ASSESSMENT OF FIRST WALL AND BLANKET OPTIONS WITH THE USE OF LIQUID BREEDER

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As candidate blanket concepts for a U.S. advanced reactor power plant design, with consideration of the time frame for ITER development, we assessed first wall and blanket design concepts based on the use of reduced activation ferritic steel as structural material and liquid breeder as the coolant and tritium breeder. The liquid breeder choice includes the conventional molten salt Li_2BeF_4 and the low melting point molten salts such as $LiBeF_3$ and $LiNaBeF_4$ (FLiNaBe). Both self-cooled and dual coolant molten salt options were evaluated. We have also included the dual coolant lead-eutectic Pb-17Li design in our assessment. We take advantage of the molten salt low electrical and thermal conductivity to minimize impacts from the MHD effect and the heat losses from the breeder to the actively cooled steel structure. For the Pb-17Li breeder we employ flow channel inserts of SiC_f/SiC composite with low electrical and thermal conductivity to perform respective insulation functions. We performed preliminary assessments of these design options in the areas of neutronics, thermal-hydraulics, safety, and power conversion system. Status of the R&D items of selected high performance blanket concepts is reported. Results from this study will form the technical basis for the formulation of the U.S. ITER test module program and corresponding test plan.

I. INTRODUCTION

As a member of the ITER Test Blanket Module Working Group (TBWG), the U.S. blanket design team has proposed to focus on the self-cooled (SC) and dual-coolant (DC) liquid breeder blanket concepts [1]. Blanket design options being evaluated by different countries in the world are listed below. Bold letters indicate coordinating countries.

- He-cooled Ceramic/Be (China, **EU**, Japan, Korea, RF, U.S.)
- He-cooled PbLi (China, **EU**, Japan, Korea, RF, U.S.)
- Water-cooled Ceramic/Be (China, **Japan**)
- Self-cooled Li with V-alloy (China, Japan, Korea, **RF**, U.S.)
- Self-cooled and DC Molten Salt, Pb-17Li (Japan, China, **U.S.**)

As can be noted, different blanket design options are being covered, including solid and liquid breeders and helium and water-cooled options.

This paper presents an assessment of liquid breeder blanket options for an advanced reactor design. These options include self-cooled (SC) and dual coolant (DC) molten salt (MS) breeder designs and the DC Pb-17Li breeder design. A condition in our assessment is that the blanket concepts can be developed in the time frame of ITER, which also means the use of reduced activation ferritic steel (RAFS) as the structural material. In Section II we outline the design parameters and materials used for this assessment. In Section III we illustrate the MS designs including experience from fission reactor development, design options with low and high melting point MS, designs with the use of Be and Pb neutron multiplier, and summary results from neutronics and thermal hydraulics assessments. In Section IV we introduce the DC Pb-17Li breeder design and in Sections V and VI we present power conversion and safety assessments, respectively. In Section VII we show the research and development status of the identified critical issues associated with selected designs. Our conclusions are presented in Section VIII.

II. DESIGN PARAMETERS AND MATERIALS

For consistency in comparison, we selected a set of advanced reactor design parameters with maximum neutron wall loading of 5.4 MW/m^2 and surface loading of 1 MW/m^2 located at the outboard mid-plane. The structural material used for this assessment is RAFS (F82H), which has a maximum allowable temperature of 550ºC [2].

The selected molten salts for our assessment are the high melting point, lower viscosity MS Li_2BeF_4 (M.P. 459ºC), the lower melting point, higher viscosity MS $LiBeF_3$ (M.P. 380ºC) and the MS FLiNaBe [3], which has a measured melting point of 305 to 320ºC [4]. For our assessment, we have also compared the use of Be and Pb as the neutron multiplier for the MS design options. For comparison, we have also included the DC Pb-17Li design, which has been extensively studied in Europe [5]. For the DC options, helium is used to cool the first wall and structure of the blanket module.

III. MOLTEN SALT DESIGNS

In general, because of much lower electrical conductivity among liquid breeders, the MS options will have a much lower MHD effect than Pb-17Li and Li breeder options. A separate MHD insulator in the coolant channel will not be needed. From the safety point of view, molten salts are inert to air and water, which makes them much safer to work with. These and other benefits have been pointed out in earlier studies [6,7]. Table 1 lists the key advantages and disadvantages for the use of MS. For the DC design, we are also making use of the low thermal conductivity property of MS, using it to act as a thermal insulator to maintain the bulk molten salt temperature at ~100ºC higher than the temperature of the structural material while keeping all of the metallic structure within its temperature limit by cooling it with the helium coolant.

