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A study of long-term cycle and sensitivity evaluation with NaCl-KCl-UCl3 fueled MSR core design

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1. Introduction

The Molten Salt Reactor (MSR) offers long-term cycle benefits and produces less radioactive waste. During operation, the MSR system can extract fission materials like noble gases or noble metals, enhancing fuel efficiency. Moreover, due to minimal carbon emissions during operation, MSR holds promise as an alternative energy source for transportation modes such as ships in the future. In order to design MSR core for ships, several conditions are required, such as weight, size and others.

This paper presents the core concept design of a NaCl-KCl-UCl_3 (U235 enriched 19.75 w/o) molten salt reactor suitable for ship applications, capable of sustained operation for 30 years.

Calculations and modeling were conducted using the OpenMC, a Monte Carlo-based particle transport simulation code. Most calculations condition have done with an inactive cycle of 100, active cycle of 150, and 100,000 particles, achieving a level of uncertainty of about 20 pcm.

First, k_{eff} sensitivity evaluations of nuclear fuel and reflector materials are conducted to determine the core size. After core sizing, depletion calculations are performed under conditions that 30-year operational cycle, and an assessment of inventories is conducted to ascertain the fission products and TRU.

2. Methods and Results

Several design objectives were considered for the marine MSR design. It was designed for a power of 100MWth, with a core design EFPY target of 30 years. Considering the operational availability and utilization rates, EFPY is ensured for a minimum of 27 years, with the ITC reactivity set to negative for reactor safety. The power distribution was designed to be around 100 W/cc on average to prevent concentration of power density in specific parts of the core. Additionally, the overall size of the core is within 4 m, and the weight is kept under 200 tons, including the reflector and reactor vessel.

2.1. MSR core model

The geometry and detailed parameters of the cylindrical core are shown as table and figure below. To prevent corrosion of the core's exterior, a coating of Alloy625 with a thickness of 0.08 cm is applied, followed by a 0.8 cm SS316H canning layer, and covered with a BeO reflector.

Table 1 Reactor Specification

	Material	Thickness
Fuel	NaCl-KCl-UCl_3 (U235 19.75 w/o)	90 cm (radius)
Coating	Alloy625	0.08 cm
Canning	SS316H	0.8 cm
Reflector	BeO	95 cm
Reactor Vessel	SS316H	5 cm

NaCl-KCl-UCl_3 molten salt is used as fuel, and the operating temperature is calculated to be 620°C. The composition of NaCl-KCl-UCl_3 is shown as table below.

Table 2 Composition of NaCl-KCl-UCl_3 (620 ℃)

Component	Density (g/cm^3)	Composition (a/o)
NaCl	1.654	0.429
KCl	1.605	0.203
UCl3	4.917	0.368
NaCl-KCl-UCl_3	3.315	1

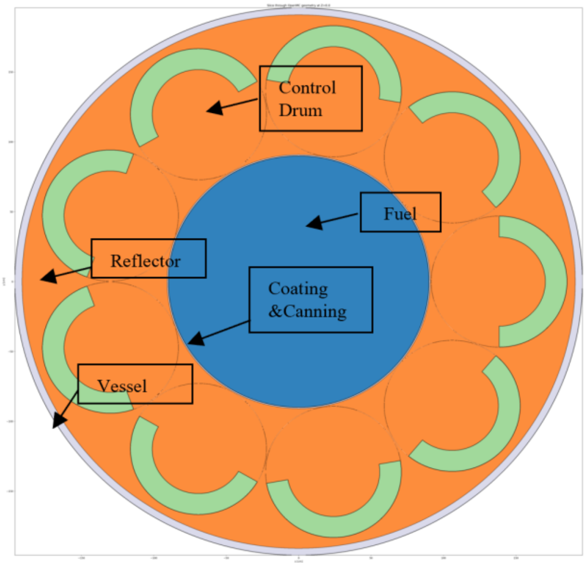

Fig. 1. Core x-y cross section with fuel, coating, canning, reflector, control drums and vessel.

