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October 1989

UWFDM-813

Presented at the 13th Symposium on Fusion Engineering, 2–6 October 1989, Knoxville TN; IEEE Cat. No. 89CH2820-9, Vol. 1, 74.

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AN AQUEOUS LITHIUM SALT BLANKET FOR ITER

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Abstract

An aqueous lithium salt blanket is designed as a candidate for the tritium production blanket of ITER. There are two rows of flat tube aqueous salt flow channels placed between beryllium blocks. The inboard and outboard blanket configurations have been optimized to maximize tritium breeding. A local outboard TBR of 1.44 is obtained using LiOH at a concentration of 5 g/100 cm^3. The overall TBR determined from 3-D neutronics calculations is 0.88. The aqueous salt flow parameters have been optimized to minimize the salt pressure and inventory leading to a pressure of 1.15 MPa and tritium inventory of 104 g. There are no crevices in the aqueous salt channels and a redundancy has been provided against a salt to water leak potential. The first wall, blanket, and shield qualify for shallow land burial. The highest temperature in the blanket after a two week period without coolant is only 440°C. The first wall and blanket are capable of withstanding the worst disruption load.

Introduction

The aqueous lithium salt blankets (ALSB) have very attractive features which make them candidates for use in the next generation of fusion devices such as the International Thermonuclear Experimental Reactor (ITER) which has a physics phase followed by a technology phase. The main attractive feature of this blanket is its flexibility where switching to tritium breeding in the technology phase can be achieved by the simple dissolution of lithium salt in the cooling water. This is particularly important if the same plasma configuration can be used in both phases, and no blanket or first wall changeout is required.

There are several other advantages of the ALSB, among the most prominent being its inherent simplicity. In the original ALSB design [1], rectangular box containers were filled with Be and steel balls and the aqueous salt coolant at low temperature and pressure simply flow through the pebble bed. Since then, the problem of radiolysis has emerged indicating that a higher pressure will be needed to keep the radiolysis products in solution. The design underwent several modifications to reduce the aqueous salt pressure, the surface area in contact with the salt, and the overall salt inventory. In this paper, the present ALSB proposed for ITER is presented.

General Design and Thermal Hydraulics

Figure 1 is a midplane cross section of an outboard (OB) blanket module. The OB blanket is divided into 48 modules, three fitting between adjacent TF coil center lines.

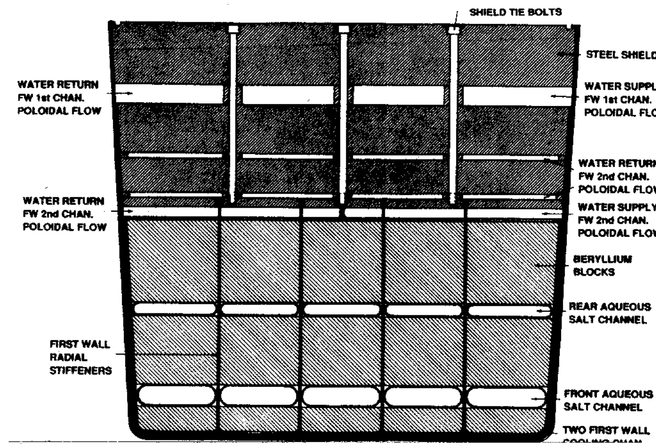

Fig. 1 Outboard side module midplane cross section.

Of the three, the central module has a penetration extending 110 cm toroidally and 340 cm poloidally, and is split into an upper and a lower half. The two side modules are 85 cm wide and extend the full height of the blanket. All the OB blankets are 85 cm deep. The inboard (IB) blanket is divided into 16 modules, 137 cm wide and 71.5 cm deep, and has 2 cm thick graphite tiles attached to the first wall (FW). In all other respects it is designed similarly to the OB blanket. Both IB and OB modules have a FW consisting of three layers of 316 stainless steel, 0.4, 0.2 and 0.4 cm thick separated by two 0.3 cm thick water coolant passages. The FW is vacuum formed and diffusion bonded. It is followed by the breeding zone consisting of Be blocks separated by flattened tubes of aqueous salt running poloidally. The rear of the blanket is occupied by steel shielding blocks with poloidal cooling passages. Blanket module modifications were considered to accommodate the various reactor penetrations. No crevices or sharp corners exist in the aqueous salt channels. The probability of an aqueous salt leak to water is minimized by having multiple barriers between the two systems.

