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A MOLTEN SALT COOLING/17Li-83Pb BREEDING BLANKET CONCEPT*

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ABSTRACT

A description of a fusion breeding blanket concept using draw salt coolant and static 17Li-83Pb is presented. 17Li-83Pb has high breeding capability and low tritium solubility. Draw salt operates at low pressure and is inert to water. Corrosion, MHD, and tritium containment problems associated with the MARS design are alleviated because of the use of a static LiPb blanket. Blanket tritium recovery is by permeation toward the plasma. A direct contact steam generator is proposed to eliminate some generic problems associaced with a tube shell steam generator.

I. INTRODUCTION

The tube bank blanket described in WITAMIR-I[1] and MARS[2] is an attractive design configuration. It is simple and easy to fabricate. By using 17Li-83Pb as the coolant and breeding material, the potentially severe chemical reaction between the liquid metal and water, which may occur if lithium is used, can be avoided. The low tritium solubility in 17Li-83Pb results in a very low blanket tritium inventory. In addition, 17Li-83Pb has excellent neutronic properties so that the blanket provides sufficient tritium breeding with an excellent energy multiplication. The heat transfer is adequate so that an efficient power conversion system can be designed.

Many potential problems exist. Most of the problems have been identified in MARS study and their severity further clarified in the BCSS study.[3] The most critical problems are corrosion product transport and tritium containment. The severity of the problems may effect the validity of the blanket concept. The other potential problems are MHD, the cost of 17Li-83Pb (90% enriched) and of the heat transport system, and the design and cost of the double-walled steam generator. Those problems may effect the attractiveness of the blanket. The logical step is to modify the MARS blanket to retain its advantages and eliminate the potential problems.

Since the problems of the MARS blanket are mostly associated with the circulating of LiPb, the logical step is to change to a static LiPb system, while maintaining the simple configuration of the MARS blanket. As part of the Mini-MARS study, a molten salt cooling/ 17Li-83Pb breeding blanket was developed and is described in this paper. A number of new concepts are incorporated in the design, such as in-situ blanket tritium recovery and the use of a direct contact steam generator and regeneration pump for pressure isolation. Experimental verification of the concepts will be required to show their validity.

II. BLANKET DESCRIPTION

The cross-sectional view of the blanket is shown in Fig. 1. The basic configuration is very similar to that of MARS blanket. However, instead of a single tube, a concentric tube is used to form the blanket. The outer tube contains 17Li-83Pb as the breeding material, while the inner tube contains draw salt (50% KNO_3-50% $NaNO_3$) as the coolant. Draw salt is chosen for its high oxidation potential and inertness toward water. Using electrically "non-conducting" draw salt as the coolant eliminates the MHD problem so that the blanket can be operated at low pressure.

The 17Li-83Pb is a complete static system. Tritium recovery is done by permeation toward the plasma. The tritium is collected with the plasma exhaust. Since the draw salt is a good oxiding agent, it is expected that the inner surface of the coolant tube will be covered by an oxide layer, thus reducing tri-

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tium permeation toward the salt coolant by a factor of 100. The ratio of tritium permeated toward the plasma and toward the coolant can be calculated by

$$X = \frac{P_m^o \, A^o \, X_{O_2}^i \, t^i}{P_m^i \, A^i \, X_{O_2}^o \, t^o}$$

in which

P_m = permeability of the material
A = surface area
X_{O_2} = oxide barrier factor
t = wall thickness
and the superscript o and i are outer wall and innter wall, respectively.

The permeation factor, X, is ~1000 if both inner and outer tube are made of HT-9. For this case, 2500 Curie/day of tritium will enter the draw salt if the production rate is 250 g/d. Therefore, a tritium clean up system will still be required for the coolant. If the outer tube is made of vanadium while the inner tube is made of HT-9, the permeation factor will be 2×10^5, or only 12 Curie/day of tritium will permeate to the coolant. The outer tube is always at a higher temperature than the inner tube. Therefore, impurities transfer from the HT-9 inner tube to the V outer tube can be ignored.

