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INTRODUCTION

Can the hybrid system combination of (1) a critical fission Molten Salt Reactor (MSR) having a thermal spectrum and a high Conversion Ratio (CR) with (2) an external source of high energy neutrons provide an attractive solution to the world's expanding demand for energy? The present study indicates the answer is an emphatic yes.

The external neutron source may be either a Deuterium+Tritium (DT) Fusion reactor emitting 14.1 Mev neutrons or an Accelerator Driven Spallation (ADS) neutron source emitting neutrons of that or even higher energy. In either case, the emitted fast neutrons are sufficiently energetic to fission any actinide, fissile or not, and the resulting daughter neutron fission yields are substantially greater than yields of fission chain reactions. Daughter neutrons are absorbed by fertile species, counterbalancing net fissile consumption in the MSR.

With fission products continuously removed the Hybrid Molten Salt Reactor (HMSR) completely fissions any supplied mix of actinide fuel. Benefits include:

- No refueling outages are necessary.
- Spent Nuclear Fuel (SNF) from Light Water Reactors (LWRs) can fuel HMSRs.
- Excess plutonium can fuel HMSRs.
- Mined uranium can fuel HMSRs.
- Energy usage from mined uranium is increased from 1% in LWRs to 100% in HMSRs.
- Depleted uranium can fuel HMSRs.
- Mined thorium can fuel HMSRs.
- The HMSR waste stream has no long-lived radioactive actinides, thus reducing repository needs.
- Some long-lived radioactive fission products can be reduced or eliminated in HMSRs.
- Uranium fuel enrichment is eliminated.
- Solid fuel fabrication is eliminated.
- Fuel recycling is eliminated.
- HMSR fissile inventories are low.
- In HMSR steady-state inventories, fertile isotopes can naturally denature fissile isotopes, thus enhancing proliferation resistance.
- Proliferation concerns are reduced by eliminating LWR needs for enrichment, reprocessing, and fissile material transportation.
- The HMSR can utilize the MSR's ability to drain fuel to passively safe dump tanks, thus allowing an engineered safety feature not possible in solid fuel reactors.

A significant finding of the present study is that the energetic neutrons needed for HMSR operation can represent less than 1% of total thermal power. This implies that a HMSR can tolerate some inefficiency in its neutron source. For instance, a HMSR producing 5000 MW thermal power at 700°C, converted to generate 1600 MWe, could divert 500 MWe to operate its 50 MW energetic neutron source and still have 1100 MWe to sell.

It is also significant that the energetic neutron source need not operate continuously. Total HMSR power can follow load demand even with an intermittent neutron source as long as neutron source power averaged over days is maintained.

HMSR Configuration

Fig.1 schematically illustrates the HMSR configuration. A closed loop of pipes connects the following components: a Blanket of Tanks (1), which surrounds an adjacent Energetic Neutron Source (5), a Molten Salt Reactor (2), a Molten Salt Heat Exchanger (3), and a Molten Salt Circulating Pump (4). This loop of pipes and components 1 through 4 is oriented so that the Molten Salt Heat Exchanger is located at the highest elevation in the loop, above the Molten Salt Reactor to which it is directly connected. A pipe extends upward from the highest elevation in the loop to a closed pressurizer volume (6). An additional pipe extends downward from the lowest elevation in the loop, through a Salt Freeze Plug (7) to a set of Dump Tanks (8). The aggregate volume of the dump tanks exceeds the closed volume for molten salt above the dump tanks.

A molten salt liquid mixture of different ionic salt components including actinides fills and circulates rapidly around the loop, extending above the loop into the lower part of the closed pressurizer gas expansion volume. Molten salt also extends downwards to the salt freeze plug where deliberate external heat leakage causes the salt temperature to stay below its melting temperature so that during normal operation molten salt does not drain into the dump tanks. Not shown is the system to restore the salt freeze plug and molten salt from the dump tanks back to the main loop in order to resume normal operation.

For present studies the MSR's graphite was modeled as a matrix of 15 cm/side prismatic hexagonal blocks with 3.5 cm diameter molten salt channels. Cylindrical MSR size was set to 8.8 m for its diameter and height.

