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ABSTRACT

A self-cooled Flibe blanket concept has been developed. The problems associated with tritium breeding, tritium containment, and corrosion have been investigated and potential solutions developed. A highly efficient and compact blanket and power conversion system has been incorporated. The resulting system is low pressure with high thermal efficiency and is inherently safe.

1. INTRODUCTION

A self-cooled Flibe blanket using V-alloy as the structural material has been developed as part of ANL's effort on TPSS/Mini-MARS work. Flibe was used as the coolant for the molten salt reactor at ORNL¹ which was successfully operated for over 10 years. In the early stage of the fusion work, Flibe was suggested as the breeding material in a helium cooled blanket.² The deficiency of the breeding potential, primarily due to the low lithium concentration, requires additional neutron multiplier which complicated the blanket design.³ The very low tritium solubility makes tritium containment more difficult.⁴ The high material melting temperature causes additional operating problems. Therefore, with the exception of work at LLNL,³ the work on Flibe in fusion reactor applications has basically stopped.

There are a number of attractive features for Flibe, such as:

1. Low pressure operation.
2. Inert to water and air.
3. Low activation.
4. High temperature stability.
5. Low electrical conductivity.

If these features can be incorporated into a self-cooled blanket, a simple, compact, high performance and reliable blanket may be possible.

The idea of Li-doping of Flibe was developed. If lithium can be dissolved in Flibe, the excess lithium will react with tritium to form LiT which will be dissolved in Flibe. Thus, the tritium activity can be reduced. The tritium can be recovered by electrolysis process which has been demonstrated. The addition of lithium can also reduce the fluorine activity so that the corrosion problem may be reduced.⁵ Interestingly, neutronic analyses conducted by E. T. Cheng⁶ identified that Flibe can breed with a breeding ratio of 1.2 for a TMR geometry if V-alloy or HT-9 is used as the structural material. Further analysis shows that it can breed in a tokamak geometry as well, even when including the effect of the impurity control system. Therefore, a self-cooled Flibe blanket suddenly becomes much more attractive.

More detailed analysis uncovered some attractive features and some potential problems. A highly compact and efficient primary loop can be developed with a conversion of 46%. The blanket is designed to be inherently safe for at least 4.7 MW/m² neutron wall loading. Since no reflector is needed, the mass utilization is very high. A similar blanket for Mini-MARS has a high mass utilization figure of 138 KWe/tonne, the highest among the four blanket candidates. A very simple blanket can be designed.

There is doubt about the lithium doping idea for tritium control. The issues cannot be resolved until experimental verification is available. However, if V-alloy is used for the structural material, a large fraction of

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tritium can be permeated back to the plasma. By the permeation through the first wall a tritium partial pressure of $<10^{-4}$ Torr can be maintained. If this is the case, the primary blanket tritium recovery system is combined with plasma exhaust. A highly efficient tritium clean up system will still be required to reduce the tritium permeation problem. A counter-current extraction system can be used for tritium cleanup.

A number of issues exist, some of which are material dependent and some are design dependent. The more critical issues which required further evaluation include tritium management, V/Flibe compatibility, high temperature operating procedures and power cycle optimization.

II. DESIGN CONFIGURATIONS

The blanket configuration for TPSS is shown in Fig. 1. Since the first wall surface heat flux is high, the cooling of the first wall becomes important. Therefore, the coolant will enter the front part of the blanket region, cool the first wall, and return through the bulk of the blanket to recover the energy from nuclear heating. Thus, the first wall coolant will have high velocity and low temperature, which provides efficient cooling to the first wall. In the bulk blanket, the nuclear heating is deposited directly in the coolant so that heat transfer is not a critical consideration.

Fig 1. Flow path for self-cooled Flibe blanket.

The primary coolant system is divided into four separate loops, each one with a steam generator. For the reference design, no IHX is required because of the inertness of Flibe toward water. However, an IHX, or some

form of a double-walled steam generator may be required for adequate tritium containment. The size of the steam generator is very small due to the large temperature difference between the Flibe and steam. Due to its compactness and the inertness of the Flibe, the steam generator can be placed right next to the reactor, as in the case of the HTGR. Thus, the cost of piping as well as the inventory of Flibe, can be minimized.

