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AN ASSESSMENT OF THE BASE BLANKET FOR ITER

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Abstract

Ideally, the ITER base blanket would provide the necessary tritium for the reactor to be self-sufficient during operation, while having minimal impact on the overall reactor cost, reliability and safety. A solid breeder blanket has been developed in the CDA phase in an attempt to achieve such objectives. The reference solid breeder base blanket configurations at the end of the CDA phase has many attractive features such as a tritium breeding ratio (TBR) of 0.8-0.9 and a reasonably low tritium inventory . However, some concerns regarding the risk, cost and benefit of the base blanket have been raised. These include uncertainties associated with the solid breeder thermal control and the potentially high cost of the amount of Be used to achieve high TBR and to provide the necessary thermal barrier between the high temperature solid breeder and low temperature coolant.

This work addresses these concerns. The basis for the selection of a breeding blanket is first discussed in light of the incremental risk, cost and benefits relative to a non-breeding blanket. Key issues associated with the CDA breeding blanket configurations are then analyzed. Finally, alternative schemes that could enhance the attractiveness and flexibility of a breeding blanket are explored.

Considerations in ITER Blanket Selection

A blanket will be required for ITER. It will need to perform heat removal functions as well as neutron capture functions. The question is whether it will also perform tritium breeding functions. Selection of an ITER breeding blanket over a non-breeding blanket has to be based on considerations of incremental risks, costs and benefits.

Risks

Major issues in this area are associated with reliability and safety.

A breeding blanket will require additional regions for the breeder and multiplier as well as a purge system, which would result in more complex design and assembly requirements. However, it is not clear that a breeding blanket would have a significantly lower reliability than a non-breeding blanket. For instance, Figures 1 and 2 show examples of a breeding blanket configuration designed as part of the CDA effort [1], and of a non-breeding blanket configuration from a pre-design study for NET [2]. Both of these configurations use a layered design. From these example configurations, there seems to be a comparable number of welds and joints in contact with the coolant in the higher fluence regions. Furthermore, failure that would require reactor shutdown and component replacement and, thus, which could have a major impact on the overall reactor availability are mostly concerned with: coolant channel rupture or blockage; and structure embrittlement and failure. These issues are common to both breeding and non-breeding blanket.

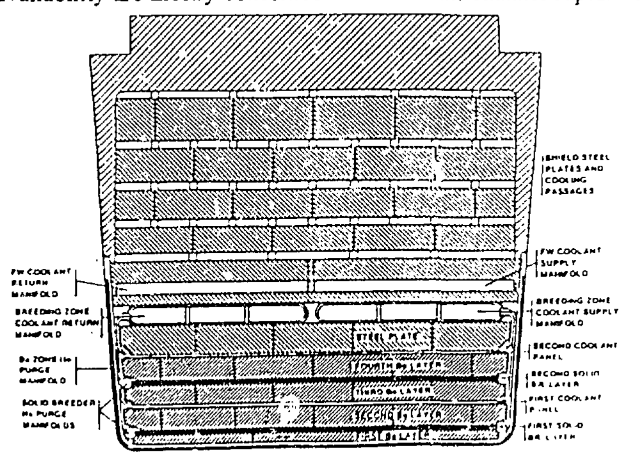

Figure 1. Cross-sectional view of a multi-layered ceramic breeder blanket design configuration developed during the ITER CDA phase [1].

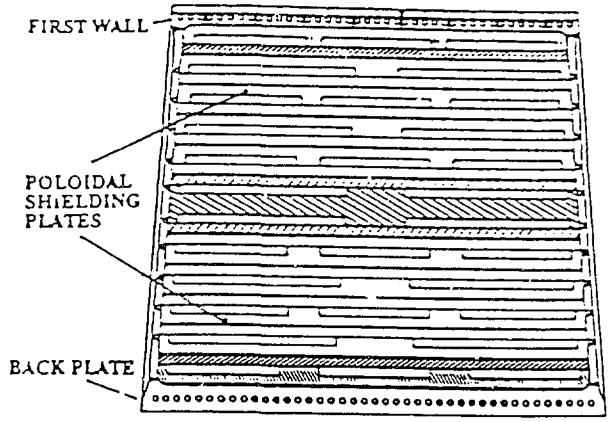

Figure 2. Cross-sectional view of a "welded plates" concept for the NET shielding blanket [2].

