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AN AQUEOUS LITHIUM SALT SELF-COOLED BLANKET AND SHIELD FOR ITER

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ABSTRACT

A low cost low risk tritium breeding concept has been developed for ITER. This concept is based on dissolving 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. The design that maximizes the outboard tritium breeding ratio utilizes a 40 cm thick zone of Be balls followed by an 80 cm thick zone of steel balls. Single size balls are used to minimize the pressure drop. The overall TBR excluding tritium bred in the test modules is 1.1. The inlet coolant temperature is 40°C and the temperature rise is 15°C in the first wall and 35°C in the blanket.

INTRODUCTION

In preparation for the international effort on the design of the International Thermonuclear Experimental Reactor (ITER), the U.S. fusion community developed a compact reactor design for ITER with a major radius of 4 m and an aspect ratio of 2.86. It produces 631 MW of fusion power leading to average and peak neutron wall loadings of 1.15 and 1.91 MW/m^2, respectively. During the past year several tritium breeding blanket options were considered for use in ITER.

The aqueous Li salt self-cooled breeding shield concept used in the TIBER-II[1] design was analyzed for the U.S. ITER design. This concept is based on dissolving small amounts of lithium compounds in the water coolant.[2] This makes it possible to breed tritium in all the shield zones, and results in tritium self-sufficiency as well as enhanced magnet protection due to strong neutron absorption in lithium. This is of particular significance in a compact technology test reactor such as ITER where the space available for separate breeding blankets is limited due to the large fraction of the outboard region devoted to test modules and plasma heating penetrations and because of the negative impact on magnet protection of using separate breeding blankets in the thin inboard and divertor shield zones. This concept also has the attractive feature of flexibility with the Li salt added to the coolant only when tritium breeding is required. Using this concept in ITER is also inspired by Ontario Hydro's demonstrated capabilities of removing tritium from water at acceptable concentrations and cost.

In this paper we will describe the aqueous Li salt blanket and shield design proposed for ITER with emphasis on nuclear and thermal-hydraulics analyses. The key technical issues associated with this concept will be identified and proposed future improvements to mitigate the present design shortcomings will be discussed. Detailed activation analysis and inboard shield design are given in separate papers in these proceedings.[3,4]

NUCLEAR ANALYSIS

Many Li compounds have high solubility limits in water allowing for significant tritium breeding. A major consideration in the choice of Li salt is the compatibility with structural materials and neutron multipliers. Other considerations include the tritium breeding and shielding performance, induced radioactivity, radiolysis, and impact on tritium recovery and containment. Our results show no significant difference in breeding potential when salts with high solubility are used close to their solubility limits. Among the different Li salts lithium nitrate and lithium hydroxide are identified as the primary candidates. Preliminary corrosion experimental results[5] indicate that while LiOH is more corrosive at high temperatures, both salts yield comparable corrosion rates at temperatures below 100°C. Both Li salts are analyzed here for use in ITER. However, because of the large amount of ^{14}C production and the radiolysis problems associated with $LiNO_3$, it appears that LiOH is a preferred candidate for use at the low temperatures envisaged for ITER.

The outboard blanket and shield concept used here employs a front multiplier pebble zone followed by a steel pebble zone. Both zones are cooled by the aqueous Li salt solution.

