A Novel Stationary Molten Salt Reactor Design for Spent Nuclear Fuel Burning

Justin Shurie, Robert Krushelnisky, Caleb King, William Strom, and Zeyun Wu

Department of Mechanical and Nuclear Engineering, Virginia Commonwealth University,
401 W Main Street, Richmond, Virginia 23284
shuriej@vcu.edu, krushelniskrl@vcu.edu, kingct@vcu.edu, stromwe@vcu.edu, zwu@vcu.edu

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INTRODUCTION

Molten salt reactors (MSRs) are a type of reactor that takes fissile material and dissolves it into a molten salt to allow the fuel to maintain a liquid state during the reactor normal operation. MSRs carry many advantages over typical light water reactors in use today such as low operating pressures and accident resistance. Conventional MSRs like the Molten Salt Reactor Experiment at Oak Ridge National Lab use circulating fuel designs where the fuel salts move onto a critical geometry to generate heat, then circulate out of the core to pass through heat exchangers. This introduces the problem of having the fuel and associated waste products moving outside of the reactor core, elevating the severity of potential failures in the primary loop. Stationary Salt Reactors (SSRs), such as the SSR-W being developed by Moltex Energy LLC [1], restrain the fuel salts within the reactor core to alleviate this concern.

The Moltex SSR-W features a static molten salt design with the fuel confined to fuel tubes within square assemblies. The fuel salt mixture contains new fuel salts and spent nuclear fuel which can be burned by the excess fluence within the reactor. It utilizes a zirconium cladding which protects the structure of the fuel tubes from corrosion due to fission byproducts dissolved in the salt. The zirconium cladding bonds with and traps the harmful redox chemicals, the newly bonded byproducts then detach from the wall of the tube and settle in the bottom section of the core [1]. A research group at Virginia Commonwealth University (VCU) Computational Applied Reactor Physics Laboratory (CARPL), under the direction of Dr. Zeyun Wu, has investigated an inverted stationary salt reactor design. This design allows for a mostly homogenous fuel pool by keeping the fuel within a single tank with coolant tubes running vertically through the reactor core. Preliminary neutronics and thermal-hydraulic analysis have shown the viability of this reactor design [2].

This senior design project attempts to take the best of these and other designs to generate a compact SSR with the purpose of burning long-lived actinides present in spent nuclear fuel. These actinides would then be reduced to fission products with much shorter half-lives, reducing concerns for nuclear proliferation and waste storage. This is intended to help bridge the gap in the fuel cycle that is present in the United States and make nuclear energy more appealing to the general populace by alleviating public concerns over long lived nuclear wastes.

REACTOR DESIGN OVERVIEW

The new design follows a similar concept to the Inverted SSR design mentioned earlier with the exception that the coolant channels will be of a U-tube design rather than the once through pattern where coolant flows from one end of the core to the other. As an inverted reactor, coolant flows through tubes that are surrounded by a pool of fuel salt. Fig. 1 presents a reactor diagram illustrating the main components and coolant flow directions in the core.

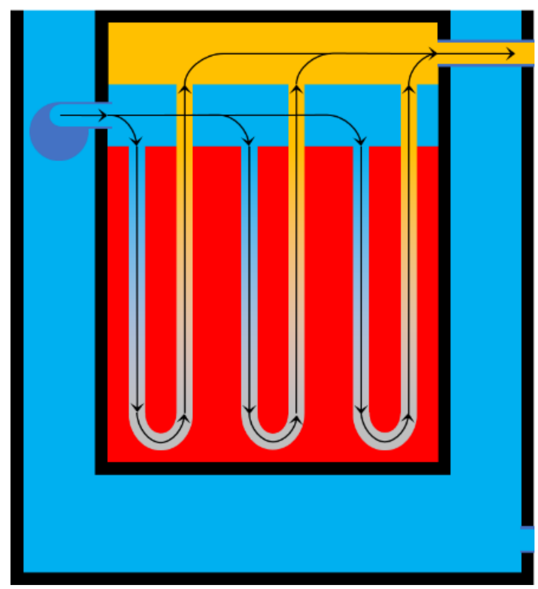

Fig. 1. A reactor diagram illustrating coolant flows (not to scale). Blue to orange gradient represents the cold to hot gradient of the FLiBe coolant, and red is fuel.