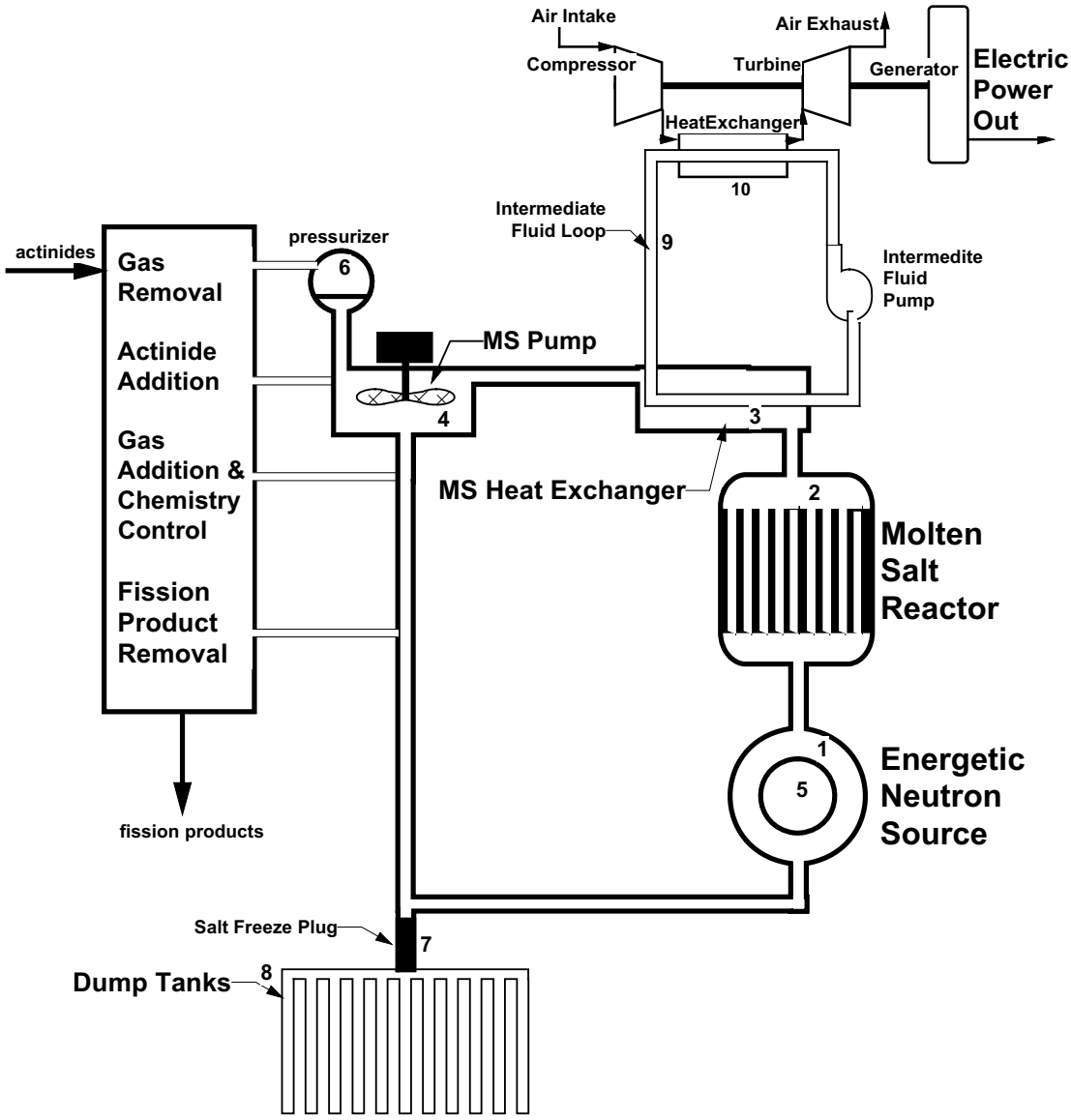

Fig. 1. Hybrid Molten Salt Reactor (HMSR) Configuration

High temperature heat extracted through Heat Exchanger (3) is converted to shaft rotation power in a thermal conversion system, then is converted to electric power. Fig.1 arbitrarily depicts a thermal conversion system using the simple Open Brayton Cycle, but other more complex schemes could confer higher efficiencies.

Molten salt mixtures investigated in the present study were 44.5 mole% lithium fluoride (LiF), 24.1 mole% sodium fluoride (NaF) and 31.4 mole% total $(HM)F_x$ where HM (Heavy Metal) represents actinide species and where x ranges from 4 for thorium through uranium to 3 for plutonium and higher. Equivalent approximate atom% or mass% values are listed in Table I.