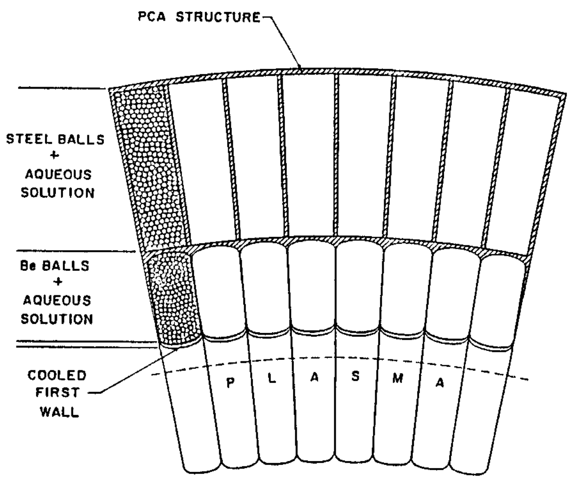

Fig. 1. Midplane cross section of blanket and shield

Figure 1 is a midplane cross section of a blanket and shield segment. Thermal hydraulics considerations resulted in using a double first wall that insures uniform cooling and minimizes system pressure.[1] A 2 cm thick first wall with 0.5 cm thick PCA layers surrounding a 1 cm thick coolant zone is used in the analysis. While a high packing fraction of the neutron multiplier required for enhancing tritium breeding can be achieved by utilizing different size balls, a design with single size balls of beryllium and PCA is used in order to minimize the pressure drop. Using balls in the design makes it possible to accommodate the complicated ITER geometry with the numerous penetrations. A 10 μm coating of PCA is used for the Be balls with the $LiNO_3$ solution. Due to the larger potential for corrosion of Be with the LiOH solution a 250 μm PCA cladding is used. The composition used for the front Be zone is 61.13% Be (0.9 density factor), 2.12% PCA, and 36.75% $LiNO_3$ solution. For the LiOH case the composition is 58.19% Be, 5.06% PCA, and 35.63% aqueous solution. The Li is enriched to 90% ^{6}Li and the results are normalized to the peak neutron wall loading of 1.91 MW/m^2 and total reactor life of 2.9 full power years (FPY). The neutronics calculations have been performed using the discrete ordinates code ONEDANT[6] with cross-section data based on the ENDF/B-V evaluation.

The effect of Li salt concentration on the outboard local tritium breeding ratio (TBR) is given in Fig. 2 for both the LiOH and $LiNO_3$ solutions. The concentrations of the Li salts are varied up to their solubility limits in cold water. Nearly the same TBR is obtained for $LiNO_3$ concentrations in the range 16-40 g/100 cm^3 and only a small increase (< 3%) in TBR is obtained by increasing the LiOH concentration from 5 g/100 cm^3 to its solubility limit. Concentration reduction below 16 g $LiNO_3$/100 cm^3 or 5 g LiOH/100 cm^3 leads to sharp decrease in TBR. These concentrations are chosen in this design to reduce corrosion and activation products without compromising the tritium breeding capability. The effect of Be zone thickness on TBR is shown in Fig. 3 for a

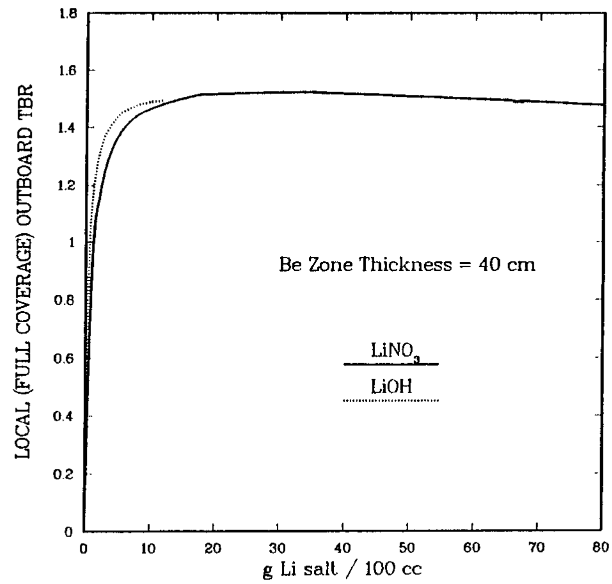

Fig. 2. Effect of Li salt concentration on local outboard TBR.

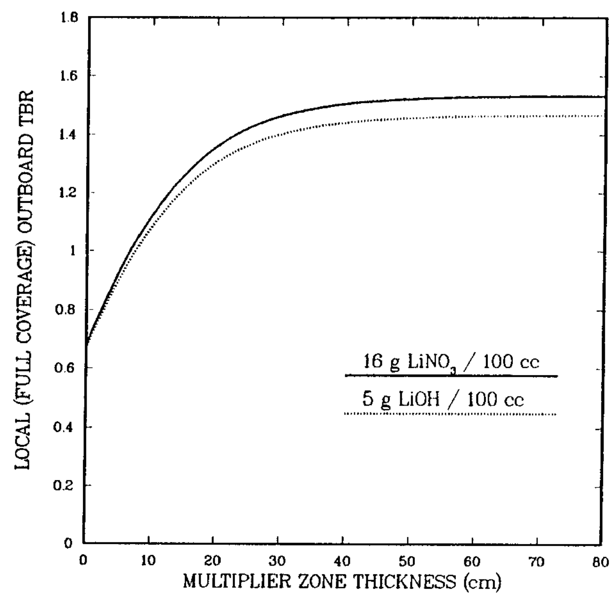

Fig. 3. Effect of Be zone thickness on local outboard TBR.

