FUEL CYCLE PERFORMANCE OF FAST SPECTRUM MOLTEN SALT REACTOR DESIGNS

Andrei Rykhlevskii,* Benjamin R. Betzler,† Andrew Worrall,† and Kathryn Huff,*

*Dept. of Nuclear, Plasma, and Radiological Engineering, University of Illinois at Urbana-Champaign, Urbana, IL
andreir2@illinois.edu
†Oak Ridge National Laboratory, 1 Bethel Valley Road, Oak Ridge, TN, USA

INTRODUCTION

A liquid-fueled Molten Salt Reactor (MSR) concepts promise one of the most desirable and competitive, sustainable energy among of many advanced reactor systems [1]. In MSR fissile and/or fertile materials are dissolved in carrier molten salt (e.g., LiF, NaCl) which leads to immediate advantages over traditional, solid-fueled, reactors. These include near-atmospheric pressure in the primary loop, relatively high coolant temperature, outstanding neutron economy, a high level of inherent safety, reduced fuel preprocessing, and the ability to continuously remove fission products and add fissile and/or fertile elements without shutdown [2].

Historically, researchers focused on thermal spectrum MSR concepts with solid graphite moderator. Oak Ridge National Laboratory (ORNL) operated ≈8 MW_{th} Molten Salt Reactor Experiment (MSRE) pilot reactor from 1965 to 1969 to test approaches/materials, demonstrate fissile recycle (both ^{233}U and ^{235}U), and determine generic MSR operational characteristics [3]. Obtained experience plus promising breakthrough in reprocessing technology [4] chained ORNL attention to the simply configured, also graphite-moderated (i.e., thermal or epithermal), single-fluid Molten Salt Breeder Reactor (MSBR) by the end of the 1960's. The primary weakness of all one-fluid thermal MSR concepts is the fact that the hundreds tons (≈300 t for MSBR [5]) of expensive, radiologically contaminated, neutron-damaged graphite would have to be replaced every 4-10 years, which raises significant waste and economical issues.

In contrast, consistent with Generation IV International Forum (GIF) sustainability and safety goals [6], unmoderated (no graphite), one- or two-fluid (blanket equipped) fast spectrum MSR concepts eventually became the EUROATOM Consortium's "reference" MSR [7]. Other identified advantages of fast MSR systems comparing with conventional reactors include: (1) they can operate in breeding (CR[1]>1) [7, 8, 9], fertile-free transuranic waste (TRU) burning, or U/Th-supported TRU burning [10] regime; (2) they can potentially work in both $^{232}Th/^{233}U$ and $^{238}U/^{239}Pu$ fuel cycles (i.e., thermal MSR supports only Th/U); (3) they can be operated in ways that would generate very little long-lived TRU waste, and (4) they would use less natural resources (e.g., natural uranium, natural thorium) per unit energy generated. To quantitatively estimate benefits of these capabilities fuel cycle performance analysis for various fast spectrum MSR concepts is necessary.

Much of the analysis herein uses unit cell representations of four different fast MSR designs: (1) European Molten Salt Fast Reactor (MSFR) [7]; (2) Molten Chloride Salt Fast Reactor (MCSFR) [8]; (3) REBUS-3700 [9]; (4) Molten Salt Actinide Recycler and Transmuter (MOSART) [10]. Some of these designs are two-fluid (1, 2), operate in thorium fuel cycle (1,4), or use TRU as start-up fissile material. The concepts (2) and (3) use chloride salts[2] while (1) and (2) employ fluorides. This paper discusses the depletion simulation of different fast MSRs to identify fuel cycle performance for deployment of this reactor technology.

METHODS

To determine Fuel Cycle Performance parameters full-core and simplified unit cell models of four different fast spectrum MSRs designs were created (Fig. 1). Table I contains summary of principal data of these designs. Fuel cycle performance analysis required depletion simulation for each reactor concept over the whole system lifetime which in this work assumed 60 years[3]. Full-core 60-years depletion computation for MSR model required massive computing power. Computation time can be significantly reduced by performing depletion simulation for simplified unit cell representation instead of full-core one with sophisticated geometry.