III. MATERIAL CONSIDERATIONS

Flibe can be corrosive in the as received form. This is primarily due to the presence of free fluorine; the problem will be more critical if tritium is bred. Since the free energy of formation of TF is similar to that of VF_4 , VF_5 , and FeF_2 on a per g atom of F basis, it is expected that vanadium or iron will react and dissolve in the salt. Mo, W, Ni are more stable; this is the reason that the molten salt reactor used a high nickel alloy as the structural material. The free energy of formation of various fluorides have been tabulated as shown in Table 1.¹ Unfortunately, high nickel alloy cannot be used in a fusion environment due to both radiation enhanced induced embrittlement and unfavorable neutronics characteristics.

Table 1
Free Energies of Formation of Fluorides

	$\Delta G_{1000^\circ K}$ (kcal/g-atom of fluorine)
$MoF_6(g)$	-50.2
$WF_6(g)$	-56.8
$HfF_2(d)$	-55.3
$VF_5(g)$	-58.0
$VF_4(cr)$	-66.0
$Hf(g)$	-66.2
$FeF_2(d)$	-66.5
$NbF_5(g)$	-72.5
$CrF_2(d)$	-75.2
$TaF_5(g)$	-82.2
$TiF_4(g)$	-85.4
$LiF(l)$	-125.2
$BeF_2(l)$	-106.9

Transmutation of beryllium (as BeF_2 in Li_2BeF_4) leads to corrosion of any system metal since the disappearance of Be^{2+} is equivalent to the increase of fluorine activity by the reactions

The best material to reduce the fluorine activity is Li or Be due to the stability of LiF and BeF₂, as can be seen from the free energy of formation data listed in Table 1. A recent experiment has shown that, after addition of Be to Flibe, the corrosion of 316SS is reduced by almost a factor of 50, as shown in Fig. 2.⁵ The addition of Be in Flibe to reduce the fluorine activity is very similar to the addition of hydrogen in water to reduce the oxygen activity and thus reduces the corrosion rate. Since the free energy of formation for vanadium fluoride is very similar to that of iron fluoride, it is expected that the addition of Be in Flibe will have a similar effect in the vanadium system as in the steel system. The actual compatibility problem of vanadium alloys in flowing Flibe with excess Be has to be experimentally measured.

Fig. 2 Weight change versus exposure time for type 316 stainless steel in LiF-BeF₂ salt at the maximum loop temperature of 650°C.

The Flibe exits from the blanket at 650°C. This high temperature is very similar to that proposed for a molten salt reactor (705°C). Therefore, the technology developed for MSR is applicable. The structural material used in MSR is high nickel alloy for its high temperature properties and corrosion resistance. Therefore, the primary loop will be fabricated by two dissimilar metals, i.e., V-alloy for the blanket and high nickel alloy for the piping. A potential problem exists for impurity transport of C, N, and O from the high nickel alloy to the blanket. It is known that V-alloy is susceptible to embrittlement by impurity pick up. The severity of the

problem caused by the bi-metallic loop must be investigated. Fig. 3 shows the effect on the DBTT (ductile brittle transition temperature) of vanadium as a function of impurity level.⁷ It can be seen that a significant change of DBTT can occur with a fraction of one percent impurity.

Fig. 3 Effect of interstitial elements on the DBTT of vanadium.

IV. TRITIUM CONTAINMENT/RECOVERY

One of the most difficult problems associated with using Flibe is tritium management. The hydrogen isotopes dissolve in Flibe in molecular form and, therefore, obey Henry's law;

$$X = K_H P$$

in which K_H is Henry's law constant = .23 wppm/atm,¹ X is concentration, and P is pressure. Due to the linear dependency, the solubility of tritium in Flibe is very low at low T_2 pressure. Tritium can also exist as TF; however, excessive TF will cause a material compatibility problem. The increase of the T_2 partial pressure per pass in the blanket is .9 Torr for the Flibe, compared to 10^{-5} Torr for LiPb and 10^{-12} Torr for Li. For this reason, the tritium containment problem with the self-cooled Flibe blanket becomes a critical issue.

Different possible solutions have been studied to reduce the tritium activity for the Flibe blanket. Each of these provide a possible solution for tritium containment. However, without experimental verification, the validity of the methods is uncertain.