III.A. MSRE Experience

The use of MS for nuclear systems is not new. Its application was demonstrated by the Molten Salt Reactor Experiment (MSRE), which was an 8 MW(th) fission reactor [8]. It can be considered as a very successful MS development experience. The reactor operated for ~5 years from January 1965 to December 1969, and prepared the groundwork for a two region MS breeder fission reactor. From the MSRE program, we have learned about the nuclear application of MS and the importance and possibility of using Be for the reduction and oxidation (REDOX) control of the fluid chemistry. Some of the operational experience from MSRE will be directly applicable to the development of the fusion blanket design, such as:

- The MSRE initial operation experience and hardware development will be directly relevant to the ITER MS test module program.
- Tritium control was a concern for a MS fission reactor, and fusion will have a much higher production rate of tritium, but some of the MSRE experience will be applicable.

Table 1
Advantages and Disadvantages with the Use of MS as Circulating Liquid Breeder

Advantages:	Disadvantages:
• Low pressure operation • Very low tritium solubility • Low MHD interaction • Relatively inert with air and water • Pure material is compatible with many structural materials	• High melting temperature • High viscosity, low thermal conductivity • Needs Be or Pb neutron multiplier • REDOX chemistry control will be needed • High tritium partial pressure is a tritium control concern • Limited heat transfer capability

- All parts of the MSRE salt-containing system are thermally insulated and heated electrically above the liquid temperature of 449ºC to 454ºC.
- A high purity helium (below 1 ppm impurity content) cover-gas system protects the oxygen-sensitive uranium bearing fuel from contact with air or moisture.

Details on the experience with MSRE are available in many papers in the literature and ORNL documents [9,10].

III.B. Molten Salt Design Configurations

In the following we present blanket configurations of the SC and DC designs.

The design configuration of the SC FLiNaBe Coolant and Breeder, Be pebble bed multiplier blanket is shown in Fig. 1. We use a recirculating scheme to achieve a much

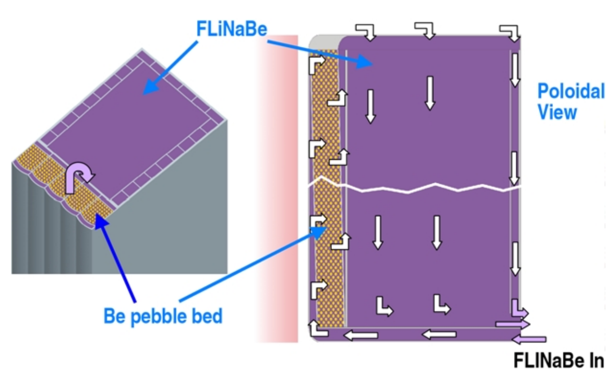

Fig. 1. Schematics of the self-cooled FLiNaBe design.

higher MS mass flow rates to the first wall then through the central channel of the module. This allows us to adjust the heat removal capability of the first wall and the temperature of the MS in the central channel [11].

As shown in Fig. 1, the coolant FLiNaBe enters the FW and blanket from the bottom of the poloidal module. It flows poloidally upward while passing radially through the Be-packed bed neutron multiplier before it turns around at the top of the poloidal module. The FW stream then splits into two streams before flowing poloidally downward. One stream flows in the side channels (two side walls and one back wall) around the module and the other stream flows in the larger central channel. The stream from the central channel is then directed to the heat exchanger outside of the reactor for power extraction and returns at much lower temperature. The return stream from the heat exchanger and the stream from the side channels are then merged in a mixer before the combined stream is pumped through the first wall again. In the central channel, we rely on the poor thermal conductivity of the MS and the poor heat transfer, which is partly due to turbulence suppression, to maintain the high average temperature of the MS for power conversion without large heat losses to the helium cooled steel structure. The selected 8 m high poloidal module has a width of 30 cm at mid-plane as shown in the 2-D figure. Behind the 3 mm thick first wall of the module we have the 10 mm FLiNaBe channels in front and back of the 70 mm thick neutron multiplier Be pebble bed zone. The three walls of the module are coolant and structural channels, which also provide for the poloidal flow of the breeder. Most of the module channel separation walls are 3 mm thick.

The design configuration of the DC, He and MS Li_2BeF_4, Be pebble bed multiplier design is shown in Fig. 2. The selected 8 m high poloidal module also has a width of 30 cm at mid-plane. Behind the 3 mm thick first wall of the module we have the toroidally oriented helium cooling channels followed by MS channels in front and back of the 5 cm thick Be pebble bed multiplier zone. We then have the helium cooled poloidal channels forming the large channels for the MS to flow in the poloidal direction. This configuration is also illustrated in Fig. 3. In this figure, the DC MS Pb multiplier design is also illustrated. As shown in Fig. 3, multiplier Pb will be contained in poloidal channels. The MS does not flow through the Pb multiplier channels; instead, they are cooled by the helium in the surrounding first wall and structural channels.