A key safety issue is accidental tritium release, which leads to the need of minimizing the tritium inventory components. Much of the tritium inventory, for instance, in the fueling system, would be unaffected by the blanket choice. However, a non-breeding blanket would increase accumulative tritium transportation inventory and storage inventory, whereas a breeding blanket would result in a finite blanket tritium inventory. Another safety issue relates to loss-of-flow or loss-of-coolant conditions. A non-breeding blanket confuguration would consist only of structure and coolant, resulting in higher activation and afterheat which would require corrective actions sooner than a breeding blanket.

Costs

The cost of Be, in particular for blankets making intensive use of Be, and the cost of the tritium extraction system are major parts of the incremental capital cost associated with a breeding blanket. This has to be compared to the additional tritium purchasing cost in the absence of tritium breeding, presently estimated at about \$1200M per MW-a/m^2 of operation assuming a unit purchasing price of \$29K/g.

For solid breeder blankets using Be as a multiplier, it is possible to minimize the amount of Be while still achieving good tritium breeding. For example, for configurations with a separate Be region in front of the breeding region, the maximum TBR is achieved with a thickness of about 6.5 cm of fully dense Be, whereas for configurations where the Be is homogeneously mixed with the solid breeder, the maximum TBR is obtained for a Be/SB volume fraction ratio of about 80/20 [3,4]. Based on the total thickness of solid breeder layers in the two variants of the CDA layered blanket configuration [1], this corresponds to a thickness of about 6.8 or 10.2 cm of fully dense Be. For ITER application, this results in a Be mass of about 60 tons. At \$600/kg, the cost of Be will then be about \$36M, which is less than 5% of the total reactor system cost and less than 1% of the total reactor cost.

MASTER

Benefits

The benefits from operating a breeding blanket relate to the value of the information obtained based on application to DEMO and commercial reactors. This includes: information on integrated behavior of solid breeder blankets; crucial data on tritium self sufficiency which cannot be obtained from test modules; information on performance and reliability of blanket and auxiliary systems ; and substantial information and experience that will be needed for DEMO qualification

DISCLAIMER

Based on the above discussions, there seems to be clear benefits for operating a breeding blanket in light of the application to DEMO and commercial reactors. The incremental capital costs seem reasonable, particularly when considering the minimization of the amount of Be while still providing adequate tritium breeding. There is a trade-off involved regarding the amount of tritium required and its costs as a function of the machine fluence. The incremental risk associated with a breeding blanket seems minimal in terms of the overall effect on the reactor availability. In terms of tritium risk, a non-breeding blanket would result in higher storage and transportation inventories, whereas a breeding blanket would result in a finite tritium inventory in the blanket. The key question, thus, is whether the tritium generated in the blanket will come out fast enough so that the resulting blanket inventory is kept at a reasonable level and the bred tritium can be recovered for fueling the reaction.

Breeding Blanket Configuration

During the Conceptual Design Activity (CDA) phase of ITER, the ceramic breeder concept has been selected as the first breeding blanket option. Since power production is not an objective, the coolant can be kept at low temperature and low pressure based on safety and reliability considerations. Water at about 60-100 C was specified as coolant, and austenitic steel, Type 316 solution annealed, as structural material. For a solid breeder design requiring high breeder temperature for tritium release, this means that a thermal resistance region must be provided between the breeder and coolant.

Two configurations were considered in detail: a multilayer configuration, as illustrated in Fig. 1, consisting of solid breeder layers separated from the water coolant by a sintered-block or packed-bed Be region; and a breeder-in-tube configuration, consisting of poloidal tubes in which the solid breeder region is separated from the water coolant by a helium gap [1]. Both configurations use Be as neutron multiplier and highly enriched lithium ceramic (Li_2O or a ternary ceramic: $LiAlO_2$, Li_2ZrO_3 or Li_4SiO_4). Calculations indicate that the net tritium breeding ratio is about 0.8-0.9.

