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A Study on Breakeven Feasibility in Molten Salt Fast Reactors

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1. Introduction

In order to achieve carbon neutrality, sustainable energy sources are being developed, including nuclear power. There is a strong emphasis on the development of Generation IV Reactors, which includes the Molten Salt Reactor (MSR). The MSR is a reactor that uses molten salt as both fuel and coolant and has many safety advantages, such as a stable operating environment with low pressure, strong negative feedback, and convenient residual heat removal [1]. The Molten Salt Fast Reactor (MSFR) is an advanced form of the MSR that utilizes a fast-spectrum and is currently being developed worldwide [2].

There have been many attempts to develop long-lived and high-burnup MSFRs. This paper introduces a feasibility study that investigates the achievement of a long-lived and high-burnup reactor through the breakeven method, which involves fuel conversion. In other words, the study observes whether the reactor can maintain criticality through self-supplement during fuel depletion.

2. Methods and Results

2.1 Basic Concept

Fertile materials, such as Th-232, U-234, U-238, and Pu-240, can be converted to fissile materials, such as U-233, U-235, Pu-239, and Pu-241, through neutron capture. The conversion ratio is defined as the number of fissile material atoms produced when one fissile atom is consumed. Breakeven is the concept of maintaining equilibrium between the loss and gain of fissile material by adjusting the value of the conversion ratio. One factor that affects the conversion ratio is the number of neutrons produced per neutron absorption (η). According to Figure 1 [3], Pu-239 has the highest η value for high neutron energies greater than 100 keV. This means that the conversion ratio of fuel with a larger fraction of Pu-239 becomes greater. For most uranium-loaded fast reactors, the conversion ratio increases during depletion as the amount of Pu-239 increases. The strategy is to use starting fuel with some amount of transuranic (TRU) material to increase the conversion ratio and achieve breakeven.

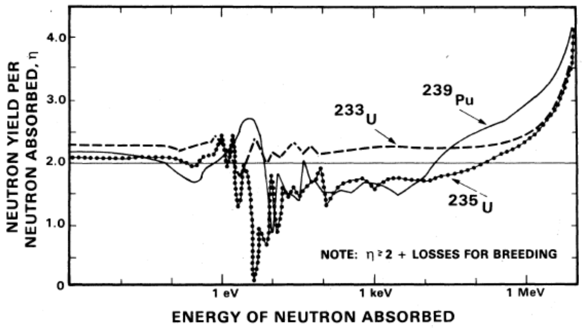

Fig. 1. Neutrons produced per absorption vs neutron energy for fissile materials [3]

2.2 Description of Molten Salts and Reactors

The basic requirements for molten salt fuel are that the heavy metal composition should be sufficiently high and that the melting temperature should be sufficiently low. Table I describes uranium-based molten salt eutectic data that meet these requirements. As mentioned earlier, a proper fraction of TRU should be included for breakeven. For preliminary analysis, $KCl\text{-}TRUCl_3\text{-}UCl_3$, $NaCl\text{-}KCl\text{-}TRUCl_3\text{-}UCl_3$, and $NaF\text{-}KF\text{-}TRUF_4\text{-}UF_4$ were selected as fuels, and their compositions are based on uranium-based molten salt eutectic information from Table I [4, 5, 6]. For example, the molar composition of $KCl\text{-}TRUCl_3\text{-}UCl_3$ is set to 46-x-y, where x and y can be adjusted to maintain the sum of x and y at 54. Note that Cl is enriched as 99 at.% Cl-37, uranium is natural uranium, and TRU is extracted from the spent fuel of PWR. Two types of TRU are being considered: pure TRU, which is composed of plutonium and minor actinides, and TRU that includes approximately 11% rare earths (RE). Their weight percentages are listed in Tables II and III [7].

Table I: Uranium-based Molten Salt Eutectic Data [4, 5, 6]

	Molar composition	Melting temperature	Uranium Density
$KCl\text{-}UCl_3$	46-54	558°C	2.179 g/cm^3
$NaCl\text{-}KCl\text{-}UCl_3$	42.9-20.3-36.8	470°C	1.740 g/cm^3
$NaF\text{-}KF\text{-}UF_4$	45.0-22.2-32.8	535°C	2.600 g/cm^3

Table II: Pure TRU composition from spent fuel of PWR (unit: wt.%) [7]

Ac	1.02E-09	Np	6.131	Cm	0.716
Th	7.40E-05	Pu	85.383	Bk	4.28E-11
Pa	5.92E-06	Am	7.770	Cf	1.92E-07

Table III: RE-included TRU composition from spent fuel of PWR (unit: wt.%) [7]