As shown in the figure, the entire reactor will sit inside a coolant pool and draw the FLiBe coolant salt from the pool, passing it through a plenum where the coolant salt will be driven into the tube inlets. The coolant will then pass through the fuel salt pool contained within the vessel and back up to the second layer of the plenum, from where the coolant salt would be circulated through a heat exchanger before being returned to the pool. The fuel pool has a pocket of argon gas to fill the gap between the pool and the plenum and prevent interaction with the salts. The cladding material selected for the tubes along with the vessel body is Hastelloy-N as it does not significantly interact with either the fluoride-based salts in the coolant nor the chloride salts in the fuel. Continuous online refueling allows for fresh and spent fuel salts to be added as needed, removing, and separating spent nuclides.

COMPUTER MODELS & RESULTS

To examine the viability and physics feasibility, computational models are continuously being developed and refined to enable component and systematic analysis of the novel design.

CAD Model

The entire core was first modelled in SOLIDWORKS in five separate parts before being joined into a single assembly, as shown in Fig. 2. The whole core CAD model can be used directly in the thermal-hydraulics evaluation and additionally as a template for the basic geometry of the neutronics model. Preliminary core characteristics are shown in Table I.

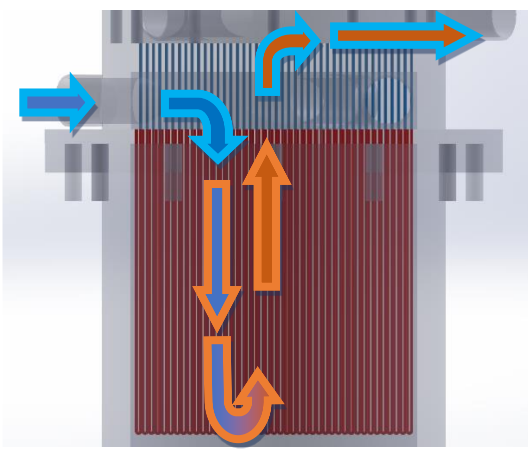

Fig. 2. Side view of the SOLIDWORKS model with flow path arrows.

Table I. Preliminary Core Design Characteristics.

Parameter	Value
Thermal Power	300 MW
Effective Fuel Height	3.75 m
Core Diameter	4.5 m
Inlet temperature	650 °C
Outlet Temperature	750 °C
Mass Flow Rate	1257.0 kg/s
Coolant Tube Diameter	1.0 cm
Coolant Tube Pitch	1.4 cm
Number of U-tubes	30922
Cladding Thickness	1.0 mm

Neutronics model

A full core neutronics model was developed in the Monte-Carlo N-Particle (MCNP) radiation transport code version 6.2 [3]. The MCNP model can be used to determine various important quantities that both indicate reactor physics feasibility and/or can be used to provide heat source for thermal-hydraulics calculations. The effective neutron multiplication factor (i.e., k_{eff}) is the most important quantity in neutronics aspect, as a value slightly above unity is desired for reactors. A neutron flux energy spectrum can also be determined from flux tallies in multiple energy bins, and the neutron flux, reaction rate density, and mean free path can also be determined on average or locally; reaction rate density can be tied back into thermohydraulic calculations as being proportional to the volumetric heat generation rate.

The reactor model geometry consists of a repeated hexagonal lattice constrained by the reactor vessel and extending the height of the reactor. Each element of the lattice has a side length equal to the coolant tube pitch. The corners of each element hold an alternating pattern of hot and cold tubes that, when meshed, form the array of tubes used for the coolant loop. These hot and cold tubes are joined together at the bottom of the reactor by a U-bend section just above the bottom of the reactor fuel pool.

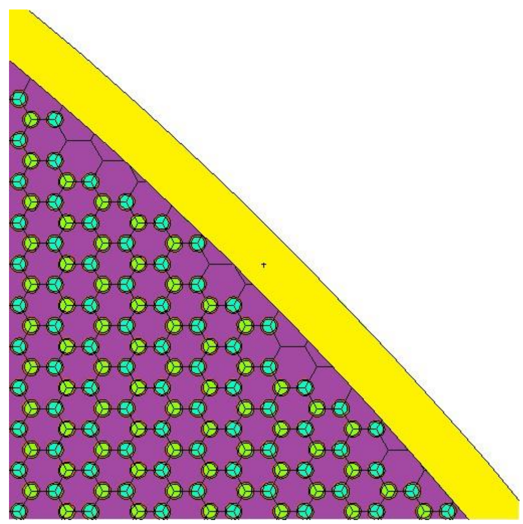

Fig. 3. The MCNP model of one core region showing the radial layout of the different materials in the core with Hastelloy-N (orange clad or yellow vessel), hot and cold FLiBe (light green and blue respectively) and fuel (purple).