The first method considered is to tie up the tritium with lithium. It is known that LiT, which is relatively stable, will dissolve in the molten salt. If Be is dissolved in the salt for fluorine activity control, there may be a small but significant amount of lithium released to accomplish the dual function, i.e., reduce both the fluorine and tritium activity. When Be is dissolved in Flibe, the following reaction may occur

$$\Delta G \quad -460.02 \text{ kcal} \quad -427.6 \text{ kcal.}$$

Although the reaction is not favored by thermodynamic considerations, as indicated by ΔG values, the reaction will proceed because

$$\Delta G = -RT \ln \frac{[\text{BeF}_2]^2 [\text{Li}]^2}{[\text{Be}]}$$

$$= 32.42 \text{ kcal/mole}$$

If Be added = 0.01 and $[\text{Be}] = x$,

$$[\text{BeF}_2] = [\text{Li}] = 2(0.01 - x)$$

It can be shown that $[\text{Be}] = 7.5 \times 10^{-3}$

$$\text{and } [\text{Li}] = 5 \times 10^{-3}.$$

Therefore, even though thermodynamically unfavorable, significant amounts of free lithium can be released if Be is added to the salt. This free lithium is expected to tie up the excess tritium to reduce the tritium activity.

The LiT dissolved in the salt can be recovered easily by an electrolysis process, which has been demonstrated as the second half of molten salt tritium recovery process from lithium.⁸ The dissociation potential of LiT is around 1 volt while the dissociation potential of Flibe is 4.7 volts. This difference in dissociation potential can be used to dissociate LiT, while the Flibe remains stable. Such a tritium recovery system, but based on different materials, has been demonstrated experimentally.

If tritium cannot be tied up by lithium, the tritium will exist in T₂ form. By using V as the structural material, tritium can permeate back to the plasma. The tritium permeation rate as a function of the partial pressure has been calculated. For a 2.5 MW/m² neutron wall loading blanket, a tritium

partial pressure of 2×10^{-5} Torr can be maintained just by permeation. This partial pressure is still too high for the steam generator. Therefore, a tritium cleanup system is required to reduce the tritium concentration by a factor of 10. Since the amount of tritium involved in the cleanup system is small and the partial pressure is reasonably high, the cleanup system is easy to design.

An intermediate loop may also be considered to improve tritium containment. A salt to salt IHX has been developed for the MSR and can be adopted here. The impact on thermal efficiency should be small. The IHX will not only improve tritium containment, but also improve safety and reliability of the system. The cost associated with the intermediate loop has yet to be evaluated.

V. THERMAL HYDRAULICS

The heat transfer characteristics of Flibe are fairly similar to those of water, as shown in Table 2. The key difference is the viscosity which is 70 times higher. Fortunately, neither heat transfer nor pressure drop depends strongly on viscosity. A typical heat transfer coefficient for Flibe is 1.5 W/cm² - °C, while not as good as that of water, is much better than that of helium. Since the pressure drop is reasonable and the volumetric heat capacity is large, the power required to circulate the coolant is small.

Table 2
Comparison of Heat Transfer
Characteristics Between Flibe and Water

	Li ₂ BeF ₄ (600°C)	H ₂ O (260°C)
Cp, J/G - °C	2.8	5.0
ρ, G/cm ³	1.92	0.79
ν, CP	7.5	0.11
K, W/cm - °C	0.01	0.006
ρCp, J/cm ³ - °C	5.4	4.0

At same velocity H₂O = 2.5 H_{salt}

$$\Delta P_{\text{H}_2\text{O}} = .5 \Delta P_{\text{salt}}$$

for V = 500 cm/sec, D = 1 cm

$$H = 1.5 \text{ watt/cm}^2 - ^\circ\text{C}$$

$$\text{if } Q = 100 \text{ W/cm}^2, \Delta T = 67^\circ\text{C}$$

Since the melting temperature of the low viscosity form of the Flibe is 460°C, the

blanket inlet temperature of 550°C is selected. A reasonable coolant temperature rise of 100°C is used to be consistent with the power cycle considerations. The reference case of the TPSS reactor has a thermal power of 1735 MW and fusion power of 1425 MW, with a neutron wall loading of 3.8 MW/m². The surface heating load is assumed to be 76 W/cm². With this set of parameters, and assuming a blanket length of 2 m, the thermal hydraulics parameters for the blanket are calculated and summarized in Table 3.