For uranium with x=4 this mixture is eutectic with a 490°C melting point, thus supporting 600-706°C loop operation where the upper temperature was chosen to limit corrosion in Hastelloy-N, an alloy developed for its long-term compatibility with fluoride molten salts in a nuclear environment.

Table I. Approximate Molten Salt Composition

Element	Atom %	Mass%
Fluorine	66	31
Lithium	15	2
Sodium	8	5
HM (e.g., Uranium)	11	62

DESCRIPTION OF THE ACTUAL WORK

Modules from the SCALE 6.1 system of codes were used along with the 238-group ENDF/B-VII Release 0 cross section library. The Energetic Neutron Source was modeled as an isotropic uniform density volumetric spherical source of 3.5 m radius emitting neutrons in group 4 which runs from 13.84 MeV to 14.55 MeV.

Initial XSDRNPM code runs were made to choose the thickness of the molten salt blanket surrounding the source. A goal was set that neutron leakage be between 0.5% and 1.0% of the total of (1) the energetic source neutrons plus (2) the net additional neutrons produced within the blanket by n2n or n3n reactions or by fission. With the molten salt actinide content set to be entirely uranium-238, an 80 cm thickness yielded the required leakage. This was adopted for all subsequent runs.

Neutron reactions in this molten salt blanket carrying uranium-238 are summarized in Table II. For each energetic source neutron, 0.21872 fissions of uranium-238 occur releasing about 43 MeV of fission energy plus (0.82170 - 0.21872 = 0.60298) additional daughter neutrons beyond those consumed to initiate the fissions. There are also additional neutrons released by n2n and n3n reactions, totaling 0.00557 + 0.02632 + 0.00197 + 0.12613 + 2 * (0.03793) = 0.23585. Of the net total 1.83883 neutrons, 1.66253 are captured by other uranium-238 nuclides converting them into uranium-239 which after double beta decays become fissile plutonium-239.

Table II: Calculated Neutron Reactions in 80 cm Thick Blanket Containing U238 as Sole Actinide

Nuclide in Molten Salt	Reaction	Reactions per 14.1 MeV source neutron
lithium-7	n2n	0.00557
	n3n	0
	(n,γ)	0.00067
fluorine-19	n2n	0.02632
	n3n	0
	(n,γ)	0.00918
sodium-23	n2n	0.00197
	n3n	0
	(n,γ)	0.01053
uranium-238	n2n	0.12613
	n3n	0.03793
	fissions	0.21872
	fission daughters	0.82170
	(n,γ)	1.66253

The ORIGEN module of SCALE6.1 was used in subsequent runs to simulate evolution of the molten salt's isotope inventory caused by reactions both in the MSR and in the blanket. ORIGEN's inputs include single-group collapsed cross sections, neutron flux, exposure duration, steady continuous removal rates (sec^{-1}) for each element, and continuous addition rates for a set of fueling isotopes. Simulated powers of the MSR vs the neutron source were adjusted to maintain keff≈1 criticality using the facts that non-breeding MSR operation depletes fissile inventories, while operation of the energetic neutron source increases fissile inventories. For this adjustment, a software feedback loop functioning as a switching controller would first evaluate keff criticality for the MSR based on the molten salt's most recent evolved isotope inventory, then would either run ORIGEN for the blanket if keff<1 or for the MSR if keff≥1. Over successive loop iterations the average power ratio adjusts itself to maintain keff near unity. The energetic neutron source power was separately adjusted to maintain average wall loading at 0.5 MW/m^2 DT fusion power equivalent.

The overall objective was to find steady-state operating conditions. These consist of steady power levels, continuous removal rates (sec^{-1}) for fission/transmutation products, steady continuous addition rates for fueling isotopes, and an associated steady inventory of isotopes in the molten salt consistent with keff=1 criticality of the MSR. Simulations were run until averaged changes in the molten salt's isotope inventory became negligible, signaling steady-state conditions. Actinide removal rates were held at zero, thus requiring that actinides go in but never come out.