Toroidally, the blanket zone is separated by radial stiffeners which are welded to the FW and to back plates which in turn are bolted to the rear structure of the shield providing support for the FW. There are two primary bending loads on the FW: a static load due to the coolant pressure, and a dynamic load due to disruption forces. These loads were analyzed using the two-dimensional finite element code ANSYS. Since Be is relatively brittle, no credit was taken for its tensile strength, meaning that the coolant pressure was assumed to act directly on the FW. The dynamic loads with peaks of 2.0 MPa and 1.45 MPa for the IB and OB blankets, respectively, are assumed to vary linearly in the toroidal direction. Combining the loads together and using an allowable stress of 165 MPa (1.5 S_{mt} at 100°C), yields space between stiffeners of 15.5 cm for the IB and 17.5 cm for the OB blankets. More recent calculations indicate the OB peak load to be 0.75 MPa, making this design very conservative.

The blanket has two cooling systems, water and aqueous salt. Operating pressure for both systems is determined by that required to keep the products of radiolysis (H_2 and O_2) in solution plus pressure needed to overcome friction. The amount of radiolysis products generated in the water and aqueous salt depends on the rate of energy deposition by the different types of ionizing radiation. The absorbed dose has been calculated for the different water and aqueous salt channels with contributions from the different radiations. These dose rates have been used together with the primary yields (G-values) [2] to determine the amount of gas liberated. The pressure required to keep the radiolysis products in solution was determined as a function of flow velocity for the different channels and used in the thermal hydraulics optimization aimed at minimizing the system pressure. The flow velocity also is important from the electrolysis standpoint since the electric potential across the channel is a function of the magnetic field, the velocity and the radial depth. A maximum velocity of 4.7 m/s occurring in the IB front channel has been selected to insure that the electrolysis potentials are within acceptable limits on the potentiodynamic polarization curve. A careful optimization of blanket parameters makes it possible to operate the aqueous salt at a maximum pressure of 1.15 MPa. There is a potential for reducing the pressure considerably by a small addition of hydrogen which acts as a catalyst for H_2O recombination [3]. We are not taking credit for this in the current design, pending experimental verification. Figure 2 shows the effect of aqueous salt velocity in the different channels on the system pressure. This graph was used to determine the velocity in the different channels required to match the pressure to that in the IB front channel where velocity is dictated by electrolysis.

With the exception of the central modules, all the blanket modules have inlet connections on the bottom and exit on the top. This helps establish natural convection in case of a loss of flow accident (LOFA). Because of the split nature of the central modules, both inlet and outlet connections have to be on the same side. In this case the aqueous salt first travels through the rear channel, then back through the front channel. The remaining modules have the salt flowing in both channels in the same direction, coming in at the bottom and exiting at the top. The salt inlet temperature is 40°C, the highest exit temperature is 60.5°C, and the maximum salt/SS interface temperature is < 95°C. A total of 2400 m^2 of surface area is in contact with the aqueous salt in the reactor. The total aqueous salt inventory in the system is 104 m^3. At a tritium concentration of 10 Ci/kg, the total tritium inventory in the blanket aqueous salt loop is 104 g.

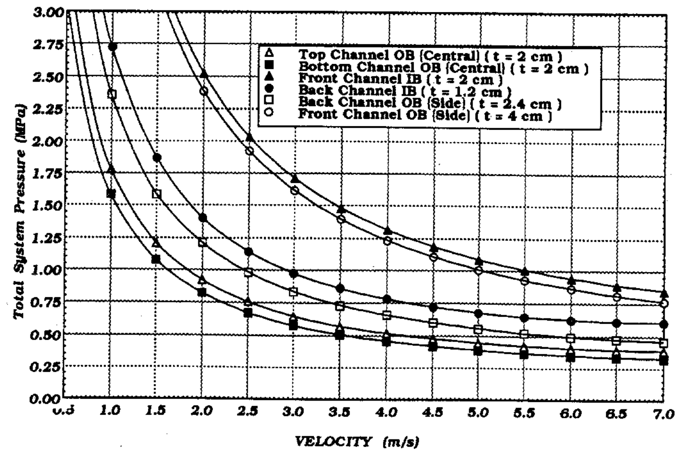

Fig. 2. Effect of velocity on pressure.