When comparing the two neutron multipliers we can make the following observations on why we are interested in the use of Pb:

Neutron irradiation of beryllium can lead to considerable swelling (>10%) and a large tritium concentration (~50000 appm). The lifetime of blanket may be limited by neutron damage in the beryllium and the allowable tritium inventory, which can be as high as 1.8 kg at the end of life if occasional bakeout is not performed [12].

Beryllium from exposed blankets has to be detritiated before shipping the blankets to a reprocessing plant.

To maintain integrity of the multiplier in spite of large swelling, beryllium pebble beds have to be used instead of plates. Fabrication of large quantities of beryllium pebbles (~30 tons for one power plant) is rather expensive when compared to the use of Pb.

Thermal conductivity of pebble beds is relatively low, resulting in rather high beryllium temperatures. This is also why pebble geometry is selected with a relatively small diameter.

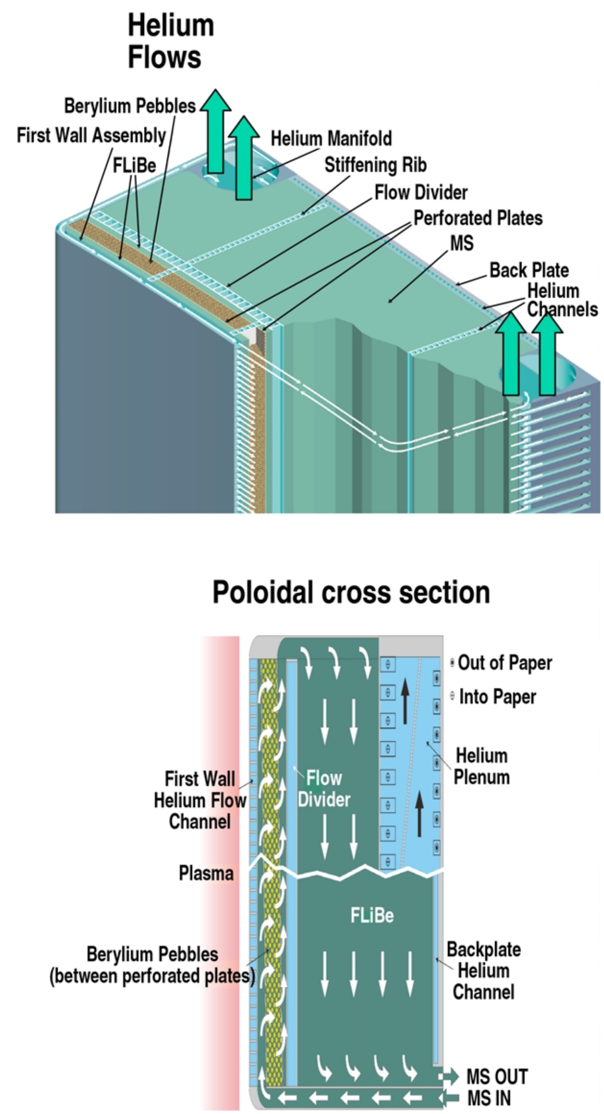

Fig. 2. Design configuration of the DC, He and MS, Be neutron multiplier design

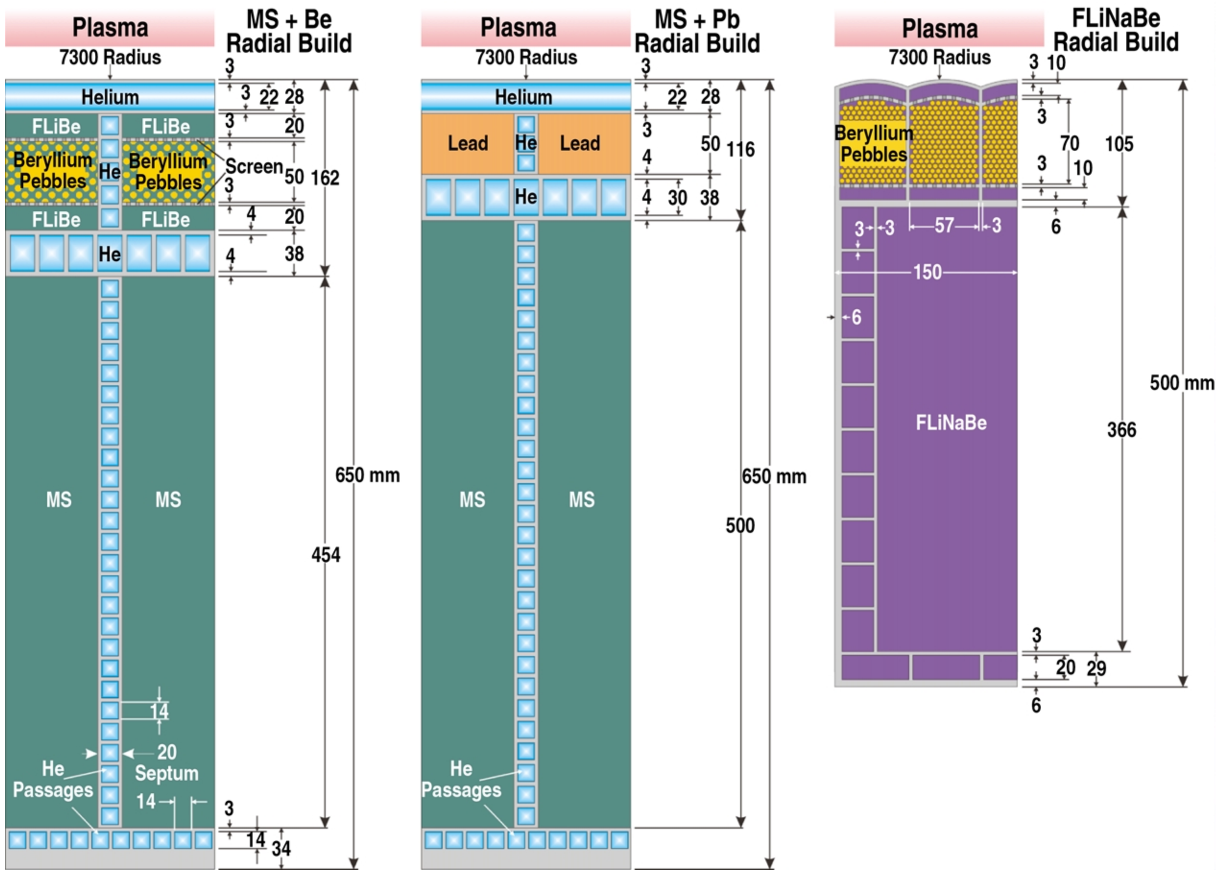

Fig. 3. Radial build dimensions for the MS Be-multiplier, and Pb-multiplier and the self-cooled FLiNaBe designs.

Weight of replacement units is higher since beryllium cannot be drained before replacement.

On the other hand, when Pb is used, because of its much higher mass density, it will have much higher static loading than Be. At the same time during operation, Pb will have to be continuously circulated for the extraction of Bi in order to control the concentration of ^{210}Po.

As shown in Fig. 3, compared with the use of helium for the DC options, the self-cooled FLiNaBe design can have a 150 mm smaller radial build, since there are no helium channels.

III.C. Neutronics