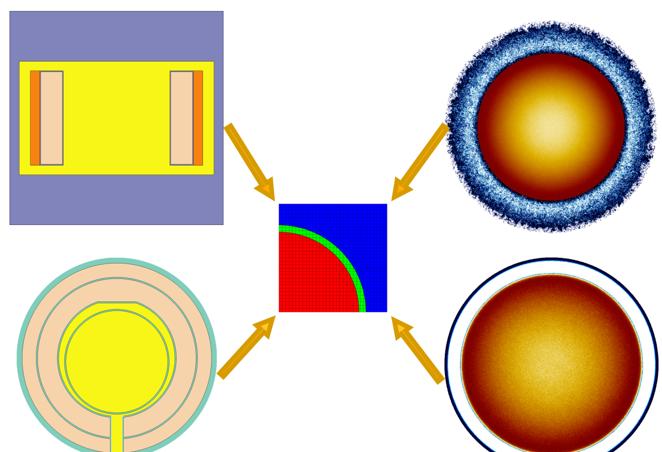

Fig. 1. Full-core 3D models of MSFR (upper left), MCSFR (lower left), REBUS-3700 (upper right), MOSART (lower right), and 2D unit cell model (center) showing fuel salt (red), fertile salt (green), structural material (blue).

[1] Conversion ratio (CR) ≡ fissile generated/fissile consumed: if CR<1 the reactor is a "converter"; CR≡1, an "isobreeder"; CR>1, a "breeder".

[2] The chlorine in the MCSFR is fully enriched in ^{37}Cl because ^{35}Cl (76% abundance) is a very strong neutron poison in fast neutron energy range.

[3] Lifetime for fast MSR limited by neutron damage on the reactor vessel. Further R&D activities should be performed to evaluate maximum fast neutron fluence and assess structural materials choice.

Models description

In contrast with thermal MSR, fast spectrum concepts do not have channel or assembly structure but contain homogenized fuel mixture in cylindrical or spherical vessel. Two-fluid systems also have cylindrical (MSFR) or spherical (MCSFR) blanket with fertile salt to reduce neutron leakage and enhance fissile material breeding (Fig. 1). Details about reactors' configuration can be found in Refs. [7, 8, 9, 11]. Two-fluid concepts 2D unit cell model contains a cylindrical fuel salt channel with thin outer layer of fertile salt inside square block of structural material (Hastelloy N). The unit cell model for single-fluid REBUS-3700 has fuel salt and structural material only; MOSART simplified model consist of fuel salt square block with graphite cylindrical channel in the center to represent 0.2 m graphite reflector which needed to increase ^{233}U breeding from thorium.

To prove viability of unit cell models for depletion simulation high-fidelity full-core models were developed using Monte Carlo code SERPENT2 (16 millions neutron histories per run) with ENDF/B-VII.1 library [12, 13]. Single average unit cell model geometry and size are optimized to obtain sufficiently accurate multiplication factor and neutron energy spectrum in a reasonable time. Next metrics are used for optimization:

1. eigenvalue discrepancy between full-core and unit cell models less than 300 pcm[4];
2. correlation coefficient (r) for neutron spectrum normalized by lethargy more than 0.995;
3. relative error (δ) of total neutron flux less than 3%.

The symmetry in a reactors is used to simplify the problem into one-quarter of the unit cell geometry. For this optimization the 16-by-16 spatial mesh for the NEWT neutron transport calculation is used in SCALE version 6.2.3 with the 238 group ENDF VII.1 cross-section library [14].

RESULTS

When importing figures or any graphical image please verify two things:

- Any number, text or symbol is in Times font and is not smaller than 10-point after reduction to the actual window in your paper
- That it can be translated into PDF

Equations, such as Eq. (1), should be centered and sequentially numbered to the flush right of the formula.

$$\text{Speedup} = \frac{1}{\frac{f}{p} + (1 - f)} \tag{1}$$

The continuation of a paragraph after an equation should not be indented. All paragraphs, as well as section or subsection headings, are separated by just one single empty line.