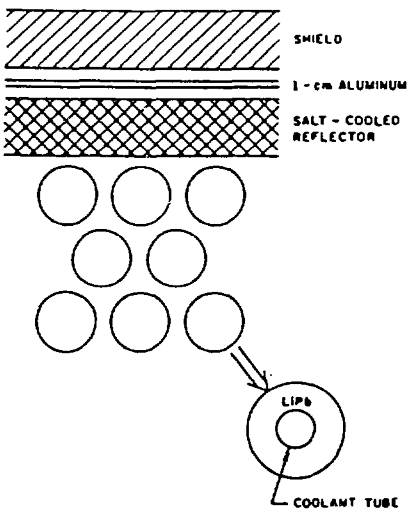

Figure.1 Cross-sectional view of the blanket.

Draw salt is a good coolant and reasonably stable under 500°C.[3] The stability in a fusion environment is not known. The heat transfer characteristics are very similar to that of water and, therefore, can provide sufficient cooling for a fusion reactor blanket. The major parameters for a Mini-MARS size machine are summarized in Table 1.

TABLE 1. BLANKET PARAMETERS

Fusion Power, MW	1000
Blanket Energy Multiplication	1.29
Tritium Breeding Ratio	1.24
Tritium Production Rate, g/d	195
Tritium Leakage to Coolant, g/d	.2
Coolant Temperature, °C	350-450
Coolant Pressure, MPa	.3
Coolant Flow Rate, kg/hr	1.8×10^7
Maximum Blanket Temperature, °C	550

III. NEUTRONICS CALCULATIONS

Neutronics calculations were performed using the Monte Carlo Code (MCNP)[4] and the continuous energy ENDF/B-V based nuclear data libraries.[5] The blanket model is an infinite cylinder consisting of three annular zones: (1) plasma and vacuum zone; (2) tritium breeding zone; and (3) reflector zone. The plasma and vacuum zone radii are 0.45 and 0.60 m, respectively. The combined breeding and reflector zone thickness is fixed at 1 meter. The breeding zone consists of 5% HT9 structure, 20% void, and adjustable fractions of draw salt coolant and lithium-lead eutectic, all by volume. The reflector is composed of 5% draw salt coolant and 95% metallic reflector (either HT-9 or Fe1422), also all by volume. The statistical errors in all calculations are less than 1% for both tritium breeding and nuclear heating values.

Table 2 shows the neutronic results of several blanket design options. As shown in Table 2, tritium breeding ratios from 1.09 to 1.24 tritons per D-T neutron can be obtained if the breeding zone thickness is from 0.4 to 0.5 m, and the draw salt coolant volume fraction ranges from 5 to 10%. The total blanket (breeding and reflector zones) nuclear heating ranges from 16.9 to 18.8 MeV per D-T neutron with the Fe1422 metallic reflector option giving higher heating values. The nuclear energy leakage out of these blanket systems is, in general, less than 2%, as shown in Table 2.

The maximum volumetric nuclear heating values at 1 MW/m^2 neutron wall are 6.7, 5.9 and 5.8 watts/cc, respectively in the 17Li-83Pb, draw salt, and HT-9 structure. Figure 2 depicts the spatial nuclear heating in the 40 cm breeding zone blanket where the coolant fraction is 5% by volume. The total blanket nuclear heating estimated for this blanket, as seen in Table 2, is 17.4 MeV per D-T neutron.

The nuclear heating values due to radioactive decay of the induced radionuclides in the blanket were also estimated for the case after two years operation at 2 MW/m^2 wall load. The decay heat in the HT-9 structure was found to be dominated by 56Mn(half life=2.6 hours) and 54Mn(half life=313 days), as shown in Fig. 3. The maximum decay heating at shutdown occurs at the first wall facing the plasma and is about 0.5 w/cc, or 4% of the operating value. At one day after shutdown, it decays by about one order of magnitude to 0.04 w/cc, and is dominated by 54Mn thereafter. The decay heat in the 17Li-83Pb breeder is mainly due to 203Pb(half-life=2.17 days). The maximum heating at shutdown is about 6.6 mw/cc, or 0.05% of the operating value. It occurs also at the location closest to the plasma. Both decay heating values in the HT9 structure and 17Li-83Pb breeder decrease exponentially as the location moves away from the first wall with an attenuation coefficient of 0.11 cm^{-1}.