Ac	9.11E-10	Cf	1.7E-07	Pm	0.013
Th	6.58E-05	Yb	8.77E-07	Gd	0.231
Pa	5.26E-06	Lu	2.87E-20	Tb	0.004
Np	5.450	Y	0.014	Dy	0.003
Pu	75.896	La	0.737	Ho	1.33E-04
Am	6.907	Ce	2.893	Er	5.11E-05
Cm	0.636	Pr	1.503	Tm	4.16E-07
Bk	3.81E-11	Nd	5.715		

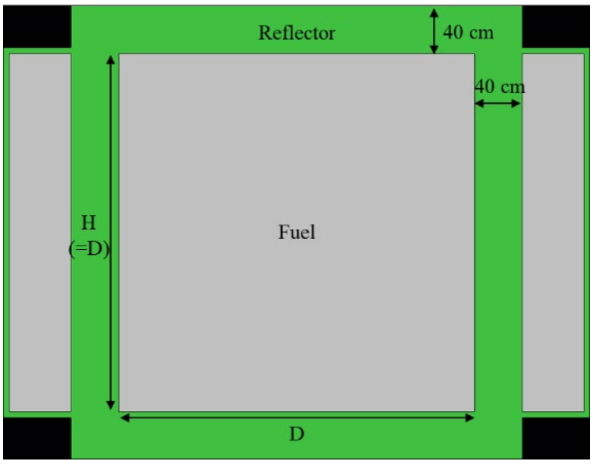

Fig. 2. Neutrons produced per absorption vs neutron energy for fissile materials

Figure 2 illustrates the reactor design, which consists of a cylindrical core with a diameter and height of the same size. The core is surrounded by a 40 cm thick stainless steel reflector, with a 0.1 cm thick Hastelloy-N coating on the inner surface to prevent corrosion. The outer region of the reflector contains an inactive core with the same volume as the active core. The active core has a diameter (equal to its height) of either 200 cm, 300 cm, or 400 cm.

For each of the three reactor sizes, six fuel salts (KCl-$TRUCl_3$-UCl_3, NaCl-KCl-$TRUCl_3$-UCl_3, and NaF-KF-$TRUF_4$-UF_4 with pure and RE-included TRU) are used, with the TRU and uranium composition adjusted to maintain criticality with less than 1,000 pcm. Tables IV and V provide information on the properties of the 18 different reactors considered. As the reactor size increases, a smaller fraction of TRU is required for criticality, and using RE-included TRU increases the required fraction of TRU. Among the salts, NaF-KF-$TRUF_4$-UF_4 has the largest mass of uranium and TRU, while NaCl-KCl-$TRUCl_3$-UCl_3 has the smallest.

Table IV: Reactors' Properties of Pure TRU

		KCl-$TRUCl_3$-UCl_3	NaCl-KCl-$TRUCl_3$-UCl_3	NaF-KF-$TRUF_4$-UF_4
H = D = 200 cm	Mole fraction	46.0-9.8-44.2	42.9-20.3-8.2-28.6	45.0-22.2-8.0-24.8
	U mass	22446 kg	17008 kg	24701 kg
	Pu + MA mass	5013 kg	4912 kg	8025 kg
H = D = 300 cm	Mole fraction	46.0-7.6-46.4	42.9-20.3-6.3-30.5	45.0-22.2-7.3-25.5
	U mass	79598 kg	61270 kg	85811 kg
	Pu + MA mass	13132 kg	12747 kg	24741 kg
H = D = 400 cm	Mole fraction	46.0-6.6-47.4	42.9-20.3-5.5-31.3	45.0-22.2-7.0-25.8
	U mass	192680 kg	148995 kg	205675 kg
	Pu + MA mass	27023 kg	26369 kg	56203 kg

Table V: Reactors' Properties of RE-included TRU

		KCl-$TRUCl_3$-UCl_3	NaCl-KCl-$TRUCl_3$-UCl_3	NaF-KF-$TRUF_4$-UF_4
H = D = 200 cm	Mole fraction	46.0-12.0-42.0	42.9-20.3-9.9-26.9	45.0-22.2-9.6-23.2
	U mass	21327 kg	15999 kg	23106 kg
	Pu + MA mass	5073 kg	4903 kg	7961 kg
H = D = 300 cm	Mole fraction	46.0-9.2-44.8	42.9-20.3-7.6-29.2	45.0-22.2-8.8-24.0
	U mass	76848 kg	58657 kg	80751 kg
	Pu + MA mass	13140 kg	12712 kg	24652 kg
H = D = 400 cm	Mole fraction	46.0-8.1-45.9	42.9-20.3-6.7-30.1	45.0-22.2-8.5-24.3
	U mass	186565 kg	143280 kg	193745 kg
	Pu + MA mass	27413 kg	26555 kg	56427 kg

2.3. Numerical Results

To examine the feasibility of achieving breakeven, fuel depletion calculations were performed for 20 effective full power years. The reactor powers were 500 MWth, 1680 MWth, and 4000 MWth for 200 cm, 300 cm, and 400 cm diameter reactors, respectively, with the power being proportional to their active core volumes. The Monte Carlo-based reactor analysis program Serpent 2.2.0 was used, along with the ENDF/B-VII.1 nuclear library. One hundred thousand histories were used, with 100 and 300 cycles chosen for inactive and active cycles, respectively.