2.2. MSR core sizing

For the selection of the 0.08 cm coating material to prevent corrosion with a nuclear fuel radius of 50 cm, Alloy 625 and Ni were compared based on reflector thickness to determine keff. The two materials showed an average difference of approximately 400 pcm, and the results are presented in the table and figure below. Alloy 625 exhibited a slightly higher keff, leading to its selection as the coating material.

Table 3 Coating Material Test with Fuel Radius 50 cm

Reflector [cm]	Alloy625, k_{eff}	Ni, k_{eff}
50	1.06858	1.06438
55	1.07338	1.06899
60	1.07693	1.07243
65	1.07916	1.07498
70	1.08128	1.07670
75	1.08241	1.07820
80	1.08387	1.07885
85	1.08382	1.07986
90	1.08479	1.08037
100	1.08565	1.08145
105	1.08578	1.08156
110	1.08519	1.08146
120	1.08573	1.08196

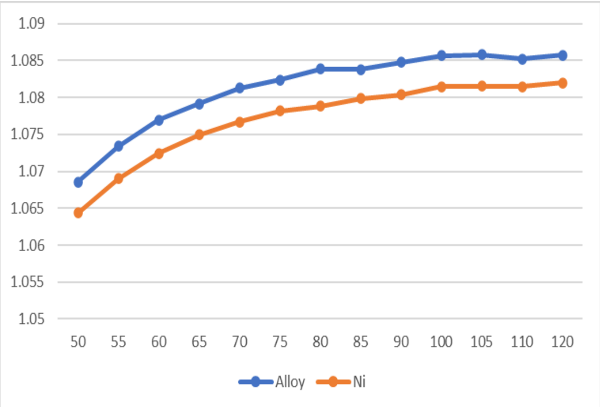

Fig. 2. Coating material test with Alloy625 and Ni by reflector thickness 5 cm intervals.

To determine the type of reflector, cases were divided into reflector thicknesses ranging from 45 cm to 120 cm at 5 cm intervals, under the condition of a fuel radius of 50 cm. Reflectors such as BeO, Beryllium metal, MgO, and PbO were selected, and calculations were performed, with the results presented in the table and figure.

Table 4 Reflector Test with Fuel Radius 50 cm

Reflector Thickness [cm]	BeO, k_{eff}	Be, k_{eff}	MgO, k_{eff}	PbO, k_{eff}
45	1.06202	1.03899	0.90825	0.79888
50	1.06859	1.04358	0.91190	0.80746
55	1.07343	1.04643	0.91452	0.81336
60	1.07684	1.04804	0.91590	0.81760
65	1.07968	1.04955	0.91714	0.82061
70	1.08176	1.05024	0.91762	0.82293
75	1.08258	1.05128	0.91846	0.82389
80	1.08378	1.05156	0.91841	0.82488
85	1.08453	1.05163	0.91871	0.82519
90	1.08494	1.05239	0.91882	0.82567
95	1.08563	1.05234	0.91872	0.82549
100	1.08603	1.05181	0.91860	0.82614
105	1.08603	1.05221	0.91855	0.82614
110	1.08616	1.05237	0.91891	0.82606
115	1.08623	1.05226	0.91880	0.82604
120	1.08645	1.05215	0.91877	0.82634

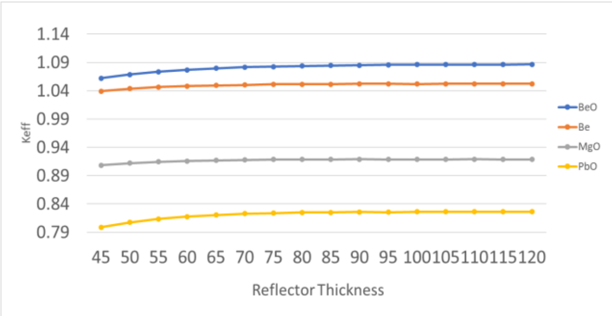

Fig. 3. Reflector sensitivity test with BeO, Be metal, MgO, PbO by 5 cm thickness intervals.