Figure 1 shows the water supply and return manifolds for the two first wall channels. The front channel at a velocity of 2.4 m/s absorbs all the surface heat while the rear channel after traversing the first wall at 1.4 m/s flows through the shield poloidally absorbing the nuclear heat. The inlet temperature is 40°C and the average exit temperature is < 60°C. The maximum FW temperature is 280°C on the IB outer graphite surface and 170°C on the OB outer steel surface. As in the salt system, the water pressure was optimized at 0.55 MPa to keep radiolysis products in solution while maintaining reasonable temperatures and mass flow rates. Table 1 gives the pertinent thermal hydraulics parameters.

Neutronics Analysis

Both one-dimensional (1-D) and three-dimensional (3-D) neutronics analyses have been performed for the ALSB. The IB and OB blanket configurations have been optimized

Table 1. Selected Thermal Hydraulics Parameters

	INBOARD	OUTBOARD CENTRAL	OUTBOARD SIDE
Aqueous Salt			
Depth of channel front/rear (cm)	2.0/ 1.2	2.0/ 2.0	4.0/ 2.4
Inlet temp. (°C)	40	40	40
Velocity in channel front/rear (m/s)	4.7/ 2.48	1.58/ 1.58	4.35/ 2.12
Avg. outlet temp. front/rear (°C)	45/ 49	60.5/ 60.5	50.8/ 51.2
System press. (MPa)	1.15	1.15	1.15
T_{max} salt/SS (°C)	60.1	94.4	67.7
$\dot{M}$ in reactor (kg/s)	2672	895	5293
Area in contact (m^2)	818	494	1096
Water Coolant			
Inlet temp. (°C)	40	40	40
Velocity in channel front/rear (m/s)	2.4/ 1.4	2.4/ 1.4	2.4/ 1.4
Avg. outlet temp. front/rear (°C)	59.1/ 66.1	55.7/ 61.4	51.7/ 58.4
System press. (MPa)	0.55	0.55	0.55
$\dot{M}$ in reactor (kg/s)	1718	1255	3536

to maximize tritium breeding. The total first wall and blanket thicknesses in the OB and IB regions are constrained to 45 and 23 cm, respectively. Two cm thick graphite tiles are utilized in the IB region. The calculations have been performed in toroidal cylindrical geometry using ONEDANT and ENDF/B-V data. A large enhancement in the tritium breeding ratio (TBR) is achieved by enriching Li in ^{6}Li. Hence, a Li enrichment of 90% ^{6}Li is used. The TBR was found to increase by ~4% as the LiOH concentration increases from 5 g/100 cm^3 to the solubility limit in cold water (12.8 g/100 cm^3). However, a concentration of 5 g/100 cm^3 is used in this design to minimize concerns about aqueous salt corrosion. The optimum radial builds for the OB and IB blankets are shown Figs. 3 and 4. These configurations result in local TBR values of 1.44 and 1.03 for the OB and IB blankets, respectively. Coupling the results of the 1-D toroidal cylindrical geometry calculations with the coverage fractions of the OB and IB breeding blankets gave an estimate for the overall TBR of 0.9. This excludes 16 horizontal ports in the OB region for heating, current drive and test modules. The peak end-of-life, dpa and helium production values in the OB first wall are 55.3 dpa and 600 appm, respectively.

3-D neutronics calculations have been performed using the MCNP Monte Carlo code with ENDF/B-V data. The neutron source was sampled from the toroidal plasma zone using the appropriate source profile. The IB, side OB, and central OB blanket modules were modeled in detail. Assembly gaps, copper side structure, layered first wall, aqueous salt channels, structural stiffeners, divertor plates, and vacuum pumping ducts were included. Sixteen standard 1.1 m x 3.4 m radial ports were used in the OB region. A typical Li/V blanket was used in the ports to represent a blanket test module. Calculations performed using different materials in the radial ports indicated that the TBR in the permanent breeding blanket is not sensitive to the material used in the ports. Figure 5 gives a vertical cross section through a central OB module in the model used. Table 2 gives the tritium breeding results for the ALSB. If tritium bred in the test modules is accounted for, an overall TBR of 0.96 is achieved. The total thermal power has been determined to be 1180 MW.