Data from small-scale fission-reactor experiments indicate good tritium release from solid breeders over a wide range of temperatures [e.g.5-8]. For example for Li_2O, an acceptable temperature range for tritium release is about 350-800 C, and up to 1000 C under certain conditions. Estimates of the tritium inventory in the blanket based on experimental data and on existing models indicate an inventory of 10 g or less under normal operating conditions. The tritium generation in Be is estimated at about 1 kg over the ITER lifetime. Recent experimental data indicate better release than previously thought and the possible dependence of tritium release on the Be microstructure which could be tailored for optimal release [9]. However, a purge flow would be required for the Be region also.

Breeding Blanket Issues

The key issues which emerged from the design analyses of these configurations can be broadly listed as follows:

1. Solid breeder thermal control. The required thermal insulation between the solid breeder and coolant, as shown in the schema of Figure 3, is provided either by a helium gap of a Be region using sintered block or packed bed form. The key concern is to maintain the desired thermal resistance during operation, and mechanisms which offer the possibility of accommodating large power variation would be more robust in allowing for uncertainties in parameters during operation that might affect the thermal resistance of the region. In addition, any means of active thermal control would significantly enhance the blanket operating flexibility.

This issue is discussed in detail in Ref. [10]. To help in determining the thermal control performance of each mechanism, calculations were done based on ITER-relevant parameters. A 1-cm thick Li_2O solid breeder region was considered, sandwiched between two thermal control regions. Water coolant at 70 C flows on the other sides of the thermal control regions. A solid breeder temperature rise of 100 C, a heat flux of 0.1 MW/m^2 from the solid breeder to the thermal control region, and a film temperature drop of 30 C were

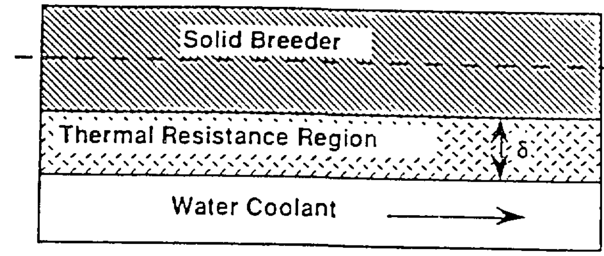

Figure 3. Schema of thermal control region between high-temperature solid breeder region and low-temperature water coolant.

assumed for a reference first wall load of 1.2 MW/m^2. The minimum and maximum allowable Li_2O temperatures were set at 400 C and 1000 C respectively and the interface conductance was assumed to be 2000 W/m^2-K. The results are briefly reported below.

The helium gap provides for a reasonable range of power variation. For the ITER-relevant conditions assumed in Ref.[10], a 1.5-mm He gap would accommodate first wall loads of 0.55 to 2 MW/m^2 excluding radiation. Radiation could significantly extend the wall load range to a maximum of 0.68-3 MW/m^2 depending on the surface emissivity. However, the gap size is small raising the concern of maintaining close tolerance during manufacture and operation. This concern is exacerbated for a poloidal configuration where the gap size must vary poloidally to account for spatial power density variation and where the gap size must be maintained while bending the tubes to accommodate space restriction or reactor contour.

Under similar conditions a 5.3-cm Be sintered block region would accommodate a wall load range of 1-2 MW/m^2. Concerns arise mainly from the high ratio of Be to He thermal conductivity (of the order of 500), whereby formation of a small gap at the Be/clad interface could significantly affect the overall thermal resistance of the region. Of particular concern are the Be block deflection under differential thermal expansion and the predictability of the Be/clad interface contact resistance during operation.

A 1-cm Be packed bed region under similar conditions would accommodate a wall load range of 0.82-1 MW/m^2. One of the advantages of a packed bed is the possibility of active control through gas pressure adjustment of the bed thermal conductivity. If provision for varying the He pressure from 0.02 to 0.2 MPa is provided, the allowable wall load range can be extended to 0.58-2 MW/m^2. Concerns include the apparent sensitivity of the bed effective thermal conductivity to small changes in the packing fraction, in particular for binary beds [11]. This means that extra care would be needed during assembly to provide the bed packing fraction called for in the design.