Table 1. Nuclear Parameters for the Outboard Aqueous Blanket and Shield

Li Salt	$LiNO_3$	LiOH
Salt concentration (g/100 cm^3)	16	5
Local TBR	1.506	1.442
Local energy multiplication	1.366	1.364
Power density at midplane (W/cm^3)		
First wall	15.20	15.85
FW coolant	24.97	24.48
Second wall	12.83	13.61
Front of Be zone	13.28	13.28
Average in Be zone	4.23	4.22
Front of SS zone	0.76	0.75
Average in SS zone	0.064	0.064
Peak dpa/FPY in first wall	16.86	16.75
Peak He appm/FPY in first wall	190.1	189.6
Peak fast neutron fluence in superconductor at 2.9 FPY (n/cm^2)	1.02×10^{16}	1.12×10^{16}
Peak insulator dose at 2.9 FPY (rads)	1.08×10^{7}	1.21×10^{7}
Peak winding pack power density (mW/cm^3)	3.81×10^{-3}	4.24×10^{-3}
Peak Cu dpa/FPY	2.88×10^{-6}	3.2×10^{-6}

total Be and steel zone thickness fixed at 1.2 m. Increasing the Be zone thickness beyond 40 cm results in a negligible enhancement in TBR with increased Be cost. A multiplier zone thickness of 40 cm is used in the baseline design. We investigated the impact of replacing Be by Pb as the neutron multiplier. This leads to ~ 25% decrease in magnet radiation effects in the outboard region. These are ~ 3 orders of magnitude less than the peak values in the

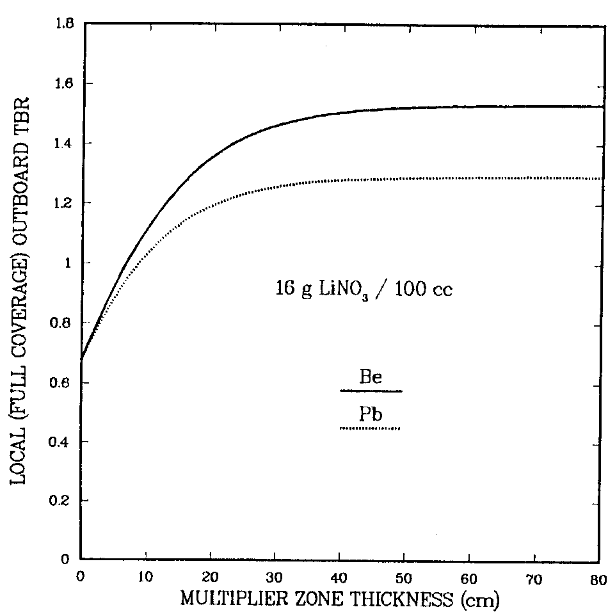

Fig. 4. Effect of multiplier material and thickness on local and outboard TBR.

inboard region. In the meantime, the TBR is reduced by 15% as shown in Fig. 4. Because of its reduced breeding potential in addition to the heavy weight, large creep, low modulus of elasticity, low strength, high stress corrosion cracking in water, low melting point and safety concerns related to polonium production, lead is not considered in this design and Be is used as the neutron multiplier. Table 1 lists the nuclear parameters for the outboard aqueous blanket and shield baseline design. The parameters are given for both cases of LiOH and $LiNO_3$ aqueous solutions.