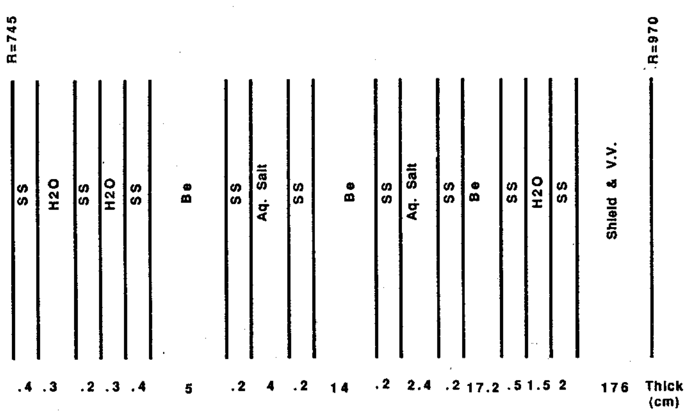

Fig. 3. Radial build of OB ALSB.

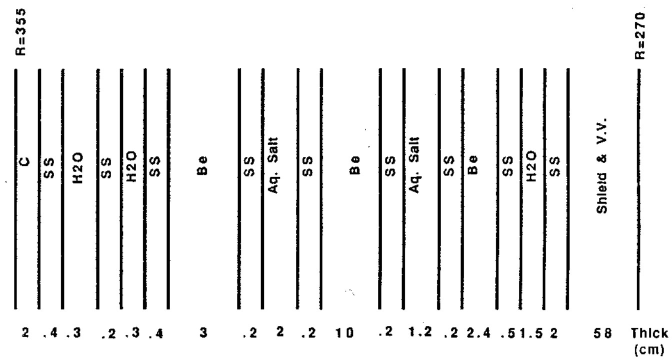

Fig. 4. Radial build of IB ALSB.

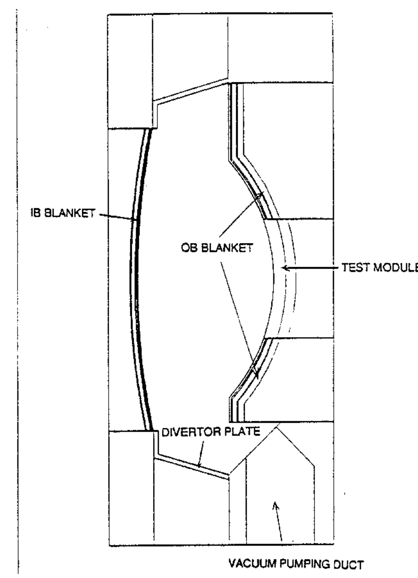

Fig. 5. Cross section in the 3-D model.

Activation and LOCA Analyses

Activation analysis has been performed for the IB and OB ALSB and shield using the DKR-ICF code and ACTL library. A toroidal cylindrical geometry model has been used with 3 FPY operation. Annealed type 316 SS is used for structure. The short term activity (t < 1 d) is dominated by ^{56}Mn, ^{55}Fe, and ^{51}Cr.

Table 2. Tritium Breeding (Tritons/DT Fusion) in the Different Regions of ALSB

Inboard Blanket	
Front Channel	0.153 (±1.4%)
Back Channel	0.059 (±2.1%)
Total	0.212 (±1.2%)
Outboard Side Modules	
Front Channel	0.361 (±0.9)
Back Channel	0.145 (±1.5%)
Total	0.506 (±0.8%)
Outboard Top Central Modules	
Front Channel	0.077 (±2.1%)
Back Channel	0.029 (±3.5%)
Total	0.106 (±1.9%)
Outboard Bottom Central Modules	
Front Channel	0.039 (±2.9%)
Back Channel	0.015 (±4.8%)
Total	0.054 (±2.6%)
Total in IB and OB Modules	0.878 (±0.6%)