For neutronics analysis, four blanket concepts were analysed. They are the SC FLiNaBe, DC Li_2BeF_4/Be, DC Li_2BeF_4/Pb, and the lower melting point MS DC $LiBeF_3$/Be designs. The FS alloy F82 H is the structural material, and the material volume fractions in the radial zones are specified based on the selected design configuration. The average neutron wall loading is 3.84 MW/m^2 and the peak neutron wall loadings are 5.45 MW/m^2 outboard and 3.61 MW/m^2 inboard [12]. The radial build of the design options were adjusted to produce adequate tritium breeding as shown in Table 2. The shield thickness was selected to provide adequate protection for the superconducting magnetic coils and allow rewelding of the vacuum vessel. The shield located behind the blanket was found to be lifetime component. In addition, heat deposition in the blanket components was used for heat transfer and structural analysis. Decay heat and radioactivity results were provided for safety and waste disposal assessments. In the following we summarize the tritium breeding results. Detailed neutronics results are presented in a companion paper [12].

Table 2
Inboard and Outboard Tritium Breeding Ratio for MS Blankets

	Inboard	Outboard	Total
SC-FLiNaBe	0.432	0.867	1.299
DC-Li_2BeF_4/Be	0.406	0.882	1.288
DC-Li_2BeF_4/Pb	0.399	0.897	1.296

The tritium breeding results for SC FLiNaBe, DC-Li_2BeF_4/Be and DC-Li_2BeF_4/Pb are given in Table 2. These cases are for ^{6}Li enrichment of 40%. If neutron coverage for the divertor is 10% the overall TBR can be designed to ~1.17 excluding breeding in divertor region. Breeding in divertor zone could add ~0.05 to 0.06. In general the blanket design concepts have the potential for achieving tritium self-sufficiency. Some design parameters can be adjusted (e.g., multiplier thickness, blanket thickness, etc.) to insure tritium self-sufficiency based on calculations with detailed multi-dimensional modeling.