The shutdown radioactivity and related safety issues of the nitrate salts were discussed in Ref. 3 and will not be repeated in this paper.

TABLE 2.
NEUTRONIC RESULTS OF SEVERAL DRAW SALT COOLING/17Li-83Pb BREEDING BLANKET OPTIONS
(TOTAL BLANKET THICKNESS = 1 m)

	Blanket Design Option A	B	C	D
Breeding Zone Thickness (cm)	40	40	50	50
Volume Fraction of Coolant (%)	5	10	10	10
Reflector Material	HT9	Fe1422	HT9	Fe1422
Tritium Breeding Ratio	1.088	1.090	1.235	1.240
Nuclear Heating (MeV/D-T Neutron)				
Neutron	8.62	8.64	9.36	9.37
Gamma-Ray	8.77	10.14	7.57	8.77
Blanket Total	17.39	18.78	16.93	18.14
Leakage	0.034	0.020	0.031	0.028

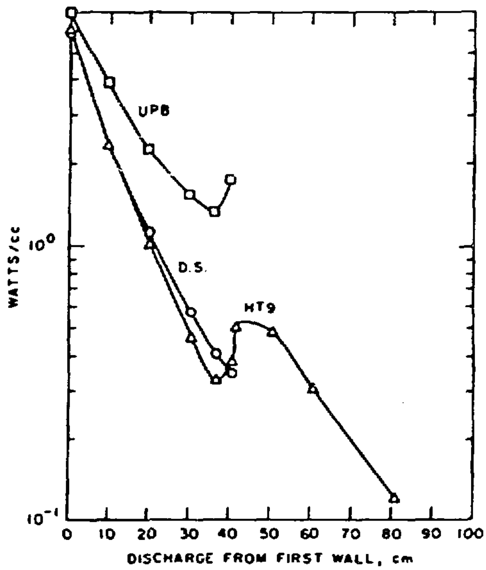

Figure 2. Volumetric Nuclear Heating (w/cc).

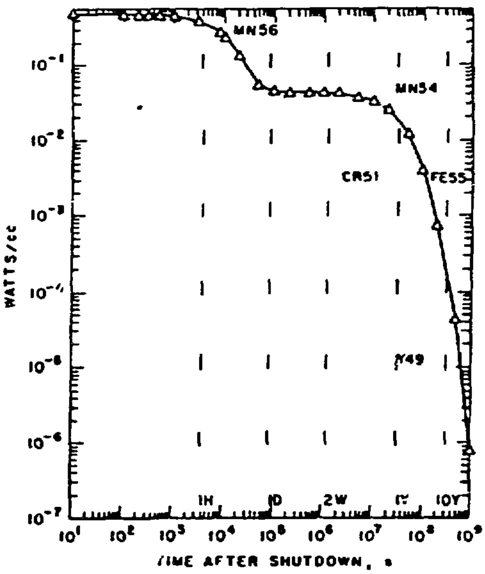

Figure. 3. Decay heat from HT-9 (HT-9/LiPb).

IV. POWER CONVERSION

The unique approach in the power conversion system involves a "direct contact steam generator," as shown in Fig. 4. The steam generator is basically a multiple stage, bubble cap distillation column. During each stage, water or steam will be intimately mixed with draw salt where heat exchange occurs as shown in Fig. 5. Since heat transfer is by direct contact mixing, the heat transfer will be rapid. In fact, in most distillation column designs, it has generally been assumed that equilibrium can be reached in each stage. Experience on the PWR has shown that it is difficult to maintain the mechanical integrity of the thousands of tubes in a tube shell steam generator. The direct contact steam generator is a possible method to eliminate the tubes. This can be done in the draw salt system because:

1. Draw salt does not react with water.
2. The solubility of water in salt is low.
3. The vapor pressure of the salt is low.