[4] 1 pcm = $10^{-5}\Delta k_{eff}/k_{eff}$

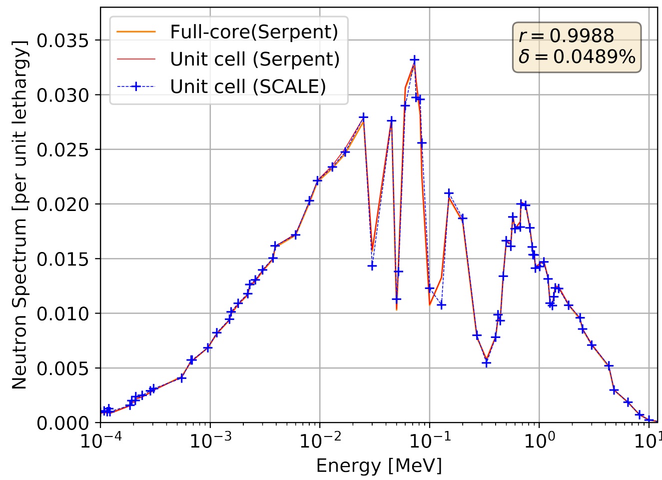

Fig. 2. Neutron flux energy spectrum for full-core and unit cell models for two-fluid MSFR.

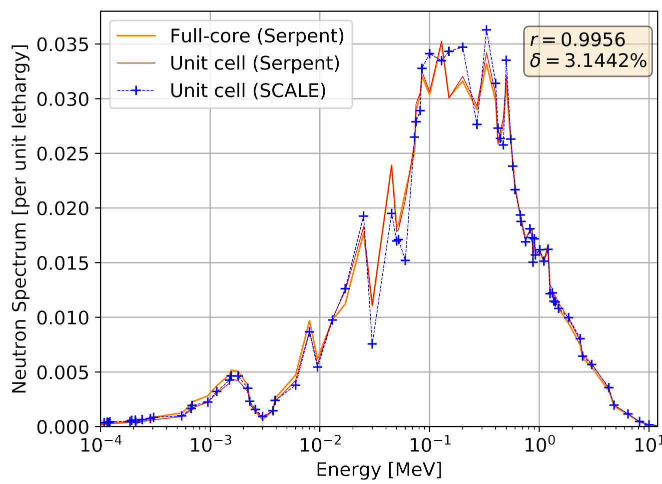

Fig. 3. Neutron flux energy spectrum for full-core and unit cell models for two-fluid MCSFR.

Sub-subsection level and lower: only first character uppercase

See Table II for a sample table. The "tabls" package is recommended for improved row and column spacing. Notice the caption appears above the table by setting the `\caption` command immediately after the `\begin{table}`. Tables are numbered in Roman numerals, with the caption centered above the table, in **boldface**. Triple-space before and after the table.

CONCLUSIONS

Present your summary and conclusions here.

ACKNOWLEDGMENTS

This research was supported by the Fuel Cycles Options Campaign of the Fuel Cycle Technologies initiative and by an appointment to the Oak Ridge National Laboratory Nuclear Engineering Science Laboratory Synthesis (NESLS) Program,

TABLE I. Principal data of selected fast spectrum MSRs designs.

Parameter	MSFR	MCSFR	REBUS-3700	MOSART
Thermal power, MW	3,000	6,000	3,700	2,400
Fuel salt volume (in/out of core)	18 (9/9)	38 (16/22)	55.6 (36.9/18.7)	49.05 (32.7/16.35)
Fertile salt volume (in/out of blanket)	7.3 (7.3/0)	75 (55/22)	—	—
Fuel and fertile salt initial composition (mol%)	LiF-ThF_4-$^{233}UF_4$ (77.5-19.9-2.6) LiF-ThF_4 (77.5-22.5)	NaCl-UCl_3-$^{239}PuCl_3$ (60-36-4) NaCl-UCl_3 (60-40)	55mol%NaCl+ 45mol%(natU+ 16.7at.%TRU)Cl_3	LiF-BeF_2-ThF_4-$TRUF_3$ (69.72-27-1.28)
Fuel cycle	Th/^{233}U	U/Pu	U/TRU	Th/^{233}U
Initial fissile inventory, t	5.060	9.400	18.061	9.637