Blanket energy multiplication values of 1.27, 1.21 and 1.13 are obtained for the SC FLiNaBe, DC-Li_2BeF_4/Be and DC- Li_2BeF_4/Pb, respectively. A smaller amount of Be is required in dual coolant design with Li_2BeF_4 than the self-cooled blanket with FLiNaBe, resulting in ~40% less tritium production in the Be multiplier. With total blanket radial build (including shield and vacuum vessel) of 1.05 m IB and 1.2 m OB it is possible for the shield to be a lifetime component, the vacuum vessel can be reweldable and the magnets can be adequately shielded.

III.D. Thermal-Hydraulics

Scoping thermal-hydraulics calculations were performed for the SC FLiNaBe, DC Li_2BeF_4/Pb and DC Li_2BeF_4/Pb design options. Results show that key performance parameters on maximum structural material temperature and overall blanket pressure drop for the DC Li_2BeF_4/Pb and DC Li_2BeF_4/Be design options are similar. Therefore, in the following we will only compare the SC FLiNaBe, and DC Li_2BeF_4/Pb results.

Design assumptions and conditions are:

- Minimum MS T ~40ºC above the melting point
- Maximum temperature of the FS structure < 550ºC
- All interface temperatures < 550ºC;
- For DC design He ΔT=150ºC;
- FLiNaBe ΔT=650–360=290ºC
- Li_2BeF_4 ΔT=700–500=200ºC.

Typical design parameters for the DC designs are:

- He velocity in the FW channels = 75 m/s;
- He velocity in the secondary wall channels = 104 m/s;
- Li_2BeF_4 velocity in central channel = 0.108 m/s;

For the SC FLiNaBe design, we used the recirculating blanket flow scheme [11] to optimize the design. The coolant recirculating fraction was varied to optimize the blanket performance. With an inlet temperature of 360ºC and an outlet temperature of 650ºC, due to the poor heat transfer characteristics of MS, we were not able to meet the RAFS temperature limit of 550ºC. The maximum first wall temperature is at 680ºC. In order to have a credible design, one option is to plate the first wall with a 2 mm thick layer of ODS-steel, which has the same strength as the RAFS but at a ~100 K higher temperature allowable. This plating approach has been demonstrated by a HIP process with excellent strength in EU studies [13].

For the DC Li_2BeF_4 Pb design a counter flow multi-pass first wall routing was used, and roughening of one side of the helium coolant channel was used at the second and third passes to enhance the heat transfer. With minor changes in the design a maximum wall temperature of ~550ºC can be maintained. The corresponding FW helium pressure drop is 0.28 MPa. The maximum Pb-multiplier temperature is 850ºC. The maximum interface temperature between Pb and FS is 555ºC. However, due to the high melting temperature of Li_2BeF_4 at 459ºC, and the relatively cool inlet temperature of 400ºC, we found a 1–2 mm thick frozen layer of Li_2BeF_4 can be formed at the secondary wall of the blanket. Since it is not advisable to have a thin frozen layer of MS in an advanced power reactor design due to potential enhancement of erosion, it is recommended that a lower melting point MS like $LiBeF_3$ or FLiNaBe should be considered for the DC design.

IV. DC Pb-17Li DESIGN

The other well known DC concept is the use of Pb-17Li as the liquid breeder. It was first utilized in the ARIES-ST design [14]. It was then extensively studied by FZK as their Advanced Dual Coolant DEMO Blanket [5]. It is very similar to the DC-MS designs, but with the distinction that additional neutron multiplier will not be needed. The FZK design was evaluated to a maximum neutron wall loading of 3 MW/m^2 and an average neutron wall loading of 2.27 MW/m^2; the maximum surface loading was at 0.59 MW/m^2.

Features of the DC Pb-17Li design are:

Inboard/Outboard poloidally segmented blanket modules, each with two-pass poloidal Pb-17Li flow are as shown in Fig. 4. The helium inlet and outlet temperatures are 300ºC and 480ºC, respectively, and the Pb-17Li inlet and outlet temperatures are 460°C and 700°C, respectively. 3-D Monte Carlo calculation with the MCNP code was used in the neutronics calculation. The global tritium breeding ratio is 1.15 with Li^6 enrichment at 90%. A key element in the DC Pb-17Li design is the use of SiC_f/SiC composite flow channel insert (FCI) [14]. This element performs the key functions of reducing the MHD effect of circulating Pb-17Li and thermally isolating the high temperature Pb-17Li and the low temperature FS structure, which is actively cooled by helium.

