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U.S. ITER SHIELD AND BLANKET DESIGN ACTIVITIES*

by

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U.S. ITER SHIELD AND BLANKET DESIGN ACTIVITIES*

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ABSTRACT

This paper summarizes nuclear-related work in support of the U.S. effort for the International Thermonuclear Experimental Reactor (ITER) Study. Primary tasks carried out during the past year include 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 blanket concepts considered are the aqueous/Li salt solution, 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.

RESULTS OF PRELIMINARY BLANKET/SHIELD EVALUATION

Several options for an ITER tritium-production blanket/shield were examined and are listed in Table 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, 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 2.

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.
- 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 higher temperatures than low-temperature water which may be more optimum (if not too high) with respect to matching the available data base.
- 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

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Table 1
U.S. ITER Blanket/Shield Options

Form of Incorporating Breeder Material into Blanket	*Type of Coolant*	
	Water	*Helium*
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 2
Some Key Particular Issues for ITER Blanket/Shield Options*

Self-Cooled Aqueous/Salt	-	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	-	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	-	predictable gap conductance between clad and breeder, including irradiation effects
	-	irradiation effects on solid breeder materials
	-	helium containment and manifolding
Li Particulates in Helium	-	helium containment and manifolding
	-	erosion of helium ducts and helium pumps
^{3}He Purge Stream	-	helium containment and manifolding
	-	source of ^{3}He

* 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.

sufficient time to do a detailed 3-D neutronic analysis based on a common approach.

DESIGN GUIDELINES

The main purpose of the design guidelines[1] 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.

EVALUATION CRITERIA

Evaluation criteria[1] 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 is based on a somewhat more quantitative definition which measures a particular attribute of an evaluation criteria.

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.

MATERIALS ASSESSMENT

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°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°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 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 the U.S. 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.

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/cm^2 above 1 MeV which corresponds to > 10,000 appm He in beryllium and a first wall fluence of ~ 4 MW.y/m^2 in ITER.

Tritium buildup in neutron-irradiated beryllium is also an issue. Both ^{6}Li, ^{7}Li and

^{3}He 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, 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.

Solid Breeder Materials

Some of the materials issues of solid breeder blankets are discussed in the sections on water and helium-cooled solid breeder blanket options.

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 $H_2O/LiNO_3$ 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, 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/cm^2 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.

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 $LiNO_3$ 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 cm^3 for $LiNO_3$ and 5 g/100 cm^3 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/cm^3.

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. Here "low" activation refers to long-term activation related to waste management. 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 subject to stringent control of impurities. 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 ^{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 same rate leading to a coolant specific activity in the reactor of only 0.05 Ci/m^3 of ^{14}C for class C waste. The volume of operational waste to be disposed of annually is 50 m^3 for $LiNO_3$ and 0.6 m^3 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 (PCA) 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 but at a possible substantial increase in the cost of the tritium processing system. This can be achieved by using the aqueous solution only in the front zones of the outboard blanket and inboard shield.

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 (Li_2O, $LiAlO_2$, or Li_4SiO_4). 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.

The solid breeder must operate 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.

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 cladded 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.

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.

HELIUM-3 BLANKETS

In this approach, terrestial 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 he-

lium 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 m^3, 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 out-gassing. 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 terrestial 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.

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