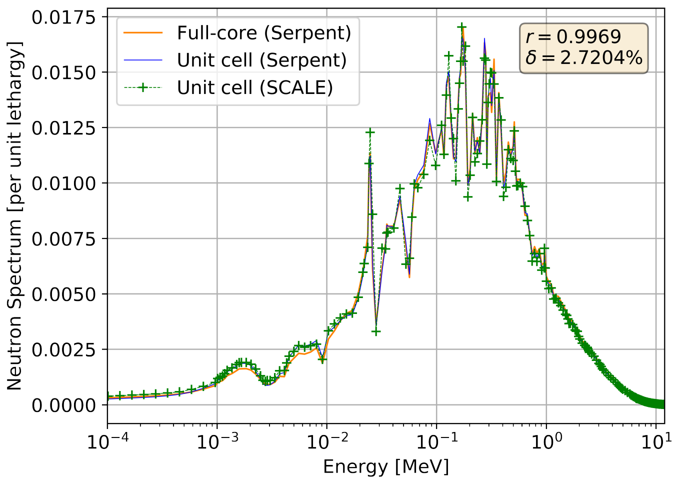

Fig. 4. Neutron flux energy spectrum for full-core and unit cell models for single-fluid REBUS-3700.

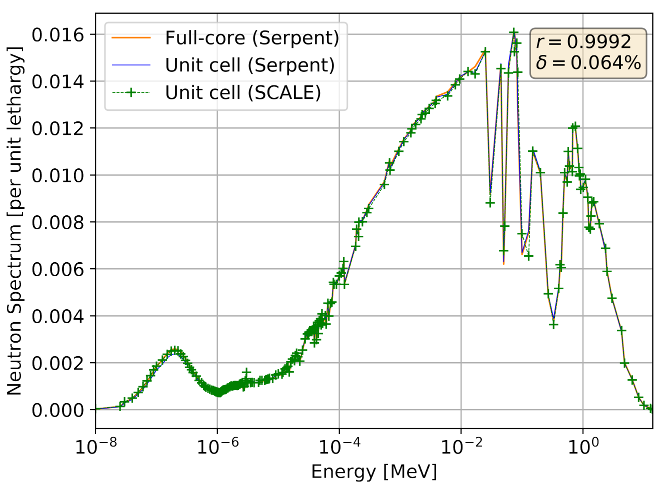

Fig. 5. Neutron flux energy spectrum for full-core and unit cell models for single-fluid MOSART with graphite reflector.

sponsored by U.S. Department of Energy and administered by

TABLE II. **Parallel Performance for the Sample Problem**

Parameter			
MSFR	(T_s/T_p)	(%)	
1	100.0	—	—
2	52.6	1.9	95.0

the Oak Ridge Institute for Science and Education.