RESULTS

Table III lists final simulated values for total fission product inventory and power ratios for eight different cases. In Cases 1 through 6 the initial actinide inventories and continuously added actinides were entirely uranium-238. In Case 7 they were Spent Nuclear Fuel (SNF) from light water reactors and in Case 8 they were thorium-232. Case 1 suppressed fission product (FP) generation, equivalent to infinite FP removal rates. Case 2 assigned arbitrary removal rates to each FP element. Case 3 reduced removal rates by a factor of ten while Case 4 increased them by a factor of 10. It is significant that although Cases 2-4 varied FP removal rates by a factor of 100, they converged to steady solutions in which all actinides were fully consumed. Case 5 attempted to apply the successful actinide non-removal strategy to onerous long-lived FPs by zeroing removal rates for seven elements. The resulting inventories failed to stabilize due to build-up of stable FP isotopes of selenium, zirconium, and samarium. Case 6 returned those elements to their Case 2 removal rates. The non-removal strategy then obtained small strontium, tin, cesium, and iodine inventories, still growing slowly due to stable FP isotopes.

Case 7 simulated SNF in the initial load and in continuous fueling. Its higher power ratio results from SNF's inclusion of fissile material with uranium-238. Case 8 simulated thorium-232 fueling, showing the HMSR can consume all actinides from a thorium cycle.

Table III. Fission Product Inventories and Power Ratios

Case #	Fuel	FP removal rates	FP atom%	fission power/ energetic neutron power
1	^{238}U	infinite	0.000	1041
2	^{238}U	arbitrary	0.101	574
3	^{238}U	arbitrary/10	0.650	244
4	^{238}U	arbitrary*10	0.065	1048
5	^{238}U	arbitrary; 0 for Se, Zr,Sm,Sr,Sn,Cs,I*	>8.62*	N/A*
6	^{238}U	arbitrary; 0 for Sr,Sn,Cs,I	0.78	1021
7	SNF	same as #6	0.74	1053
8	^{232}Th	same as #6	0.53	184

*=unconverged

Table IV lists steady isotope inventories for fueling cases with uranium-238, with SNF from light water reactors, and with thorium. It is noteworthy that these cases have low fissile concentrations and fissile isotopes are mixed with non-fissile isotopes of the same elements.

Table IV: Steady Isotope Inventories (atom%)

Isotope	Case 2 (U238)	Case 7 (SNF)	Case 8 (Th232)
li7	15.124	14.958	14.769
f19	65.893	66.434	66.268
na23	8.204	8.084	8.010
FPs	0.101	0.739	0.527
th230	0.000	0.000	0.005
th232	0.000	0.000	10.027
pa231	0.000	0.000	0.002
pa233	0.000	0.000	0.002
u232*	0.000	0.000	0.006
u233*	0.000	0.000	0.208
u234	0.000	0.000	0.101
u235*	0.000	0.000	0.009
u236	0.005	0.029	0.053
u238	9.789	8.843	0.002
np237	0.002	0.002	0.003
np239	0.000	0.003	0.000
pu238	0.004	0.003	0.002
pu239*	0.047	0.042	0.000
pu240	0.082	0.065	0.001
pu241*	0.025	0.022	0.000
pu242	0.222	0.198	0.001
pu244	0.001	0.001	0.000
am241	0.001	0.000	0.000
am243	0.090	0.084	0.001
cm242	0.001	0.001	0.000
cm244	0.178	0.218	0.001
cm245*	0.004	0.006	0.000
cm246	0.175	0.209	0.001
cm247*	0.005	0.006	0.000
cm248	0.047	0.054	0.000

*fissile isotopes

CONCLUSION

The present study finds that by combining a graphite-moderated molten salt reactor implementing continuous fission product removal with a driven source of 14.1 MeV neutrons carrying less than 1% of total plant power, the resulting hybrid system can fully consume any combination of actinides as its fuel. The small energetic neutron fraction required suggests the hybrid could be built soon using existing DT fusion or accelerator-driven spallation neutron sources without much further technology development. The hybrid's greatly increased energy utilization, its reduction of long-lived radioactive wastes, its elimination of proliferation-vulnerable fuel cycle steps including enrichment and recycling, its elimination of expensive fuel fabrication, and its enabling of new engineered safety features, together make it an attractive solution to the world's expanding demand for carbon-free energy.

ENDNOTES

The first author would like to thank the second author for serving as major professor for the University of Tennessee, Knoxville dissertation research whose results are summarized here. While the first author is an employee of Princeton Plasma Physics Laboratory, a National Laboratory managed by Princeton University for the Department of Energy, this work was completed without employer sponsorship or involvement.

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