Coupled Neutronics and Thermal Hydraulics Calculations for the MSRE Pump Startup and Coastdown Transients

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INTRODUCTION

Molten Salt Reactor (MSR) systems are among the selected technologies for Gen IV nuclear reactors [1]. MSR employs a molten salt mixture as both fuel and coolant. This configuration gives the system operation and safety advantages over water and other types of reactors including online refueling, processing, and fission product removal; high coolant outlet temperature; low operating pressure; and inherent safety characteristics [2]. The flowing nature of MSR fuel is challenging for the current computational tools due to Delayed Neutron Precursors (DNP) and fission products drift. Due to these unique features, high-quality data is required for code development and validation. The Molten Salt Reactor Experiment (MSRE) performed in the 1960s [3] is the only experimental data source for this class of reactors. During the operational period of the MSRE several transient experiments were conducted.

In this work, the fuel pump startup and coastdown experiments are modeled with a simplified one-dimensional (1D) model for the neutronics and thermal hydraulics coupled evaluations. The aim of this work is to develop an accurate and computationally efficient Multiphysics simulation platform for the MSRE system. The MSRE fuel pump startup and coastdown tests are well documented in Ref. [4] and used for code validation.

COMPUTATIONAL MODELS

To capture the strong coupling between neutronics and thermal hydraulics in MSRE design, a fully coupled system of equations is considered. The 1D two group (2G) diffusion model is used for describing the neutron flux

$$\frac{1}{v_g}\frac{\partial \varphi_g}{\partial t} - \frac{\partial}{\partial z} D_z \frac{\partial \varphi_g}{\partial z} + \Sigma_{r,g}\varphi_g = Q_g, \tag{1}$$

where all the terms are in the standard notations, and

$$Q_1 = (1-\beta)\sum_g \upsilon\Sigma_{f,g}\varphi_g + \sum_k \lambda_k C_k + \Sigma_{s,2\to1}\varphi_2 \,,$$
$$Q_2 = \Sigma_{s,1\to2}\varphi_1 \,. \tag{2}$$

The DNP concentrations are described by the 1D drift equation

$$A\frac{\partial C_k}{\partial t} + Au\nabla C_k = -A\lambda_k C_k + \beta_k A \sum_g \upsilon\Sigma_{f,g}\varphi_g, \tag{3}$$

where A is the flow area and u is the 1D flow speed. The flow speed is approximated by the incompressible continuity and momentum equations [5]:

$$\nabla(Au) = 0$$
$$\rho\frac{\partial u}{\partial t} + \rho u.\nabla u = -\nabla p - \nabla p_{friction}, \tag{4}$$

where ρ and p are the density and pressure respectively. The model presentation is limited to the flow equations as the scope of this paper is limited to the isothermal tests.

The neutron diffusion equation is solved for the reactor core and the albedo boundary condition is constructed and applied to account for and the axial neutron leakages. A fictitious leakage cross section is used to account for the radial flux. Six DNP families are used. The cross sections, albedo factors, and DNP data are generated using the MSRE Serpent model developed in Ref. [6].

Equations (3) and (4) are solved for the entire fuel circulation loop. The pressure at the pump intake is fixed to the initial pressure value. The thermophysical properties of the fuel salt and the geometry of the fuel circulation loop are collected from ORNL legacy reports [3, 4] and their recent publication [7] aimed to predict the steady state DNP concentration along the entire primary loop.

All the equations are solved using COMSOL Multiphysics software [8] for this work. The diffusion equation is implemented in the mathematics module while the DNP equations and the fluid flow is implemented in the Reacting Pipe Flow interface. A fully coupling between the two components is achieved by exchanging the fission source and the delayed neutron source between the two components.

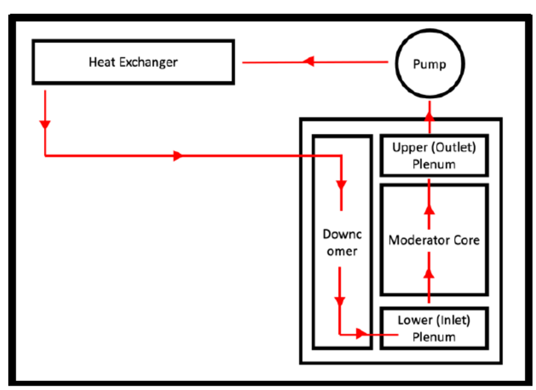

Fig. 1. A schematic view of the primary loop of MSRE.