REFERENCES

1. D. D. SIEMER, "Why the molten salt fast reactor (MSFR) is the âĂIJbestâĂİ Gen IV reactor," *Energy Science & Engineering*, **3**, *2*, 83–97 (2015).
2. D. LEBLANC, "Molten salt reactors: A new beginning for an old idea," *Nuclear Engineering and Design*, **240**, *6*, 1644–1656 (Jun. 2010).
3. H. G. MACPHERSON, "The Molten Salt Reactor Adventure," *Nuclear Science and Engineering*, **90**, *4*, 374–380 (Aug. 1985).
4. M. E. WHATLEY, L. E. MCNEESE, W. L. CARTER, L. M. FERRIS, and E. L. NICHOLSON, "Engineering development of the MSBR fuel recycle," *Nuclear Applications and Technology*, **8**, *2*, 170–178 (1970).
5. R. C. ROBERTSON, "Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor." Tech. Rep. ORNL–4541, comp.; Oak Ridge National Lab., Tenn. (Jan. 1971).
6. GIF, "Generation IV International Forum 2015 Annual Report," Tech. rep., Generation IV International Forum (2015).
7. "Final Report Summary - EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor System) | Report Summary | EVOL | FP7| European Commission," Final report 249696, France (2015).
8. W. E. SIMMONS and J. SMITH, "AN ASSESSMENT OF A 2500 MWe MOLTEN CHLORIDE SALT FAST REACTOR," Tech. Rep. AEEW-R956, United Kindom Atomic Energy Authority (Aug. 1974).
9. A. MOUROGOV and P. M. BOKOV, "Potentialities of the fast spectrum molten salt reactor concept: REBUS-3700," *Energy Conversion and Management*, **47**, *17*, 2761–2771 (Oct. 2006).

10. V. IGNATIEV, O. FEYNBERG, I. GNIDOI, A. MERZLYAKOV, V. SMIRNOV, A. SURENKOV, I. TRETIAKOV, R. ZAKIROV, V. AFONICHKIN, A. BOVET, V. SUBBOTIN, A. PANOV, A. TOROPOV, and A. ZHEREBTSOV, “Progress in development of Li,Be,Na/F molten salt actinide recycler and transmuter concept,” in “Proceedings of ICAPP 2007,” (May 2007).
11. V. IGNATIEV, O. FEYNBERG, I. GNIDOI, A. MERZLYAKOV, A. SURENKOV, V. UGLOV, A. ZAGNITKO, V. SUBBOTIN, I. SANNIKOV, A. TOROPOV, V. AFONICHKIN, A. BOVET, V. KHOKHLOV, V. SHISHKIN, M. KORMILITSYN, A. LIZIN, and A. OSIPENKO, “Molten salt actinide recycler and transforming system without and with ThâĂŞU support: Fuel cycle flexibility and key material properties,” *Annals of Nuclear Energy*, **64**, *Supplement C*, 408–420 (Feb. 2014).
12. J. LEPPANEN, M. PUSA, T. VIITANEN, V. VALTAVIRTA, and T. KALTIAISENAHO, “The Serpent Monte Carlo code: Status, development and applications in 2013,” *Annals of Nuclear Energy*, **82**, 142–150 (Aug. 2015).
13. M. B. CHADWICK, M. HERMAN, P. OBLOÅ¿INSKÃ¡, M. E. DUNN, Y. DANON, A. C. KAHLER, D. L. SMITH, B. PRITYCHENKO, G. ARBANAS, R. ARCILLA, R. BREWER, D. A. BROWN, R. CAPOTE, A. D. CARLSON, Y. S. CHO, H. DERRIEN, K. GUBER, G. M. HALE, S. HOBLIT, S. HOLLOWAY, T. D. JOHNSON, T. KAWANO, B. C. KIEDROWSKI, H. KIM, S. KUNIEDA, N. M. LARSON, L. LEAL, J. P. LESTONE, R. C. LITTLE, E. A. MCCUTCHAN, R. E. MACFARLANE, M. MACINNES, C. M. MATTOON, R. D. MCKNIGHT, S. F. MUGHABGHAB, G. P. A. NOBRE, G. PALMIOTTI, A. PALUMBO, M. T. PIGNI, V. G. PRONYAEV, R. O. SAYER, A. A. SONZOGNI, N. C. SUMMERS, P. TALOU, I. J. THOMPSON, A. TRKOV, R. L. VOGT, S. C. VAN DER MARCK, A. WALLNER, M. C. WHITE, D. WIARDA, and P. G. YOUNG, “ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data,” *Nuclear Data Sheets*, **112**, *12*, 2887–2996 (Dec. 2011).
14. B. T. REARDEN and M. A. JESSEE, “SCALE Code System,” Tech. Rep. ORNL/TM-2005/39, Version 6.2.1, Oak Ridge National Lab.(ORNL), Oak Ridge, TN (United States) (2016).