2. Solid breeder and multiplier material characterization. Many of the design uncertainties stem from the lack of property data, particularly for irradiation conditions. For example, the tritium inventory in the solid breeder is estimated as being reasonably low. However, effects such as LiOT precipitation at low temperature and mass transfer at high temperature, and irradiation-induced trapping and microstructure changes have to be better understood and addressed through a series of laboratory as well as in-reactor experiments. Fundamental data on basic surface processes under different conditions are also required, so that model predictions for the blanket tritium behavior could be based on accurate property data for individual transport mechanisms. The models would then be calibrated by analyzing data from small-scale integrated experiments before extrapolating to the range of ITER conditions.

Similarly for Be, irradiation effects such as swelling, tritium retention and compatibility with the structure have to be experimentally determined. Concerns about Be-steam reaction under accident conditions, which could pose significant hydrogen generation risks, need to be addressed.

3. Structural material performance.

Issues relate mainly to aqueous stress corrosion and irradiation effects on low temperature fracture toughness of Type

,316 steel. Indications are that aqueous stress corrosion tends to be low at low temperature whereas embrittlement and irradiation creep tend to be more severe at low temperature. Irradiation effects on mechanical properties and on welded and brazed joints also need to be better characterized.

Configuration Assessment

The above-mentioned issues are important, and an R&D program is underway to address them. However, in view of the consequences associated with them and of the existing range of alternative materials, configurations and mechanisms, they are not judged as being feasibility issues, but rather engineering issues.

Within the blanket configurations considered during the CDA phase, alternatives already exist reducing the impact of some of the key issues. For example, three mechanisms for providing thermal control of the solid breeder have been proposed. For the case of the Be region providing thermal control, if the blanket cost driven by Be is an issue, use of the packed bed form results in a significant reduction in the amount of Be. The packed bed form offers also the potential for active control which is a most attractive feature for accounting for uncertainties during operation. Use of Be as neutron multiplier also provides the flexibility of designing a breeding blanket using highly enriched ^{6}Li and switching to pure ^{7}Li for minimal tritium production if further R&D casts a shadow on the expected tritium release from the solid breeder. This follows from the high neutron energy threshold for ^{7}T production which will be minimal based on the low energy of multiplied neutrons. This would result in a low breeding blanket with the same configuration as a breeding blanket and with lower afterheat generation when compared to a non-breeding blanket.

Other alternatives that offer attractive features which would help resolve key issues are: use of ^{7}Li as thermal control region to relieve the concern of having Be perform both neutron multiplying and thermal control functions; and use of He as coolant which would relieve the need for and concerns associated with a thermal control region, as well as safety concerns associated with the Be-steam reaction under accident conditions. These alternatives are discussed below in light of the benefits provided and of issues raised.

^{7}Li Thermal Control Region

A possible thermal control mechanism that would resolve the concerns associated with the uncertainty in the thermal performance of Be and with the maintenance of a thin He gap is a ^{7}Li region. Basically the region is a continuation of the solid breeder region but includes only ^{7}Li which shows minimal tritium breeding in the presence of multiplied neutron. This results in the elimination of one clad and of the associated interface conductance uncertainty.

A similar calculation to those done for the other thermal control mechanisms was performed to evaluate the allowable wall load range as a function of the thickness of the ^{7}Li region. The calculations were based on the ITER-relevant parameters described in the previous section. The configuration consisted of a highly-enriched Li_2O sintered-block breeding region separated from the water coolant by a packed bed region of Li_2O with ^{7}Li only. The results are summarized in Figure 4. For a reasonable thermal-control region thickness of about 0.8 cm, the allowable wall load range is about 0.7-1.7 MW/m^2.