V. POWER CONVERSION

For the liquid breeder designs of interest, both MS and Pb-17Li options have high liquid breeder outlet temperatures. For power conversion, we selected the dual Brayton cycle + Molten Coolant Gas Cycle (MCGC)

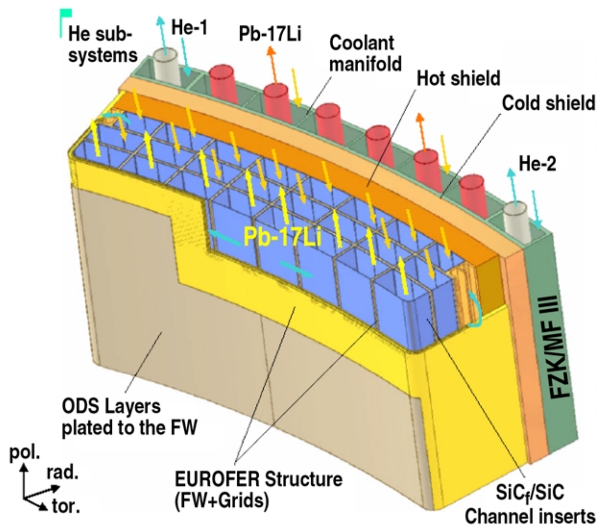

Fig. 4. DC Pb-17Li blanket for the EC advanced DC blanket design.

[15,16] option. This is used to optimize the thermal efficiency. Since both are closed cycle systems, the design would also help to reduce the possible permeation of tritium to the environment by eliminating the need of a steam generator [11]. For the DC MS design with 40% of the power transferred to the helium with inlet and outlet temperatures of 300ºC to 450ºC, respectively, and 60% power transferred to the salt with inlet and outlet temperatures of 500ºC to 700ºC, respectively, the dual cycle option can obtain a thermal efficiency of 40% (28% for helium direct cycle with one turbine and two compressors and 48% for MCGC with three turbines and six compressors). This combined efficiency could be further optimized by increasing the power fraction deposited into the liquid breeder zone with the use of a more effective thermal insulation layer in the liquid breeder channel.

We looked into the design of the MS-to-helium heat exchanger. In the heat exchanger channel, with the high viscosity of MS, the fluid is in laminar flow. The traditional tube and shell heat exchanger design will not be applicable due to low heat transfer coefficient and unacceptably high pressure drop. Therefore a compact plate fin heat exchanger will be used. The heat transfer in a compact plate fin heat exchanger is dominated by the heat conduction through the boundary layer. Because the flow channel hydraulic diameter is an the order of 1 mm, the heat transfer coefficient can be estimated as thermal conductivity divided by the thickness of boundary layer: (1 W/m-K)/(0.5 mm) = 2000 W/m^2-K for MS. This is still a good heat transfer coefficient. The pumping power requirement for a compact plate fin heat exchanger on the salt side is very low compared to the pumping power for helium because the volumetric flow rate for MS is less than 1% of the helium flow rate.

VI. SAFETY

Safety comparisons were made for the SC-FLiNaBe, DC-Li_2BeF_4/Be, DC- Li_2BeF_4/Pb and DC-Pb-17Li blanket options. On design specific safety issues, no clear safety advantage was identified. For the SC FLiNaBe design, one would have to deal with the decay heat from Na-24. This might require the draining of the MS or the use of in-vessel natural convection loops to be designed into the FW/blanket system. For the DC designs, helium pressurization could cause failure of radioactive material confinement barriers (module box vacuum vessel, cryostat and confinement building).

On common safety issues, the following will have to be addressed:

- Tritium in Be multiplier or Po-210 and Hg-203 in Pb multiplier or in the Pb-17Li breeder
- Decay heat from FS, F82H structural material
- Radioactive inventories: tritium in F82H and MS radioactive isotopes
- Tritium permeation: pipe walls and CCGT system
- Molten salt freezing due to electrical pipe heating system failure leading to loss-of-flow accidents
- When Be neutron multiplier is used, the multiplier must be baked (~700-750°C) to reduce tritium inventory from ~2 to ~0.5 kg. This temperature range may be a problem for the F82H structural material.

In general at least two possible design solutions are available to limit helium pressurization during in-vessel LOCA: a large volume to vent helium into, and to limit total helium inventory. Application of these approaches will be design specific.

From waste disposal considerations, and to achieve the Class-C limit for these FW/blanket design options, the following will need to be considered. The induced radioactivity of F82H alloy could be dominated by Tc-99, produced from Mo. Therefore it is important to reduce the Mo content from 0.02% to <0.01%. For MS designs the waste disposal rating is low at «1, with 10CFR61 limits, and the major long lived radioactive isotope is C-14, which is from neutron reaction with F. However, a detailed safety assessment will need to be performed for specific designs in the future.

VII. RESEARCH AND DEVELOPMENT

As presented above, we have identified high performance liquid breeder design options that have the potential of adequate tritium breeding and high thermal performance while using RAFS as the structural material. These designs will be suitable as FW/blanket candidates to be tested in ITER. However, among these design options we have found common and specific R&D issues that will have to be addressed.