BeO was chosen as the reflector since it exhibited the highest keff values. Sensitivity evaluations were conducted by varying the thicknesses of nuclear fuel and reflector for BeO, and the results are illustrated in the figure below.

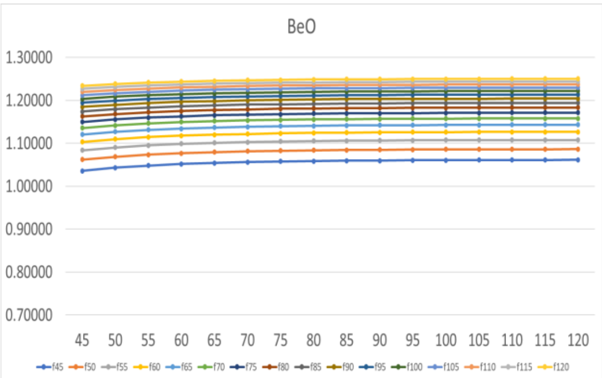

Fig. 4. BeO reflector sensitivity test with fuel radius and BeO thickness by 5 cm intervals.

2.3. Depletion calculation

Based on the data obtained from sensitivity evaluations of k_{eff} concerning nuclear fuel thickness and reflector thickness, several cases were selected to derive a model capable of sustaining combustion for 30 years with a 100MWth power. Considering the simulation of future control devices, cases with a final k_{eff} value providing a margin of about 5000 pcm after depletion were chosen.

Four cases were calculated: all reflector thickness are 100 cm and fuel radius at 80 cm, 85 cm, 90 cm, and 95 cm respectively. The results are shown as figure and table below.

Table 5 Fuel & Reflector Case Depletion Test

Fuel & Reflector	80, 100 [cm]	85, 100 [cm]	90, 100 [cm]	95, 100 [cm]
0 EFPY	1.18251	1.19376	1.20349	1.21291
5 EFPY	1.15171	1.16461	1.17604	1.18649
10 EFPY	1.12423	1.13797	1.15122	1.16344
15 EFPY	1.09799	1.11321	1.12794	1.14127
20 EFPY	1.08861	1.08861	1.10472	1.12019
25 EFPY	1.06550	1.06550	1.08256	1.09912
30 EFPY	1.02186	1.04108	1.06052	1.07817

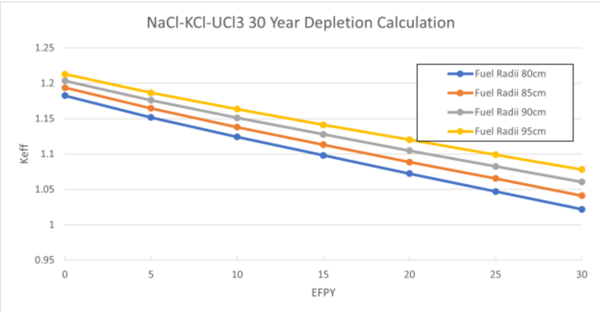

Fig. 5. 30year depletion test with four cases.

The results indicated that for the 85 cm case, depletion for 30 years was deemed unfeasible because of the effect of control drums. However, for the 90 cm case and 95 cm case were judged that depletion for 30 years was achievable. These cases show margin of approximately 6000 pcm and 7800 pcm were identified after 30 years of depletion. However, concerns arise regarding surpassing the 4 m constraint, considering the inclusion of cladding materials and canning. Therefore, for the fuel radii 90cm case, with a k_{eff} of 1.06052 after 30 years of depletion, this option seems capable of sustaining depletion for 30 years and meet under 4 m condition. Further assessments will be conducted considering future fission product removal, control drum simulations, and other factors.

The power distribution of this case is shown as figure below. It is part of core, half of radial and height, and it divides into 5 cm interval both in radial and height. It shows power in range of 13.30 W/cc to 83.50 W/cc levels. Reflector makes the corner of the core is higher than others.