MSRE SYSTEM

The MSRE fuel circulation loop, depicted in Fig. 1, consists of the reactor vessel, fuel pump, primary heat exchanger, and piping system. The fuel salt enters the cylindrical reactor vessel through an annular volute around the top of the cylinder and flows downwards between the vessel and the graphite matrix which is designed to occupy 77.5% of the vessel volume. A dished head at the bottom forces the flow in the upward direction through rectangular passages in the graphite matrix to the top head. The fuel then flows the suction line of the primary pump and then discharge to the shell side of a U-tube heat exchanger in which a secondary fluoride melt (LIF-BeF_2, 66-34 mole %) is used to cool the fuel salt. Fuel pump is sump-type centrifugal pump rotates at 1160 rpm delivers 1200 gpm at 49 ft of fluid [9]. The heat exchanger (HX) is designed for heat load of 10 MW following the configuration of conventional 25%-cut, baffled shell-and-tube units. The pipe size is 5 inch, and the flow speed at 1200 gpm is 20 ft/s. The total volume of the fuel salt in the primary loop is 73 ft^3. The lengths and flow areas of the 1D representation of the fuel circulation loop components is calculated to preserve the fuel salt volume in each component.

The pump startup and pump coastdown tests are conducted at zero power (~10 W). The power was kept constant during the transients by adjusting the fuel rod position inside the core. During the pump startup test, the pump speed was increased linearly from zero to 100% in 1s then it was kept constant during the transient [10]. In the pump coat down test, starting from steady state flowing conditions, the pump motor was turned off. Unfortunately, the fuel salt flow rate in neither case was documented. Furthermore, the fuel pump curve data is not available. The change in the control rod position during the pump tests is shown in Fig. 2.

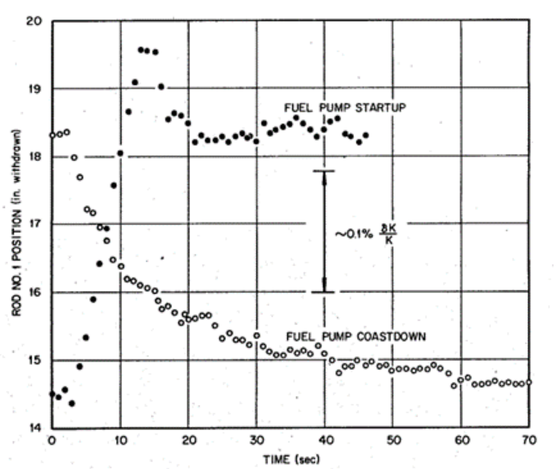

Fig. 2. Control-Rod Response to Fuel Pump Startup and Coastdown (Original Fig. 24. in Ref [4]).

To model the pump startup, for simplicity, we assumed that the flow rate is proportional to the pump speed. Thus, a prescribed pump flow rate was assumed in the form of

$$Q = \begin{cases} Q_{op}t, & 0 \le t \le 1 \\ Q_{op}, & t > 1 \end{cases}, \tag{5}$$

where $Q_{op} = 1200\,gpm$ is the operational flow rate. The initial conditions for the test are $u = 0,\ p = 5\,psi$. The initial flux was obtained by solving the steady eigenvalue diffusion model and scaling the flux to the test power (10 W).

To model the pump coastdown, we assumed that the pump head was decreased exponentially with time constant of 1 s^{-1}. This assumption is based on pump motor stop time, which takes *10 s* to completely stop after the power supply interruption [3]. This gives the pump heat the functional form

$$\Delta p_{pump} = \Delta p_0 e^{-t/\tau}, \tag{6}$$

where Δp_0 is the pump head required to sustain the steady state flow rate in the pump startup transient, and τ is the time constant. The initial conditions for this test are obtained from the steady state of the previous test.

The power was kept constant during the simulation by scaling the fission cross section to keep the fission rate at constant level which is needed to give the test power. This scaling factor is a measure of the reactivity required to sustain the reactor criticality.

RESULTS

For the neutronics and T/H coupled calculations, the fuel salt flow rate is assumed to be known during the pump startup and coastdown tests. The fuel flow rate as a function of time for both transient events are illustrated in Fig. 3.

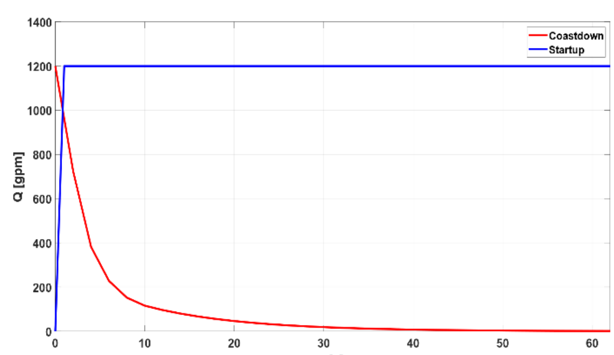

Fig. 3. Fuel salt flow rate in the pump startup and coastdown tests.

The reactivity of the system is required to be adjusted to maintain the critical status during both transient periods. However, the k-eff value yielded from the steady state neutronics model at the initial time is not unity. Thus we assumed its value to be k_0, and used it as a scaling factor to estimate the dynamic reactivity along the transient as

$$\rho(t) = \frac{k(t) - k_0}{k(t)} \ , \qquad (7)$$

where $k(t)$ is considered as a factor used to scale the fission source term to keep the power constant at time step t and k_0 is the scaling factor at $t = 0$. The reactivity insertion based on Eq.(7) and the average concentrations of the DNP as a function of time are presented in Fig. 4 for the pump startup test and in Fig. 5 for the pump coastdown test.