Fig. 3 presents a detailed radial view of one core region established by MCNP. The fuel and coolant portions of the lattice were divided vertically into 9 equal segments to more accurately follow the temperature fluctuations that would be present in the core. The model uses four materials in the calculation, Hastelloy-N, Argon, FLiBe, and the fuel mixture of $NaCl$-UCl_3-$PuCl_3$. These materials are modified according to the temperature region the material occupies. The temperature of the 9 fuel regions and the 19 coolant regions can be altered along with all other reactor parameters easily for the sake of design iteration and optimization when coupled with the thermohydraulic model. The MCNP input deck is generated in a MS Excel [4] spreadsheet, every parameter of the reactor corresponding to a cell, allowing for the implementation of necessary changes with minimal time and with reduced error potential.

Multiple fuel compositions were explored to measure how the composition might affect the reactivity and what would be needed to maintain criticality, the selected fuel composition being shown in Table II. The current fuel composition yields a k_{eff} of 1.05037 verifying that the core will be able to maintain operation at the beginning of cycle. The in-core average neutron flux spectrum was also calculated using Scale 252 energy groups [5]. The resulting energy spectrum is a soft mixture of intermediate and fast neutron energies as shown in Fig. 4. This is ideal as the fast neutron spectrum is desired for waste burning purposes.

Table II. Selected Fuel Composition.

Element	Z	A	weight (wt.%)	atom (at.%)
Na	11	23	1.30521E-01	3.00000E-01
Cl	17	35	1.98619E-02	3.00000E-02
		37	3.98941E-01	5.70000E-01
U	92	234	5.53648E-05	1.25080E-05
		235	1.90247E-03	4.27975E-04
		236	1.98793E-03	4.45305E-04
		238	3.42221E-01	7.60149E-02
Np	93	237	2.25918E-04	5.03931E-05
Pu	94	238	1.59035E-04	3.53251E-05
		239	8.66857E-02	1.91742E-02
		240	1.61780E-02	3.56355E-03
		241	4.83049E-04	1.05960E-04
		242	4.34744E-04	9.49700E-05
Am	95	241	1.95077E-04	4.27916E-05
		242	5.83374E-07	1.27438E-07
		243	9.25224E-05	2.01284E-05
Cm	96	242	1.60892E-09	3.51470E-10
		243	2.87971E-07	6.26486E-08
		244	4.94196E-05	1.07072E-05
		245	4.94196E-06	1.06635E-06

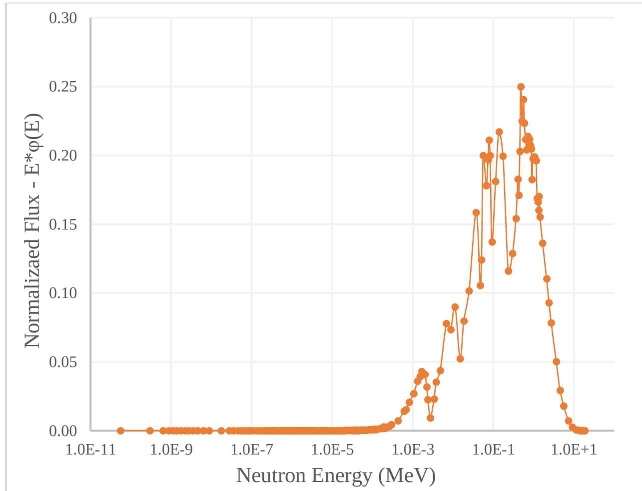

Fig. 4. In-core average neutron flux spectrum of the U-tube Stationary Salt Reactor.

The fuel U/Pu ratio is approximately 3.35, with the fresh plutonium fuel being enriched to 85 at.% Pu-239. Salt within the fuel pool is 60 at.% NaCl, 40% trichloride fuel salt. The trichloride fuel salt is composed of 75 at.% chlorine, 5.125 at.% fresh plutonium, 19.875 at.% spent LWR fuel. Chlorine is enriched to 95 at.% Cl-37.