VII.A. Materials

If the low melting point MS FLiNaBe is to be considered, its melting point needs to be more accurately measured. Sandia National Laboratory has taken the task of performing the measurement [4]. This is not a simple task since FLiNaBe is actually a mixture of three different compounds, namely: BeF_2, NaF and LiF. A suitable combination of these compounds will be needed to find the lowest melting temperature. At the same time the stability of the melting point of this mixture will have to be understood. Preliminary results show that FLiNaBe can have a melting temperature of 305ºC to 320ºC [4]. Further study of the melting point and thermal properties such as viscosity and thermal conductivity will be needed.

For the DC designs, one of the design goals is to bring the selected liquid breeder to high temperature in the range of 650ºC to 700ºC for high thermal performance. However, considering the external piping and heat exchanger systems, when Pb-17Li or MS is used, suitable high strength and temperature alloys will be needed. They will also need to be compatible with the selected liquid breeder. An alternative solution is to employ concentric tubes for the piping with the tube wall cooled by the "cold" inlet fluid to a temperature <550°C.

VII.B. Tritium Control

As presented in Sections III and VI, both the MS and Pb-17Li liquid breeder have low solubility of tritium; relatively MS is lower. Therefore, corresponding designs can have relatively low tritium inventory, but at the same time, the control of tritium permeation to the environment will be a concern. Tritium migration and permeation control will need to be studied. At the same time the extraction of tritium from these breeders will have to be developed. Much is known for the Pb-17Li system [13]. An R&D program will need to be established for the MS system.

VII.C. REDOX Control

As presented in Section III, when MS is used in the fusion environment with the presence of tritium, TF will be formed, which is very corrosive to the metallic structural material in the coolant loop. REDOX has been proposed as a means of chemistry control, with the use of Be as a proposed REDOX agent, which was demonstrated in the fission MSRE program. Similar demonstration will be needed for the fusion application. The basic reaction is described by $2F+Be=BeF_2$. This reaction goes to the right side, but the reaction kinetics has to be demonstrated. This is now being studied under the U.S./Japan JUPITER-II collaboration program [17].

VII.D. Flow Channel Insert

For the DC Pb-17Li concept, as mentioned above, the critical component for the control of the MHD effect and high thermal performance is the SiC_f/SiC composite FCI. For the development of this critical component, the following performance requirements have been identified [18]:

1. Electrical and thermal conductivity of the SiC_f/SiC FCI should be as low as possible, specially in the direction perpendicular to the wall to avoid velocity profiles with side-layer jets and excess heat transfer to the He-cooled structure.
2. The inserts have to be compatible with Pb-17Li at temperatures up to 700-800ºC
3. Liquid metal must not "soak" into pores of the SiC_f/SiC composite insert in order to avoid increased electrical conductivity and high tritium retention. In general "sealing layers" are required on all surfaces of the inserts. It should be noted that even if the change in conductivity is modest from a pressure drop point of view it could also affect flow balance.
4. The primary stresses in the inserts are small, since they only need to carry their own weight. However, secondary stresses caused by temperature gradients must not endanger the integrity under high neutron fluence.
5. The inserts must be pratical and affordable.

Development of SiC_f/SiC FCI has just been initiated and we will have to rely on the industrial experience from the SiC_f/SiC development industry to tailor the composite material in providing low electrical and thermal conductivities for our application.

VIII. CONCLUSIONS

As candidate blankets for a U.S. Advanced Reactor Power Plant design, and in consideration of the time frame for the ITER development, we assessed first wall and blanket design concepts based on the use of reduced activation ferritic steel as structural material and liquid breeder as the coolant and tritium breeder. The liquid breeder choice includes the conventional molten salt Li_2BeF_4 and the low melting point molten salts such as $LiBeF_3$ and FLiNaBe. Both self-cooled and dual coolant molten salt options were evaluated. We have also included the dual coolant lead-eutectic Pb-17Li design in our assessment. We based our first wall and blanket assessment on an Advanced Reactor Power Plant design, which has a maximum neutron wall loading of 5.4 MW/m^2 and a maximum surface heat flux of 1 MW/m^2 at the outboard mid-plane of the tokamak reactor. Molten salt blankets will require an additional neutron multiplier like Be or Pb to provide adequate tritium breeding. For the dual coolant design, helium is used to remove the first wall surface heat flux and to cool the entire steel structure. The liquid breeder is circulated to external heat exchangers to extract the heat from the breeding zone (a "self-cooled" breeding zone). We take advantage of the molten salt low electrical and thermal conductivity to minimize impacts from the MHD effect and the heat losses from the breeder to the actively cooled steel structure. For the Pb-17Li breeder we employ flow channel inserts with a low electrical and thermal conductivity to perform respective insulation functions. For the DC MS blanket options, to avoid the formation of a thin solid layer of high melting point MS in the blanket, the utilization of a lower melting point MS is recommended. For the lower melting point MS FLiNaBe, physical properties like melting point will need to be further established. For the MS designs, REDOX control will need to be demonstrated to mitigate the compatibility issue between the generated TF and structural material. Due to the low tritium solubility of MS and Pb-17Li, tritium permeation control will be needed to minimize contamination of the environment. For the DC Pb-17Li design, successful development of the FCI, such as SiC/SiC composite, is required to arrive at an acceptable MHD pressure drop and to thermally insulate the high temperature self-cooled breeder from the RAFS structure. Results of the above R&D items will then form the technical basis for the selection of the reference blanket concept for the U.S. and provide input for the formulation of the U.S. ITER test module program and its corresponding test plan.