While most of the overall TBR is obtained from the outboard blanket and shield, some additional breeding will occur in the aqueous solution used as coolant in other zones. In addition, the impact of neutron reflection from these zones on tritium breeding in the outboard region has to be considered. This is properly assessed by performing neutron transport calculations using a toroidal cylindrical model. Several toroidal cylindrical geometry calculations have been performed using the ONEDANT[6] code with different materials being used in the inboard region of the model to simulate the effect of the different inboard and divertor shield zones as well as penetrations. The results of these calculations were then corrected for the actual coverage fractions of the different breeding zones (55.65% outboard blanket, 16.98% divertor shield, and 18.6% inboard shield), calculated by the NEWLIT[7] code, to determine the overall TBR as illustrated in Fig. 5. This approach was previously used in the early stages of the design of the HWL reactor[8] and TIBER-II[1] and found to agree with

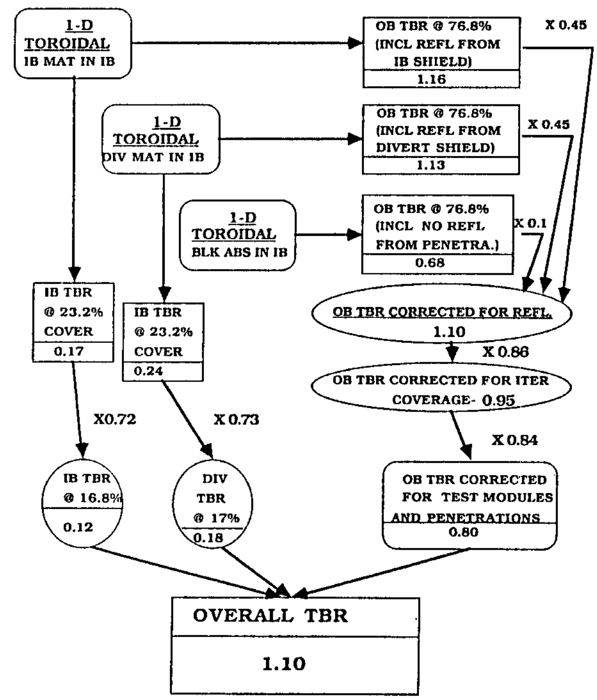

Fig. 5. Estimation of the overall TBR using the aqueous blanket.

the three-dimensional results to within ~ 4%. The estimated TBR of 1.1 is conservative since penetrations and plasma heating zones are treated as black absorbers. In addition, no credit is taken for breeding in the test modules. They will contribute about 0.07 to the overall TBR assuming a local TBR of ~ 1.1 in the test modules. Following the same procedure the estimated overall TBR for the LiOH solution is 1.05 excluding test modules. The cost impact of using the $LiNO_3$ solution and Be in ITER is given in Table 2.

THERMAL HYDRAULICS

The outboard blanket is cooled with an aqueous salt solution of 16 g/100 cm^3 $LiNO_3$. The coolant enters the reactor through a header at the bottom of the blanket and flows upwards in a single pass, a distance of 6.2 m, finally exiting the reactor through another header at the top. Coolant flows in three separate zones of the blanket: the first wall, the Be zone and the steel zone. The first wall consists of a double layer of steel with coolant in between. The front layer was taken to be 0.8 cm thick to make allowance for possible material sputtering and the rear layer was taken as 0.2 cm thick.

The Be and steel zones consist of closely packed 3 cm diameter balls. An optimization study[1] has shown that this size balls gives the lowest pressure for the selected operating condition. Coolant at 40°C enters each of the zones and flows at different velocities. The maximum velocity at the first wall is 4.9 m/s. In the Be and the steel zones the average superficial velocity is 15.5 cm/s and 0.3 cm/s respectively. The coolant temperature rise in the first wall zone is 15°C and in the Be and steel zones, it is 35°C. Coolant pressure drop in the first wall and the Be zone is 0.12 MPa

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

	No Be No Li Salt	No Be Li Salt	Be Li Salt
Local TBR	0	0.676	1.506
Overall TBR	0	0.42	1.10
Local M	1.367	1.2	1.366
T_2 purchased in technology phase (kg)	97	60	0.1
Cost of T_2 purchased in technology phase (M$)	970	600	0.1
Be needed (tonnes)	0	0	90
Be cost (M$)	0	0	36
T_2 extraction cost (M$)	0	24	60
Cost of additional heat removal (M$)	10	0	10
^{14}C waste disposal cost (M$)	0	1	1
$LiNO_3$ cost (M$)	0	7	7
Total of relevant cost items (M$)	980	632	115
Cost savings over reactor life (M$)	0	348	865

Table 3. Thermal Hydraulics Parameters for Outboard Blanket

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

[a]Test zones and penetrations excluded.

and much lower in the steel zone. The maximum coolant pressure is 0.27 MPa assuming no over-pressure for suppressing gas release from radiolytic decomposition.