One concern of this design is that the small amount of tritium produced in the pure ^{7}Li region would be trapped. Neutronics calculations indicate that for the tube dimensions assumed above, about 5% of the tritium production is bred in the pure ^{7}Li region. However, if under normal conditions the highly enriched Li_2O region operates between about 500 C and 600 C , a good part of the pure ^{7}Li region would also be at a temperature higher than 400 C resulting in part of the tritium produced there being released. Any period of higher temperature operation would also release a substantial part of the tritium from that region. Thus, it is expected that the total tritium inventory there would be of the order of 1-2 kg over the life of ITER (3 FPY).

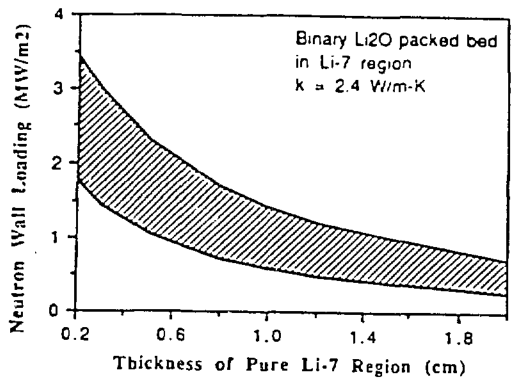

Figure 4. Range of allowable neutron wall load as a function of pure ^{7}Li thermal control region thickness.

He Coolant

At this stage, it seems reasonable to consider again the possibility of He as a coolant. Use of He would help resolve some of the key issues associated with the breeding blanket, such as thermal control and Be-steam reaction under accident conditions. Helium would be operated at a high enough temperature so that no thermal control region would be required between the coolant and breeder. In addition, flexibility exists to set the inlet temperature of helium at a range of levels in order to optimize tritium release. Other advantages of He include: the provision of more reactor-relevant information since most recent commercial power reactor design studies use He as coolant (e.g. ARIES[12] and PROMETHEUS[13]); and the possibility to boost TBR if required by flowing ^{3}He although at the cost of processing the tritium out of the coolant.

Two major reasons for the consideration of water instead of helium for the coolant were the concern of high pressure operation and of helium leakage. However, ITER power density is much lower than that of previously studied power reactors, and studies have shown that He could be operated at moderate pressure in ITER to enhance reliability and safety. A moderate pressure would also reduce any tritium leakage. For example, Ref. [14] shows that He coolant at 1.5 MPa could be used in the breeding blanket of ITER, even for first wall cooling. Ref. [13] shows an ICF commercial blanket design using low pressure He (1.5 MPa) by taking advantage of the fact that a substantial part of the power is taken by the first wall protection system. An example calculation was done based on an effective power deposition of 1.8 MW/m^2 in the blanket. Figure 5 shows the helium pressure drop as a function of the inlet pressure and the flow area for a blanket module in which helium flows toroidally in channels of thickness 0.5 cm and length 2 m. The helium temperature rise is set to the low value of 100 C for better uniformization of the blanket temperature and tritium release. For a 1.5 MPa inlet pressure and a fractional flow area of 6% (which corresponds to a total helium volume fraction of 12% when considering both He inlet and outlet), the pressure drop is 47 kPa. This calculation assumes commercial piping roughness. This blanket pressure drop is quite reasonable and even if the external cooling system pressure drop is included, the total value would still be about 5-10% of the inlet pressure. The corresponding helium velocity is 45 m/s. Note that the helium void fraction could be reduced by relaxing the temperature rise requirement or by allowing for higher pressure drop.

Key issues that need to be addressed and put into perspective are the possible requirements of a separate first wall water cooling system, helium leakage from the cooling lines, and incremental shielding and manifolding sizes. At this stage, a separate first wall cooling system already exists, the question with helium being that a different coolant might be needed. A helium first wall coolant would be advantageous for bake-out. Concerns arise when using super-heated steam at high pressure for this purpose. Thus, even if water is used as coolant for the first wall, it is possible that hot He might be circulated through the cooling system prior to operation to bake-out the first wall. A key issue is the effect of regular dry-out of the water