Figures 3, 4, and 5 show the burnup vs full-power operation time curves for the reactors. The slopes of the burnup curves are almost identical for reactors with the same salt, since their power levels are proportional to their volumes. The slope of the burnup curve is proportional to the reciprocal of the initial heavy metal inventory. The reactor using NaCl-KCl-$TRUCl_3$-UCl_3 has the fastest burnup increase, reaching about 170 MWd/kg, due to its small mass of uranium and TRU.

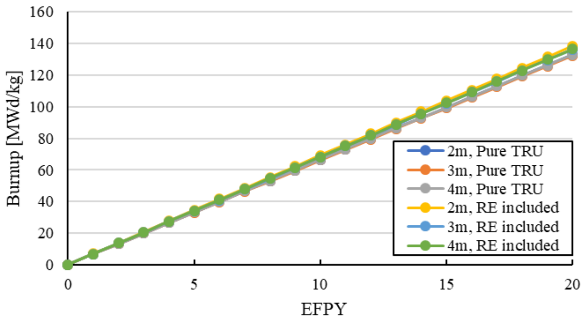

Fig. 3. Burnups of reactors with KCl-$TRUCl_3$-UCl_3 vs full-power operation time

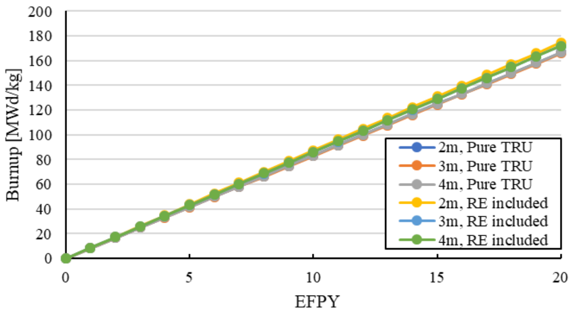

Fig. 4. Burnups of reactors with NaCl-KCl-$TRUCl_3$-UCl_3 vs full-power operation time

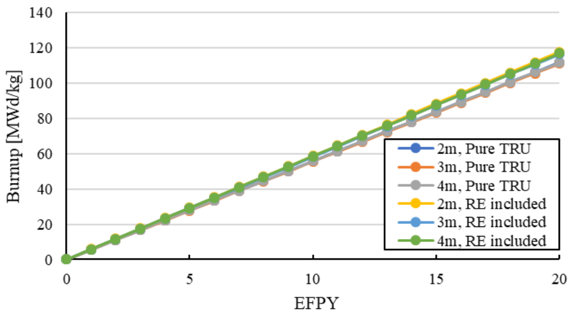

Fig. 5. Burnups of reactors with NaF-KF-$TRUF_4$-UF_4 vs full-power operation time

Figures 6, 7, and 8 show the reactivities vs full-power operation time, and Figs. 9, 10, and 11 show the conversion ratios vs full-power operation time. Overall, a high conversion ratio can be achieved because most of the fissile material is Pu-239. If the reactor size is increased, both the reactivity curve slope and the conversion ratio increase due to a decrease in neutron leakage and an increase in the amount of main fertile material, U-238. However, the inclusion of rare earths slightly decreases the conversion ratio due to neutron absorption. For KCl-$TRUCl_3$-UCl_3, if the reactor size is 4 m, reactivity increases up to over 4,000 pcm and then decreases. For NaCl-KCl-$TRUCl_3$-UCl_3, if the reactor size is 4 m, reactivity reaches around 1,000 pcm and then decreases. In all other cases, reactivity monotonically decreases. Especially for fluorides, the conversion ratio is smaller than with other fuels, and the reactivity monotonically decreases for any size due to the reactor's softer neutron energy spectrum, which gives a smaller η value of Pu-239.