Thermal-hydraulics model

The thermal-hydraulics were handled with ANSYS Discovery and Fluent. The focus of this analysis was to determine a stable single channel analysis which would provide high fidelity estimates of fuel, cladding and coolant temperatures. The fuel was modelled as a static fluid and does not currently evaluate natural convection within the core. The model itself consists of 397468 nodes with 1.24 million elements as this is the maximum that the student version of ANSYS Discovery will allow. The single channel consists of a rhomboid shape encompassing 4 adjacent U-tube elements. This was chosen to give the most accurate representation of the temperature distribution around the coolant channels. The boundary conditions for fluid flow within the channels consisted of no-slip flow along the walls of the tubes, an inlet mass flow rate of 0.033 kg/s, and an inlet temperature of 650 °C (923 K). The walls of the tube were defined as convection/conduction boundaries with a convection coefficient of 138.8 W/m^2 °C. A horizontal cross section of the ANSYS heat map is shown in Fig. 4.

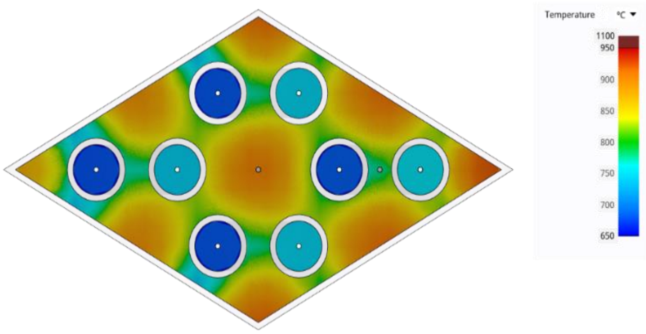

Fig. 4. Color map of the horizontal cross section of the single channel temperature model.

CONCLUSIONS AND FUTURE WORK

This work presents an innovative stationary molten salt fast reactor design with the primary purpose of burning long lived radioactive nuclides found in spent nuclear fuel. The reactor design is selected be a small modular reactor capable of producing 300 MWth. A coolant flow configuration with U-tube shaped coolant channels is proposed and investigated using CAD, neutronics and thermal-hydraulic analysis tools. Different computational models are being developed by these models to take into account of various physics considerations. A preliminary analysis embracing MCNP neutronics model and computation fluid dynamics (CFD) thermal-hydraulic model has being performed to justify the viability and physics feasibility of the design.

Many future developments are needed to optimize both models and interconnect their results to the opposing model's input. The current neutronics model has only been analyzed to the extent that the reactor is confirmed to be slightly supercritical at initial steady state operating conditions (ignoring burnup of the initial fresh and used fuel), and a brief confirmation of the neutron energy flux spectrum containing a significant number of fast neutrons. The current thermal-hydraulic model employs a uniform heat generation and only encompasses coolant tubes, not accounting for natural convection within the fuel salt, while also not including axial and radial differences in volumetric heat generation rate.

In the neutronics model, the temperature of various core regions should be determined based on the distributions calculated by the ANSYS Fluent model. Additionally, the flux in various regions of the reactor determined by MCNP should be used as a proportionality constant when determining the magnitude of the volumetric heat flux in the Ansys model. The result of these two changes is that the neutronic and thermodynamic models are linked together, allowing for iteration of both models until additional iterations cause negligible changes in either flux or temperature. Perhaps the most important consideration of the whole core model is ensuring that the coolant flows somewhat evenly throughout all the coolant channels. As coolant is injected into the sides of the plenum and peak heat generation is expected in the center of the reactor, a whole core model is necessary to ensure the central coolant channels receive enough coolant to remove the generated heat.

The thermal-hydraulic model, currently 4 u-tubes with a constant volumetric heat generation rate throughout the fuel, should not only account for natural convection within the fuel, but also for different heat generation rates at different locations. Initially, this will be implemented in the 4 u-tube model by applying a radial peaking factor to the axial flux profile, but a later whole core model is planned to better match the neutronics model which already encompasses the whole core. Additionally, neither the MCNP model nor the thermohydraulic model accounts for the larger coolant pool surrounding the reactor vessel, something that is actively being worked on.

Burnup calculations in MCNP will be used to monitor the actinide inventory of the reactor, allowing for the effectiveness of the reactor as a waste burner to be analyzed at multiple levels of burnup. This process will additionally be used to ensure that the reactor remains critical at these various levels of burnup. Failure to burn waste (likely due to poor neutron economy or a disadvantageous neutron energy spectrum) as well as failure to maintain criticality, should they occur, can be changed significantly by adjusting the ratio of fresh and used fuel in the reactor initial loading. Online refueling and its implications on the reactor fuel inventory are also being explored. Further, reactivity control is non-existent within the present MCNP model, not even leaving spaces for control rods to be inserted for the purposes of shutdown or power control. Shutdown will require these control rods, although reactivity control is planned to be provided by thermal feedback coefficients, a steady state k_{eff} of 1.05 being sought.

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