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

The blanket is designed to be capable of operating with a power variation from much lower than the nominal power to 150% of the nominal power at a pressure of 0.42 MPa. Table 3 gives the thermal hydraulics parameters for operating at 100% and 150% of power.

Since the present blanket design has been performed, it has been shown[5] that radiolytic effects will dominate the pressure in the blanket and this changes the ground rules for the design. Instead of minimizing the pressure drop due to fluid flow in the blanket, an optimization has to be performed to minimize overall pressure including that needed to suppress gas evolution due to radiolytic decomposition. Increasing the flow rate will decrease the pressure needed to suppress gas evolution and will result in lower temperatures in the blanket. Preliminary results show that the needed pressure can be kept below 1.5 MPa. The new optimization and redesign for the higher pressure will be performed in the coming year.

KEY TECHNICAL ISSUES

Long term corrosion of the aqueous solution in contact with the blanket and shield materials is a critical issue for this blanket concept.[5] Based on preliminary experimental results,[5] no particular corrosion problems for 316L SS have been revealed for both LiOH and $LiNO_3$ at temperatures < 100°C. No stress corrosion cracking (SCC) was observed at these low temperatures. The electrochemical and long-term corrosion experimental programs underway need to be extended to assess compatibility of the aqueous solution with other blanket and shield materials such as Be. A relatively inexpensive R&D experimental program can provide useful data in the next two years.

The radiolytic decomposition of the aqueous solution is another critical issue for this concept. An experimental program is needed to evaluate radiolysis and the synergistic effects on corrosion and evaluate the possible counter-measures. A conservative calculation performed for NET indicated that an overpressure of ~ 1.3 MPa is required to suppress the release of the gases produced by radiolysis.[5] Using the same calculational method we estimate a pressure of ~ 2.5 MPa is needed for the aqueous blanket design presented here. Design improvements that reduce the required pressure need to be investigated. Increasing the flow rate in regions with high tritium production reduces the gas production and the required pressure to suppress gas release. On the other hand, the impact on safety and thermal performance of operating at a low pressure and allowing some gases to be released and collected at the free surface in a tank for recombination, needs to be investigated. The possibility of using in situ recombiners must also be considered.

The time average tritium inventory in the reactor coolant loop of ~ 400 g is lower than that found in current CANDU reactors. This inventory can be reduced at little or no penalty in TBR by using the aqueous solution only in the front zones of the outboard blanket and inboard shield. Reducing the aqueous solution volume in the reactor by a factor of 8 leads to a reduction in TBR of only 5%. Further reduction in tritium inventory can be achieved by processing the aqueous solution to a tritium concentration < 10 g/ℓ. The cost inpact of this needs to be assessed. The amount of tritium that can be released accidentally can also be reduced by the use of parallel coolant loops which feed into a common tritium extraction system. The relatively large amount of ^{14}C produced in $LiNO_3$ is a concern. Since the radiolysis problem is less pronounced with LiOH and comparable corrosion rates are produced by both salts at low temperatures, LiOH is proposed for use in ITER. As indicated before, tritium self-sufficiency can still be achieved.

CONCLUSIONS

A simple, low cost, flexible, and low risk tritium breeding concept is proposed for ITER. The design is based on utilizing an aqueous Li salt solution coolant flowing through a pebble bed of Be and steel. The design allows for operation at relatively low pressure and low temperature. Using the pebble beds allows for accommodating the complicated ITER geometrical configuration with its numerous penetrations. The design provides the flexibility of changing from the nonbreeding physics phase to the breeding technology phase by simply adding the Li salt to the water coolant with no need for blanket/ shield changeout. Tritium self-sufficiency as well as enhanced magnet protection can be achieved by using the aqueous solution as a coolant in all blanket and shield zones. Critical issues associated with this blanket concept have been identified and it is believed that an inexpensive R&D experimental program can resolve these issues in the next two years.

ACKNOWLEDGEMENT

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