Table 3
Blanket Thermal Hydraulic Parameters

Blanket Power	1735 MW
Coolant Temperature	550/650°C
Coolant Flow Rate	2.23×10^7 kg/hr
First Wall Channel Thickness	1 cm
Blanket Thickness	60 cm
First Wall Coolant Velocity	2.1 m/sec
Blanket Coolant Velocity	.04 m/sec
First Wall Heat Transfer Coefficient	7500 W/m ² - °C
Maximum First Wall Coolant Temperature	580°C
Maximum First Wall Temperature (coolant side)	641°C
Maximum First Wall Temperature (plasma side)	750°C
Maximum First Wall Thickness	5 mm
First Wall Sputtering Life	5 FPY

The high temperature and the large volumetric heat capacity results in a very compact and highly efficient power conversion system. The parameters of the power conversion system are summarized in Table 4. The surface area of the steam generator is only 2300 m², as compared to 3.9×10^4 m² for MARS. The small size is primarily due to the large pinch point ΔT of 150°C, compared to only ~10°C for MARS. The thermal power is divided into four parallel loops, with each loop handling 434 MW. The primary coolant pipe has an I.D. of 64 cm, which is a very reasonable size. The cost of the primary loop, due to its compactness and low pressure, should be very modest. The operation parameters of the primary loop are very similar to those of the molten salt reactor as indicated in Table 5.

Table 4
Power Conversion System Parameters

Thermal Power	1735 MW
Primary Coolant Temperature	550/650°C
Structural Temperature	750°C
Primary Coolant Flow Rate	2.2×10^7 kg/hr
Maximum Blanket Pressure	.3 MPa
ΔT] pinch point	150°C
Number of Loops	4
Diameter of Coolant Pipe	.64 m
Coolant Velocity	2.5 m/sec
Length and Diameter of SG	25 m x 1 m
Steam Generator Area	2300 m ²
Steam Condition	31 MPa, 566°C/566°C RH
Thermal Efficiency	46%

Table 5
Thermal Hydraulic Parameters of MSR and TPSS

	MSR*	TPSS
T_{in} , °C	565	550
T_{out} , °C	705	650
V_{max} , m/sec	2.6	2.5
$\Delta P]_{n.f.}$, KPa	124	100
$Q"]_{hx}$, W/cm ²	55	
$Q"]_{sg}$, W/cm ²	36	67
Maximum Design Pressure, MPa	.5	.3
Steam Generator Size (Dm x Lm)	.5 x 22 (121 MW _e)	.8 x 25 (250 MW _e)
Gross Efficiency	46%	46%
Net Cycle Efficiency	44.5%	45.5%

*Molten Salt Reactor.

VI. NEUTRONICS

This section presents the neutronic analysis for the self-cooled Flibe blanket designs. In the past, Flibe blankets have been considered not viable for fusion application due to insufficient tritium-breeding capability without neutron multipliers. However, Flibe blanket concepts have been reexamined anew in the current TPSS

study because of their potential design advantages.

Figure 4 shows a cross-sectional view of the computational model used for the analysis. This two-dimensional infinite cylinder model simulates the BCSS limiter and its plenum configurations as illustrated. The limiter model which covers about 10% of the first wall area consists of three layers, viz., (1) 1 cm beryllium coating which faces the plasma region; (2) 0.65 m thick (38.5% V15Cr5Ti + 61.5% Flibe) coolant channel; and (3) 5 cm thick V15Cr5Ti heat sink layer.

Fig. 4 A cross-sectional view of 2-D computational mode for Flibe blanket analysis.

The reference blanket is defined as follows:

First Wall : 3 mm : 100% V15Cr5Ti
Blanket : 60 cm : 8% V15Cr5Ti + 92% Flibe
Shield : 30 cm : 80% Fe1422 + 20% H₂O

All computations were carried out using the MCNP Monte Carlo transport technique. The associated nuclear data library is based on the ENDF/B-V nuclear file.

Figure 5 shows the TBR variation with ⁶Li enrichment for the reference V15Cr5Ti structure case and a case where the V15Cr5Ti structure is replaced by HT9 ferritic steel alloy. In the latter case, the limiter has been also replaced by the H₂O-cooled, Cu-28Be-heat sink, HT9 structure limiter of the type used in the BCSS study. One finds an optimal TBR at around 30% of ⁶Li enrichment, but the difference from the natural lithium (~7.4%) case is trivial. Figure 6 examines the accumulation of TBR as a function of depth in the blanket for natural lithium and 30% ⁶Li enrichment. It is found that for a TBR goal of 1.0, the blanket thickness required differs only by ~5 cm, being increased to ~8 cm for

1.1 TBR and to ~10 cm for 1.15 TBR. The implication is that the ultimate choice of ⁶Li enrichment depends upon the breeding criterion to be met. As the Flibe blanket does not pose any difficulty with ⁶Li burnup, the required TBR is perhaps the sole driving force to determine the ⁶Li enrichment.