56.72	56.60	55.85	56.15	55.42	55.02	53.89	53.85	52.87	52.17	51.32	50.43	49.38	48.27	47.48	47.60	51.61	83.50
23.02	22.81	22.72	22.75	22.61	22.42	22.22	21.91	21.71	21.60	21.22	20.99	20.71	20.62	20.75	21.66	25.83	52.73
17.08	17.05	17.12	17.09	16.89	16.78	16.69	16.56	16.38	16.25	16.15	15.95	15.88	15.99	16.36	17.56	22.00	49.16
15.24	15.28	15.23	15.20	15.09	14.97	14.88	14.79	14.67	14.57	14.48	14.40	14.37	14.58	15.10	16.47	21.14	48.91
14.41	14.36	14.38	14.35	14.28	14.21	14.12	14.03	13.93	13.85	13.78	13.76	13.82	14.06	14.70	16.14	21.05	49.63
13.98	13.95	13.98	13.90	13.90	13.83	13.71	13.62	13.58	13.50	13.47	13.48	13.61	13.90	14.56	16.17	21.28	50.82
13.76	13.82	13.81	13.76	13.70	13.61	13.55	13.44	13.39	13.34	13.32	13.38	13.53	13.87	14.57	16.25	21.62	52.01
13.62	13.73	13.69	13.65	13.56	13.54	13.47	13.38	13.35	13.32	13.30	13.37	13.56	13.95	14.69	16.45	21.79	52.85
13.69	13.65	13.64	13.64	13.55	13.51	13.45	13.42	13.35	13.33	13.36	13.43	13.62	14.03	14.81	16.58	22.01	53.61
13.73	13.69	13.67	13.61	13.56	13.54	13.46	13.45	13.42	13.42	13.42	13.49	13.70	14.11	14.96	16.77	22.38	54.64
13.72	13.76	13.69	13.67	13.62	13.56	13.51	13.49	13.45	13.45	13.47	13.54	13.77	14.22	15.05	16.97	22.65	55.18
13.78	13.81	13.74	13.72	13.68	13.63	13.57	13.54	13.53	13.53	13.56	13.66	13.89	14.33	15.18	17.07	22.83	55.87
13.80	13.80	13.81	13.78	13.75	13.72	13.66	13.63	13.61	13.60	13.62	13.72	13.98	14.43	15.31	17.29	23.06	56.41
13.93	13.86	13.82	13.85	13.81	13.76	13.75	13.69	13.67	13.66	13.69	13.82	14.07	14.52	15.41	17.37	23.30	56.93
13.98	13.92	13.87	13.90	13.87	13.83	13.77	13.70	13.71	13.70	13.78	13.87	14.11	14.58	15.51	17.39	23.33	57.27
13.98	13.97	13.99	13.95	13.91	13.88	13.81	13.74	13.70	13.75	13.79	13.95	14.19	14.65	15.53	17.47	23.40	57.46
14.13	14.03	14.03	13.98	13.95	13.88	13.85	13.79	13.76	13.78	13.84	13.97	14.20	14.69	15.56	17.55	23.52	57.56
14.20	14.12	14.04	14.01	13.95	13.92	13.83	13.82	13.78	13.82	13.86	13.97	14.22	14.69	15.65	17.61	23.57	57.61

Fig. 6. Power distribution of 1/4 core [W/cc].

MSR can remove noble gases and noble metals through the system. Therefore, an examination of the fission product was conducted following depletion calculations to ascertain the nuclear isotope inventory. The nuclear isotope inventory was examined in intervals of 5 years, and the results are presented in the table below. The amount of the noble gas (Kr, Xe) is 9.21E+01 kg, noble metal (Ru, Rh, Pd) is 5.78E+01 kg, fission products (Zr, Mo, I, Cs, La, Pr, Nd) is 3.25E+02 kg. The sum of these materials is 4.76E+02 kg. If the filtering and extracting system are designed, the margin of the weight and more fuel can add. The Pu and other TRU materials are calculated as 1.05E+02 kg.