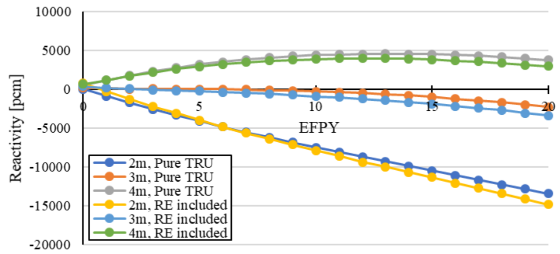

Fig. 6. Reactivities of reactors with KCl-$TRUCl_3$-UCl_3 vs full-power operation time

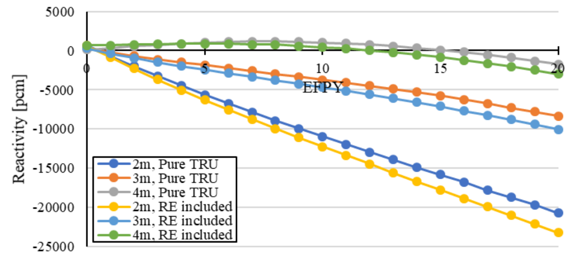

Fig. 7. Reactivities of reactors with NaCl-KCl-$TRUCl_3$-UCl_3 vs full-power operation time

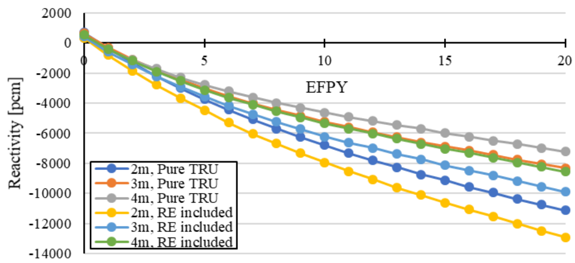

Fig. 8. Reactivities of reactors with NaF-KF-$TRUF_4$-UF_4 vs full-power operation time

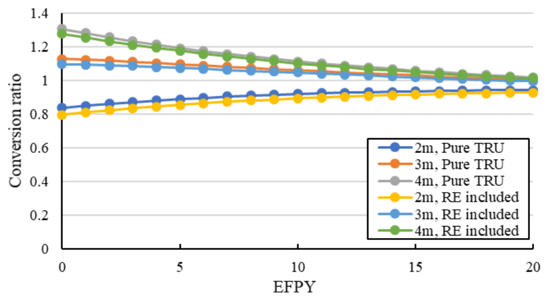

Fig. 9. Conversion ratios of reactors with KCl-$TRUCl_3$-UCl_3 vs full-power operation time

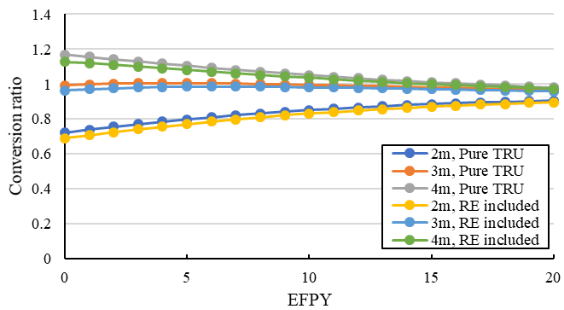

Fig. 10. Conversion ratios of reactors with NaCl-KCl-$TRUCl_3$-UCl_3 vs full-power operation time

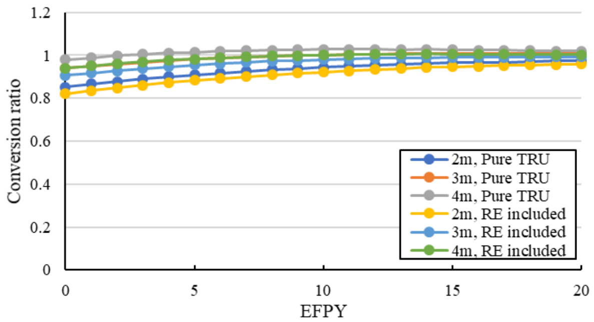

Fig. 11. Conversion ratios of reactors with NaF-KF-$TRUF_4$-UF_4 vs full-power operation time

3. Conclusions

The feasibility of achieving breakeven was examined for six molten salt fuels and three different reactor sizes through fuel depletion calculations. The results indicate that KCl-$TRUCl_3$-UCl_3 can achieve breakeven if the reactor size is slightly larger than 3m, while NaCl-KCl-$TRUCl_3$-UCl_3 can achieve breakeven with a reactor size of around 4m. However, NaF-KF-$TRUF_4$-UF_4 does not have sufficient conversion ratio to achieve breakeven. Therefore, further research will focus on optimizing the reactor with chloride fuels to achieve breakeven and enable long-term operation of the reactor while effectively utilizing natural or depleted uranium and spent fuel.

ACKNOWLEDGMENTS

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