Fig. 5 Effect on TBR of ⁶Li enrichment in V15Cr5Ti-structure and HT-9-structure Flibe blanket designs (limiter based on BCSS model; MCNP/ENDF/B-V; 5,000 neutron histories).

Fig. 6 Accumulation of tritium-bred in Flibe blanket (limiter based on BCSS model; MCNP/ENDF/B-V; 10,000 neutron histories).

The TBR of ~1.15 obtained for the natural lithium Flibe with V15Cr5Ti structure appears adequate to warrant the fuel self-sufficiency for this blanket design. Evidently the reasonably high breeding ratio has been brought about by the use of V15Cr5Ti as the structural material. In fact, the breeding decreases when HT9 is used instead of V15Cr5Ti by ~0.03 for natural lithium and by ~0.05 for

30% ^6Li enrichment as shown in Fig. 5. This is due primarily to a larger neutron loss in HT9 than in V15Cr5Ti.

The systems examined here yield a neutron multiplication of ~ 1.25 ($\pm 1\%$). This multiplication is comparable to that for a compositionally similar blanket such as the helium-cooled (90% Li_2O + 10% Be) mixture design. However the TBR obtained for the Flibe design is substantially lower than that for such a mixture design. The underlying neutronic process is appreciable neutron loss and spectrum softening in the Flibe design due mainly to fluorine in Flibe, resulting in a significant decrease in the $^7\text{Li}(n,n'\alpha)$ reaction rate.

The tritium breeding performance examined above is also nontrivially affected by the amount of structural material assumed. Figure 7 illustrates the TBR variation with structure volume fraction for the two alloy cases, via, V15Cr5Ti and HT9. The Flibe is assumed to have the naturally abundant lithium. The TBR varies from ~ 1.17 at 0% to ~ 1.08 at 20% of the V15Cr5Ti structure, by volume; approximately 8% TBR variation over 20% structure-volume variation.

Fig. 7 Effect on TBR of structure volume fraction in V15Cr5Ti-structure and HT-9-structure Flibe blanket design (limiter based on BCSS model; MCNP/ENDF/B-V; 5,000 neutron histories).

The TBR variation shown does not seem overly sensitive to structure content in the neighborhood of the reference point (8% V15Cr5Ti). However the mechanical design needs to be carefully carried out in terms of determining structure fraction in the blanket because the breeding degradation is quite appreciable beyond a fraction of 10%.

The impact on TBR of a large HT9

structure volume is undoubtedly more detrimental as clearly shown in Fig. 7. At 10% HT9, for instance, the TBR decreases ~ 1.1 , being further reduced to ~ 1.01 at 20% HT9. Note that the present design assumes a quite thin first wall; only three tenths of a centimeter.

The neutronic analysis presented so far has assumed a Flibe-cooled, V15Cr5Ti-heat-sink limiter based on the BCSS limiter configuration. Table 6 summarizes four cases to identify the impact of limiter on TBR. These cases include: (1) no limiter; (2) the reference Flibe-cooled V15Cr5Ti limiter; (3) water-cooled Cu-28Be heat-sink limiter; and (4) self-pumped limiter.

Table 6
Impact of Limiter Designs on
Tritium Breeding for Self-Cooled
Flibe Blanket Designs

	No Limiter	Flibe/V Limiter	H ₂ O/Cu-28Be Limiter	Self-Pumped Limiter
T ₆	1.05	1.02	1.04	1.05
T ₇	0.14	0.13	0.12	0.12
T ₆ +T ₇	1.19	1.15 ($\pm 1.3\%$)	1.16	1.17

All three different limiter concepts result in about the same TBR within a statistical fluctuation of $\sim \pm 1\%$. Comparing these TBRs with that of the no-limiter case, one finds no strong adverse effects on TBR, even when considering the relatively large limiter coverage ($\sim 10\%$) at the first wall that was assumed in these calculations.

As for the system nuclear heating, the reference blanket design results in a blanket energy deposition of 16.9 MeV, a shield energy deposition of 0.83 MeV and a system total of 17.7 MeV ($\pm 0.8\%$) per D-T neutron. The corresponding blanket and system energy multiplication factors (EMFs) are 1.20 and 1.26, respectively. These EMFs are quite comparable to those for the BCSS self-cooled, liquid-lithium blanket design. One notable feature of the Flibe blanket is a much steeper attenuation of local nuclear heating rate as illustrated in Fig. 8. This is due to the relatively soft neutron spectrum in the Flibe blanket as mentioned earlier. Also shown in Fig. 8 is the local tritium production rate, which has a similar profile because the primary source of nuclear heating in Flibe lies in the $^6\text{Li}(n,\alpha)t$ reaction.