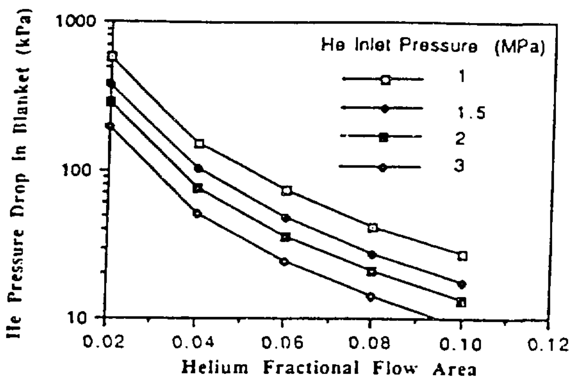

Figure 5. He coolant pressure drop in ITER blanket as a function of fractional flow area for different inlet pressures, assuming an effective power deposition of 1.8 MW/m^2 and inlet and outlet temperatures of 250 C and 350 C respectively.

system on corrosion and structural performance of the cooling system. The extent of He leakage has to be assessed although the moderate operation pressure would tend to result in lower leakage than for the usual high pressure He cooling system. Additional shielding and larger manifolds will be required for helium and would depend on the fractional flow area. This is mainly a problem for the inboard where space is restricted and where inclusion of a breeding blanket is more problematic.

Summary

Placing a breeding blanket in ITER is likely to have minimal incremental effect on reliability since failures requiring shutdown and component replacement are associated with coolant channel blockage or rupture and structure embrittlement, which are essentially the same for both breeding and non-breeding blanket cases. The economics of a breeding blanket compared to a non-breeding blanket depends on the level of fluence desired. If most of the incremental cost of a breeding blanket is due to Be, its volume fraction can be minimized while keeping the tritium breeding ratio at an acceptable level. On a purely economic basis, the tritium purchasing cost for 0.1 MW-a/m^2 of operation is estimated at about $120M, which would have to be weighed against the incremental cost for a breeding blanket. Other considerations such as the reactor-relevant information obtained and the future R&D cost saving increases the attractiveness of a breeding blanket. For device fluences significantly greater than 0.1 MW-a/m^2, the benefits of a breeding blanket are clear.

Issues exist with a breeding blanket. However, the depth of blanket design is considerable in view of the maturity of the field and of the number of alternate schemes and configurations developed over the years. Consequently, the issues are viewed more as engineering issues to be addressed by an R&D program rather than feasibility issues. For example, a key question regarding the breeding blanket relates to the tritium inventory in the solid breeder and Be. Experimental evidence tends to show that tritium will come out of the solid breeder quite readily over a wide range of temperatures. However, if low temperature coolant is used based on reliability considerations, a thermal control region is required between the solid breeder and coolant. There are issues that relate to the predictability of the thermal conductance of such a region. Three mechanisms have been considered as part of the CDA effort and each is found to provide substantial flexibility in accommodating power variation or uncertainties. In addition, the Be packed bed mechanism offers the possibility of active control through gas pressure adjustment.

Another thermal control mechanism using a pure ^7Li solid breeder region was explored. It provides for separation of neutron multiplication and thermal control functions and results in a reasonable region size. It also eliminates one clad and the associated interface contact uncertainty. However, a major concern is the end-of-life tritium inventory which is estimated at 1-2 kg for 3 FPY of operation.

Risks associated with the breeding blankets must be weighed against risks associated with the absence of a breeding blanket. For example, hazards associated with tritium transportation to the site depends strongly on the ITER site and the location of the tritium source. It is clear though that from an accident standpoint, the larger the amount of tritium that needs to be shipped the larger is the transportation risks.

Consideration of He as coolant would substantially reduce the perceived risks associated with thermal control and the associated tritium release concern and with the possibility of Be-steam reaction under accident conditions. He coolant would also provide more reactor-relevant information. The decoupling of the pressure and temperature of helium allows selection of operating temperatures which are optimum for providing adequate tritium release from the breeder without the need of a thermal control region and for minimizing radiation effects in the structure. For example, based on ITER power density, He could operate at a moderate pressure of 1.5 MPa and at temperatures of about 250-350 C. This is beneficial from the point of view of embrittlement and radiation creep of austenitic stainless steel. Concerns relate mainly to helium leakage, first wall cooling, additional shielding and larger manifolding.

Acknowledgement

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