To use a direct contact steam generator an intermediate draw salt heat exchanger (IHX) is required to isolate corrosion products from the blanket region and dissolved tritium from the steam generator. This heat exchanger is operated at very low pressure (~.2 MPa). It should be reliable and cheap. The IHX is located near the nuclear island to reduce the size of the high radiation area. The secondary loop, including the steam generator should be free of activation. Therefore, hands on maintenance should be feasible.

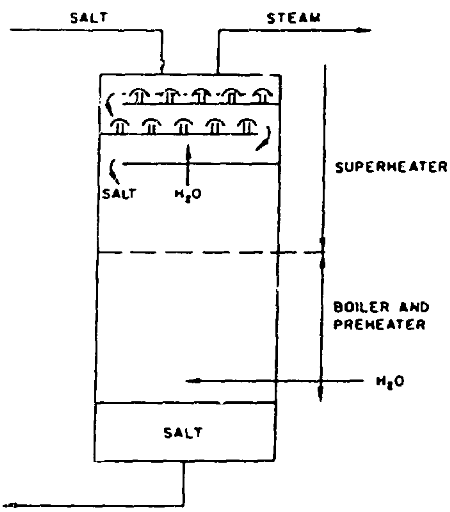

Figure 4. Direct contact steam generator (distillation column).

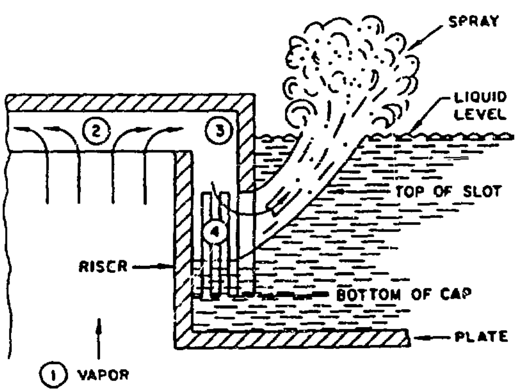

Figure 5. Cross section of bubble cap.

If the IHX is operated at low pressure, a pressure isolation system has to be placed between the steam generator and the IHX to decouple the high pressure in the steam generator (>100 MPa) from the low pressure IHX (.2 MPa). A regenerator pump concept is developed to accomplish this. The salt exiting the steam generator is at high pressure. This pressure head is used to drive a compressor to pressurize the low pressure salt from the IHX before it is fed to the steam generator. To account for the inefficiency of the pressure transfer, a booster pump is needed to further pressurize the incoming salt to the steam generator. The arrangement of the power conversion system is shown in Fig. 6.

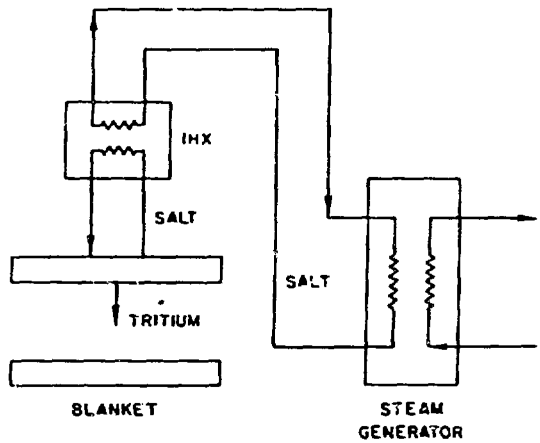

Figure 6. LiPb breeding/salt cooled blanket concept.

V. CONCLUSIONS

A blanket concept is presented which is a possible candidate for Mini-MARS design. The blanket is an improved version of the MARS blanket. The new concept alleviates or eliminates problems associated with tritium containment, corrosion, and MHD. A direct contact steam generator is incorporated in the design with low pressure (<0.2 Pa) in the primary system. This low pressure should improve the reliability of the blanket system.

The key unresolved problems are: 1) the activation of the salt and 2) the stability of the salt in fusion environment. Effort is underway to identify some other coolant which has lower activation and higher stability, but maintains the advantage available from draw salt.

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