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REFERENCES

[1] M. ABDOU et al., "Blanket Testing Issues and Requirements," 16th ANS Top. Mtg. on the Technology of Fusion Energy, Madison, Wisconsin, September 14–16, 2004 to be printed in *Fusion Sci. and Technol*.

[2] S. J. ZINKLE, N. M. GHONIEM, "Operating Temperature Windows for Fusion Reactor Structural Materials," *Fusion Engineer. and Design* **51-52**, 55 (2000).

[3] R. E. NYGREN et al., "A Fusion Chamber Design with a Liquid First Wall and Divertor," Proc. 20th IEEE/NPSS Symp. on Fusion Engineering, San Diego, California, 2003, to be published in *Fusion Sci. and Technol*.

[4] J. McDONALD et al., "Measurement of the Melting Point Temperature of Several Lithium-Sodium-Beryllium Fluoride Salt (Flinabe) Mixtures," 16th ANS Top. Mtg. on the Technology of Fusion Energy, Madison, Wisconsin, September 14–16, 2004 to be printed in *Fusion Sci. and Technol*.

[5] P. NORAJITRA et al., "Conceptual Design of the Dual-Coolant Blanket within the Framework of the EU Power Plant Conceptual Study (TW2-TRP-PPCS-12)," Final report FZKA 6780, May, 2003.

[6] M. A. ABDOU et al., "On the Exploration of Innovative Concepts for Fusion Chamber Technology," *Fusion Engineer. and Design* **54**, 181 (2001).

[7] SAGARA et al., "Studies on Flibe Blanket Designs in Helical Reactor FFHR," *Fusion Technol.* **39**, 753 (2001).

[8] P. HAUBENREICH, J. R. ENGEL, "Experience with the Molten-Salt Reactor Experiment," *Nuclear Applications and Technol.* **8** (1970).

[9] R. H. GUYMON, "MSRE Systems and Components Performance," ORNL-TM-3039, June 1973.

[10] "MSRE Design and Operations Report, Part I, Description of Reactor Design," ORNL-TM-728, January 1965.

[11] C. WONG et al., "Molten Salt Self-Cooled Solid First Wall and Blanket Design Based on Advanced Ferritic Steel," submitted for publication in *Fusion Engineer. and Design* (2004).

[12] M. E. SAWAN, "Neutronics Assessment of Molten Salt Breeding Blanket Design Options," 16th ANS Top. Mtg. on the Technology of Fusion Energy, Madison, Wisconsin, September 14–16, 2004 to be printed in *Fusion Sci. and Technol*.

[13] P. NORAJITRA et al., "The EU Advanced Dual Coolant Blanket Concept," *Fusion Engineer. and Design* **61-62**, 449 (2002).

[14] F. NAJMABADI et al., "Spherical Torus Concept as Power Plants-the ARIES-ST Study," *Fusion Engineer. and Design* **65**, 143 (2003).

[15] P. F. PETERSON, "Multiple-Reheat Brayton Cycles for Nuclear Power Conversion with Molten Coolants," *Nuclear Technol*. **144**, 279 (2003).

[16] H. ZHAO et al., "Optimized Helium-Brayton Power Conversion for Fusion Energy Systems," 16th ANS Top. Mtg. on the Technology of Fusion Energy, Madison, Wisconsin, September 14–16, 2004 to be printed in *Fusion Sci. and Technol*.

[17] D. A. PETTI et al., "JUPITER-II FLiBe Tritium/Chemistry and Safety Experimental Program," *Fusion Sci. and Technol.* **41**, 807 (2002).

[18] N. B. MORLEY, S. MALANG, I. KIRILLOV, "Thermofluid Magnetohydrodynamic Issues for Liquid Breeders," 16th ANS Top. Mtg. on the Technology of Fusion Energy, Madison, Wisconsin, September 14–16, 2004 to be printed in *Fusion Sci. and Technol*.