Table 6 Nuclear Inventories by Depletion Steps BOC to 15 EFPY, 'F90R100'

		BOC	5EFPY	10EFPY	15EFPY
Uranium	U234	0.00E+00	3.60E+01	7.12E+01	1.07E+02
	U235	2.19E+06	2.06E+06	1.94E+06	1.82E+06
Noble Gas	U236	0.00E+00	2.59E+04	5.07E+04	7.46E+04
	U237	0.00E+00	6.72E+00	1.12E+01	1.48E+01
	U238	8.90E+06	8.85E+06	8.80E+06	8.75E+06
	Kr	0.00E+00	1.46E+03	2.89E+03	4.28E+03
	Xe	0.00E+00	1.40E+04	2.79E+04	4.19E+04
Noble Metal	Ru	0.00E+00	6.07E+03	1.22E+04	1.84E+04
	Rh	0.00E+00	1.55E+03	3.12E+03	4.68E+03
	Pd	0.00E+00	1.30E+03	2.89E+03	4.74E+03
Fission Product	Zr	0.00E+00	1.35E+04	2.69E+04	4.02E+04
	Mo	0.00E+00	1.11E+04	2.22E+04	3.34E+04
	I	0.00E+00	4.81E+02	9.71E+02	1.47E+03
	Cs	0.00E+00	1.20E+04	2.38E+04	3.54E+04
	La	0.00E+00	4.16E+03	8.28E+03	1.24E+04
	Pr	0.00E+00	5.30E+01	1.00E+02	1.49E+02
	Nd	0.00E+00	1.40E+04	2.80E+04	4.18E+04

Table 7 Nuclear Inventories by Depletion Steps 20 EFPY to 30 EFPY, 'F90R100'

	Mat	20EFPY	25EFPY	30EFPY	change [kg]
Noble Gas [kg]	Kr	1.44E+02	1.83E+02	2.26E+02	2.26E-01
	Xe	1.71E+06	1.60E+06	1.49E+06	-6.97E+02
Uranium [kg]	U234	9.75E+04	1.19E+05	1.40E+05	1.40E+02
	U235	1.83E+01	2.16E+01	2.49E+01	2.49E-02
	U236	8.70E+06	8.65E+06	8.59E+06	-3.04E+02
	U237	5.63E+03	6.96E+03	8.26E+03	8.26E+00
	U238	5.58E+04	6.98E+04	8.38E+04	8.38E+01
Noble	Ru	2.47E+04	3.11E+04	3.76E+04	3.76E+01

Metal [kg]	Rh	6.24E+03	7.79E+03	9.33E+03	9.33E+00
	Pd	6.83E+03	9.16E+03	1.17E+04	1.17E+01
Fission Product [kg]	Zr	5.34E+04	6.65E+04	7.95E+04	7.95E+01
	Mo	4.45E+04	5.55E+04	6.65E+04	6.65E+01
	I	1.98E+03	2.49E+03	3.02E+03	3.02E+00
	Cs	4.67E+04	5.77E+04	6.86E+04	6.86E+01
	La	1.64E+04	2.05E+04	2.45E+04	2.45E+01
	Pr	2.05E+02	2.70E+02	3.48E+02	3.48E-01
	Nd	5.56E+04	6.92E+04	8.27E+04	8.27E+01

3. Conclusions and future works

MSR for ship has been designed, capable of operating up to 27~30 EFPY with a 100MWth power. The design with 90 cm fuel radius allows for a margin of approximately 6000 pcm for future control system simulations. Through sensitivity evaluations and reflector assessments, the reactor's size and type of reflector have been determined, ensuring power distribution and a negative ITC reactivity for stability.

Furthermore, depletion calculations have been conducted to explore the potential for efficient removal of noble gases and noble metals. This for extending reactor lifetimes or reducing sizes in future designs. Future works involve implementing control device simulations to regulate initial criticality and exploring alternative structural designs.

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