Fig. 8 Spatial profiles of nuclear heating rate and tritium production rate in the reference self-cooled Flibe blanket (limiter based on BCSS model; MCNP/ENDF/B-V; 10,000 neutron histories).

VII. SPECIAL ISSUES

A. High Temperature Operation

The high melting temperature of Flibe may cause special operating problems for the reactor system. A trace heating system, a coolant dump system, etc. will be required. As a rule, the coolant cannot be allowed to freeze in the blanket and piping system. Similar problems exist for Li and ^{17}Li - ^{83}Pb , however, the higher Flibe melting temperature makes this problem particularly important.

The blanket and the piping are all well insulated from the environment. The small but finite amount of after heat will keep the blanket temperature above the melting temperature of Flibe for at least 10 days. The blanket should be designed such that the coolant can be dumped readily to a dump tank when it is required. A dump system such as this is a standard requirement for any liquid system and does not represent extra cost. The Flibe can be allowed to freeze in the dump tank on which a preheating system is built in to melt the Flibe.

For a molten salt reactor system, the whole reactor is enclosed within an "oven" which keeps the reactor temperature above the melting temperature of the Flibe in case the molten salt cannot be dumped within a certain time. Similar types of facilities can be used in the fusion reactor. A set of heating elements can be placed in between the blanket and the shield. Since the blanket temperature has to be kept above 500°C all the time, a very modest heating rate of $.6 \text{ W/cm}^2$ has to be

supplied. The total heating supplied is about 2 MW for the whole reactor.

For start up, the Flibe can be melted in the dump tank by a set of resistance heaters. At the same time, the blanket can be heated up using the built in heating element in between the blanket and shield, which serves the dual purpose of preheating the blanket and maintaining the temperature of the blanket. For a heating rate of 2 MW, the temperature of the blanket can be increased by 500°C in 5 hrs, which is a reasonable time for start up considerations.

B. Be Resources

The Flibe inventories in the blanket and primary loop are 3.2×10^5 and 2.8×10^5 kg, respectively. This represents total Be inventory of 5.5×10^4 kg, which corresponds to a 10-cm-thick equivalent of Be layer over the blanket. This inventory represents 0.075% of the estimated U.S. resource in each reactor which is quite significant. However, since Flibe can be re-used, there is no recycle problem. It is estimated that 4.5×10^4 GWe-Y of resource life is available based on U.S. resource estimates. If the world resource is considered, or a new resource is discovered, the power generation can be significantly increased.

VIII. CONCLUSIONS

The self cooled Flibe blanket is an attractive design due to its simplicity, low pressure operation, high allowable neutron wall loading for inherent safety, low activation, efficient thermal conversion, compact power cycle, and high mass utilization. It has the potential to become one of the most efficient blanket designs for D-T fusion, with only a reasonable extrapolation of state of art technology. However, many uncertainties exist which may impact the feasibility of the blanket, among which are:

1. Tritium containment: The thermodynamics and kinetics of interaction among lithium, Flibe, and tritium must be studied. The idea of using lithium to reduce the tritium and fluorine activities is interesting, but no experimental confirmation is available at this time. If lithium does not work as expected, Y may be dissolved in Flibe in ~ 100 ppm range.
2. Corrosion of V by Flibe: Addition of Be (or Li) to Flibe reduces the 316 ss corrosion by a factor of 100. Will similar phenomena occur to V-alloy? If it does, the chemical reaction problem between V and Flibe will be small, and the

dissolution of V in Flibe may be important.

3. V/High Nickel Alloy bimetallic loop: For cost effectiveness, it has been assumed that the primary coolant piping and steam generation are made of high nickel alloy. The transport and deposition of the impurity to the V structure and its impact on the material properties has to be investigated.
4. High temperature system: The melting temperature of Flibe is 460°C. The operation problem associated with a high melting temperature material must be assessed.
5. Be resonance: The Flibe blanket uses an equivalent thickness of 10 cm Be. Its impact on the Be resource may be significant.

A reasonable size experimental program can be set to provide answers for many of the issues, the most critical of which are listed in 1 to 3.

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