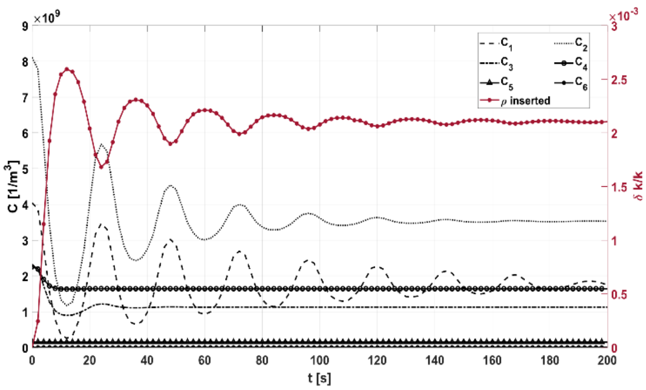

Fig. 4. Reactivity insertion and average DNP concentrations for the pump startup test.

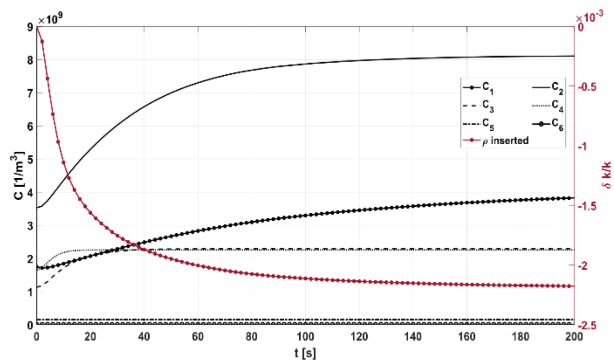

Fig. 5. Reactivity insertion and average DNP concentrations for the pump coast test.

Based on the results shown in Fig. 4 and 5, it is evident that the reactivity insertion is synchronized with the change of DNP concentration inside the core. Although a numerical comparison between the calculated and measured reactivity is not provided here, the following conclusions can be drawn from the time evolution:

- The total reactivity loss due to fuel circulation calculated from the steady state of the pump startup test is $\sim 211\ pcm$, and the pump coastdown test is $\sim 214\ pcm$ are in great agreement with the experimentally measured value $\sim 212\ pcm$ [10]
- The calculated reactivity for the startup test in the initial phase $(t < 15\ s)$ is shaper than the measured values. This indicates that the assumption made about the flow rate is not descriptive of the MSRE pump flow rate.
- Similarly, the calculated reactivity for the coastdown test is sharper than the measured values, indicating that the assumption about the pump head change after power interruption is not descriptive of the MSRE pump.

CONCLUSIONS

In this work, a simplified neutronics and T/H coupled model of the MSRE fuel circulation loop was developed. The model was employed to analyze the fuel pump startup and coastdown transients and was demonstrated to successfully capture the main characteristics of the reactivity changes due to fuel circulation. In the future work, the reactivity measurements will be used to infer the fuel salt flow rate during both transients. It is also planned to extend the model capability for non-isothermal and compressible flow conditions.

ACKNOWLEDGEMENTS

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REFERENCES

1. Gen IV International Forum - GIF, 2010 Annual Report. Nuclear Energy Agency of the OECD, (2010).
2. T. J. DOLAN, *Molten Salt Reactors and Thorium Energy*, Woodhead Publishing (2017).
3. R. C. ROBERTSON, "MSRE Design and Operations Report Part 1: Description of Reactor Design," ORNL-TM-728, ORNL (1965).
4. B.E. PRINCE et al., "Zero-power Physics Experiments on the Molten-Salt Reactor Experiment," ORNL-4233, ORNL (1968).
5. N. E. TODREAS and M. S. KAZIMI, *Nuclear Systems. Volume 1 - Thermal Hydraulic Fundamentals*. 2nd Ed. London, Taylor & Francis (2011).
6. D. SHEN et al., "Reactor Physics Benchmark of the First Criticality in the Molten Salt Reactor Experiment," *Nucl. Sci. Eng.*, **195**(8) (2021).
7. K. LEE et al., "Transient Convective Delayed Neutron Precursors of 235U for the Molten Salt Reactor Experiment," *PHYSOR 2022 proceedings*, (2022).
8. COMSOL Multiphysics Reference Manual (2014).
9. R. B. BRIGGS, Molten-Salt Reactor Program Semi-annual Progress Report, ORNL, Oak Ridge, Tennessee, ORNL-3708 (1964).
10. R. B. BRIGGS, Molten-Salt Reactor Program Semi-annual Progress Report, ORNL, Oak Ridge, Tennessee, ORNL-3872 (1965).