In the period between 1 d and 10 y, ^{55}Fe and ^{60}Co are the major contributors. The long term activity is dominated by ^{93m}Nb, ^{63}Ni, ^{93}Mo, and ^{59}Ni. The total activity at shutdown in the FW/B/S is 1623 MCi and drops to 591 MCi in one day and 235 MCi in one year. The total decay heat at shutdown is 12 MW and drops to 1 MW in one day and 0.3 MW in one year. The activity produced in the aqueous salt and water coolant has been calculated assuming that the coolant spends 1/4 of the time inside the reactor. The coolant/breeder activity is very small with the total long term ^{14}C activity being 42 and 39 Ci in the aqueous salt and water coolant, respectively. The waste disposal rating (WDR) for the ALSB and shield has been calculated using the 10CFR61 waste disposal limits. The WDR values for the different regions are given in Fig. 6. ^{94}Nb and ^{63}Ni are the major contributors to the WDR. Both IB and OB blankets qualify as Class C LLW but the FW requires deep geological burial even when combined with the blanket. Disposing of the FW, blanket, and shield together allows for shallow land burial of both IB and OB regions.

The temperature history during a loss of coolant accident (LOCA) in the aqueous salt blanket was analyzed with the two-dimensional finite element code ANSYS. During LOCA, the coolant is assumed to be lost instantly, and the plasma is assumed to remain on for 10 seconds. In the model the toroidal field coils act as uncooled heat sinks. The only heat transfer mechanism considered in the plasma chamber is radiation. Both conduction (through air) and radiation are assumed to be operative in the coolant channels. No heat transfer to the surroundings (outside the TF coils) is considered. Three cases were run: one with conduction only and no stiffeners, one with conduction only including stiffeners, and a third with conduction and radiation. The highest temperatures in the blanket, after a two week period without coolant, for the three cases run are 1627°C, 733°C, and 439°C, respectively. The last result indicates that no damage is expected to any of the structural materials in the blanket.

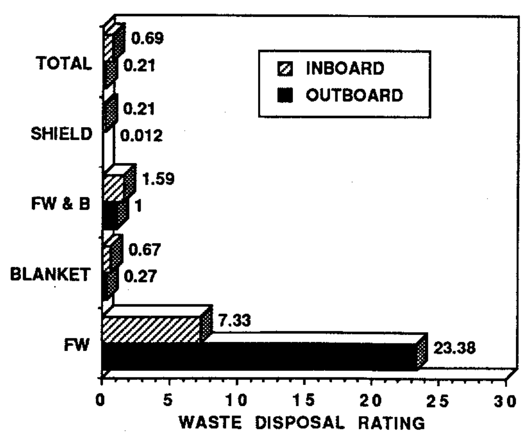

Fig. 6. Class C WDR.

Tritium Recovery

The tritiated water concentration in the ALS is maintained at ~10 Ci/kg Two options were considered [4]. In one option, all of the ALS flow is directed to a flash evaporator operating at one atmosphere pressure; consequently, the dissolved gases are released and ~1 kg/s of H_2O/HTO is vaporized and sent to the Tritium Recovery System (TRS). In the second option, a liquid phase catalytic recombiner is used for the ALS. This recombiner forms water from the dissolved gases and < 1% of the ALS flow is sent to the flash evaporator. Water distillation columns are used to concentrate the tritiated water from 10 Ci/kg to 1000 Ci/kg before sending it to the TRS where 56% of the tritium is removed. This water distillation is based upon detailed analyses and operating experiences for such columns having a flow rate of 1700 kg/hr. Two columns are required to remove tritiated water at 40% of the breeding rate. These columns will continue to remove tritiated water during reactor shutdown. During extended continuous operation periods the tritium concentration can build up to a maximum of 26 Ci/kg. Five columns are needed if tritium is to be removed at the same rate as the breeding rate. In this case the tritium concentration will be maintained at 10 Ci/kg even during extended periods of operation.

Summary

An aqueous lithium salt blanket is designed as a candidate for the ITER. The blanket utilizes LiOH at a concentration of 5 g/100 cm^3. The aqueous salt pressure is 1.15 MPa and the tritium inventory is 104 g. There are no crevices in the aqueous salt containers. The overall TBR achieved is in the range 0.88 to 0.96 depending on whether tritium bred in test modules is taken into account. The first wall and blanket are capable of withstanding the worst disruption load.

Acknowledgments

Funding has been provided by the U.S. Department of Energy.

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