket concepts that exhibit tritium breeding margin. Breeding margin should be evaluated in the context of the following issues:

1. breeding ratio achieved with outboard breeding alone
2. machine impact resulting from breeding in the inboard region
3. ease of breeding in and around divertor regions
4. requirements for neutron multipliers.

VIII.G. Reactor Relevance

Some credit should be assigned to blanket concepts that yield information relevant to DEMO conditions. In assessing reactor relevance, each of the following items should be considered:

1. coolant operating temperature and pressure
2. breeder temperature and temperature gradient
3. structure temperature and stresses
4. resource implications.

IX. NUCLEAR TESTING

The "terms of reference" for ITER provide a reasonable definition of a device of testing of fusion nuclear components. These terms of reference call for an average neutron wall loading of 1 MW/m^2, a machine design to accommodate up to an average neutron fluence of 3 $MW \cdot yr/m^2$, an overall availability of ~10%, availability during years of peak reliability of 25%, and periods of very high availability (~100%) for 1 to 2 weeks.

It was agreed that steady-state operation in the technology phase of ITER is highly desirable for nuclear testing. However, useful testing can also be done during pulsed operation. It is desirable to have, in particular, short dwell times as well as long burn times. Typical burn times ranged from 200 to 800 s, with dwell times not longer than 70 s.

It was generally agreed that the wall loading in ITER should not be less than two to three times the wall loading expected in the reactor after ITER (usually referred to as DEMO) in order to provide reasonable engineering scaling. The desired wall loading in ITER for testing is then dependent on the perceived wall loading in DEMO. In general, attention should be given to the design of test devices coming after ITER when defining what ITER should look like.

All classes of blankets should be considered at this time as candidates for testing in ITER. Two basic types of blanket test geometries were discussed: (a) segment tests that extend from the top to the bottom of the machine with a toroidal dimension at the midplane of typically 1 m, and (b) module tests with a typical first-wall area of 1 to 2 m^2 and a radial thickness of ~0.5 m. It is desirable to have access for two segment tests and from two to about six module tests. It is reasonable to consider running about four (i.e., two segment and two module) tests at the same time. This would require four testing loops. Solid breeder blankets are best suited for module tests while liquid blankets should be tested in segments. Module tests can also be a step prior to segment testing. The exact number of module and segment spaces available will depend on the design of ITER and space requirements for impurity control, plasma heating and fueling, diagnostics, etc.

Safety-related tests were discussed. It was generally thought that limited safety tests of blankets could be performed, probably late in the life of ITER. Hybrid blanket tests were also discussed. While there appear to be no technical reasons why such tests could not be done in ITER, some participants expressed reservations about such tests for programmatic reasons.

It was generally viewed that materials tests would be performed in ITER to investigate both plasma/surface interactions and neutron effects. Plasma/surface tests will require access at the first wall (e.g., as with a blanket module at the midplane) or perhaps in a segment of the divertor plate. Blanket modules without access directly to the plasma (i.e., with a separate first wall) will be useful for some initial blanket tests, but blanket access to the plasma will be required for prototypical test conditions.

It was also agreed that some nuclear-related tests should be done during the physics phase of ITER. Examples include neutronic/shielding tests, plasma surface tests, tritium system (without blanket processing), liquid-metal MHD, instrumentation checkout, and selected low fluence materials irradiations.

The question of changing out a substantial portion of the shield/blanket between the physics and technology phases was discussed. While this appears to be possible, more studies are required to assess the practicality of such change out.

It should also be noted that much useful information will be obtained from monitoring the performance of the basic structure and shield of the ITER device in addition to test segments and modules.

In summary, there were remarkably similar views on nuclear testing in ITER among the various parties at the workshop.

These papers will be published in a special issue of *Fusion Engineering and Design*.