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Multiphysics Analysis of the MSRE Experiment Using Griffin-SAM Coupled Code System

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INTRODUCTION

Molten salt reactors (MSRs) with flowing fuel have the unique feature of utilizing the fuel salt for heat generation and extraction at the same time since the fuel salt is circulating through the whole primary loop of the reactor. This movement of the fuel salt results in a partial decay of the delayed neutron precursors (DNPs) outside the core and corresponding redistribution in the active core region. To capture this phenomenon accurately, neutronics and thermal hydraulics computational tools need to be modified to handle the movement of the DNPs and their decay [1, 2, 3, 4].

Idaho and Argonne National Laboratories are actively working on developing neutronics and thermal hydraulics tools to model MSRs with flowing fuel. The neutronics code Griffin [5] and the thermal hydraulics code SAM [6] were extended to handle flowing fuel with the drift of the DNPs under the MOOSE framework [7]. In this work, a simplified multiphysics model of the Molten Salt Reactor Experiment (MSRE) is developed and utilized to perform steady-state and transient analyses. The MSRE experiments were designed to show the impact of the DNPs losses on core reactivity. The following section provides a description of the MSRE experiment with the core main parameters used to develop the model. Then, the developed multiphysics model of the MSRE is presented. Finally, steady-state verification tests and unprotected loss of flow transient tests are discussed followed by a summary and conclusions.

MSRE DESCRIPTION

The MSRE was built in 1964 at Oak Ridge National Laboratory (ORNL) to be the first reactor designed and operated with liquid fuel and moderated with graphite. It utilized a thermal neutron spectrum by allowing the liquid fuel salt to flow into graphite moderator channels. The reactor was initially operated with U-235 fuel, which was later replaced with U-233 fuel. The MSRE reactor assembly is shown in Fig. 1, and Table I provides the reactor's main characteristics [1, 2].

The MSRE lattice is made of vertical graphite stringers with a 5.08 cm x 5.08 cm cross section. The fuel salt flows through a rectangular channel (3.05 cm x 1.016 cm with round corners of radius 0.508 cm) on the sides of the stringers. The liquid fuel salt is composed of $LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$. The thermophysical properties of the fuel salt and solid moderator are provided in Table II. These data were collected from several design and analysis reports of the MSRE [8].

TABLE I. MSRE core specifications [1, 2].

Parameter	Value
Core power [MWth]	10.0
Core Height [m]	1.39
Core Diameter [m]	1.63
Fuel Salt Composition	$LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$
Fuel Salt Molar Mass [%]	65.0-29.1-5.0-0.9
Fuel Enrichment [%]	33.0
Core Inlet temperature [K]	908.0
Core outlet temperature [K]	936.0
Fuel Circulation Time [s]	25.2

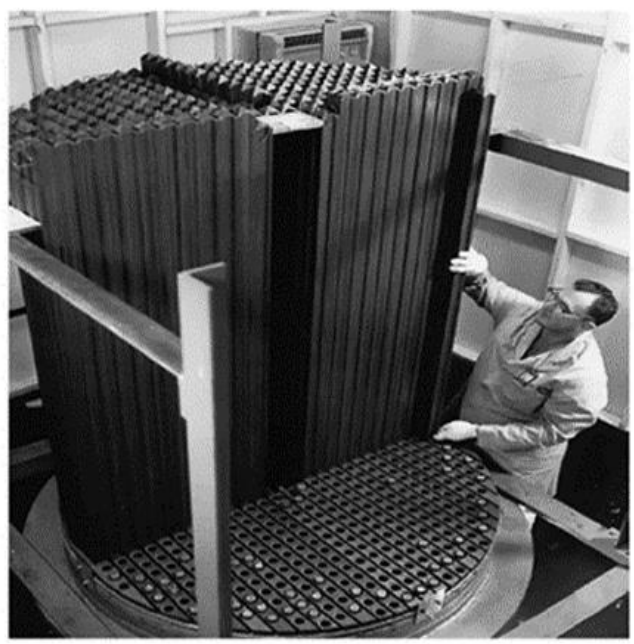

Fig. 1. MSRE reactor assembly [8].

TABLE II. Fuel Salt and Graphite Thermophysical Properties of the MSRE [1, 2].

Parameter	Fuel	Graphite
Density [kg/m^3]	2263.5	1860.0
Thermal Conductivity [W/m·K]	1.4	40.1
Specific Heat [J/kg·K]	1868.0	1757.3
Dynamic Viscosity [Pa·s]	0.263371	-

ANALYSIS METHOD AND MODEL DESCRIPTION

In this section, the developed models and multiphysics coupled system of the MSRE experiment are described. Griffin code (MOOSE-based neutronics application) is used to provide the neutronics solution with capabilities of solving the DNP equations with drift term given the core velocity field with outer loop residence time. While SAM code (MOOSE-based system analysis code) solves the porous medium for fluid flow equations with conjugate heat transfer and heat conduction through a solid medium [5]. Also, SAM has the capability of solving DNP equations of the whole system given the power density and delayed neutron source distributions.

The multiphysics model consists of three models coupled using the under MOOSE MultiApp system [6]: (1) Griffin neutronics model [9], (2) SAM core thermal-hydraulics model with porous media [10], and (3) SAM outer loop thermal-hydraulics model [10]. In this model, the DNP equations are solved within SAM and transferred to the Griffin neutronics model to construct the total fission source. Fig. 2 shows the components of the multiphysics model and the exchanged parameters between different codes. The details of the model including multi-D meshes, transfer algorithm of the passed field variables, and iteration and integration schemes used in coupling Griffin and SAM multi-D and 1D models are discussed in Ref. [10, 11].

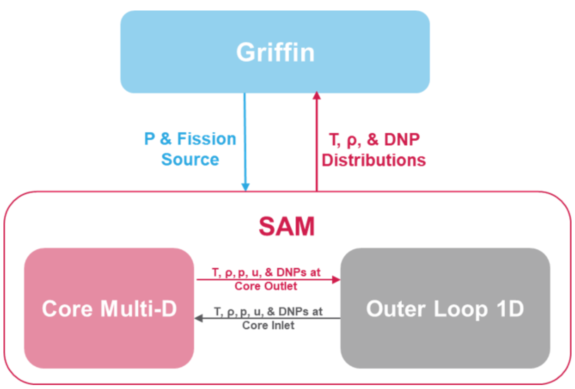

Fig. 2. Coupling scheme for multiphysics core calculations of MSRE using Griffin and SAM [9].

The Monte Carlo code OpenMC [11] was used to generate microscopic cross sections and the delayed neutron data (of each delayed group for fissionable isotopes) of the MSRE as a function of the fuel salt temperature and density in order to perform neutronics steady-state and transient calculations. A full core 3-D model was developed for MSRE using the OpenMC code to generate multigroup microscopic cross sections along with the ENDF-VII.1 data set. The OpenMC model of the MSRE was adopted from Ref. [3, 4], where an optimized energy group structure was selected for thermal spectrum MSRs that consist of 16 energy groups.

Fig. 3 shows the developed full core model utilized for multiphysics analysis of the core and outer loop. The Griffin neutronics model is limited to the core region and more details can be found in Ref. [9]. The SAM thermal-hydraulics model of the MSRE was developed considering porous media along with the DNP drift within the core and its decay in the outer loop. The model is described in detail in Ref. [10].

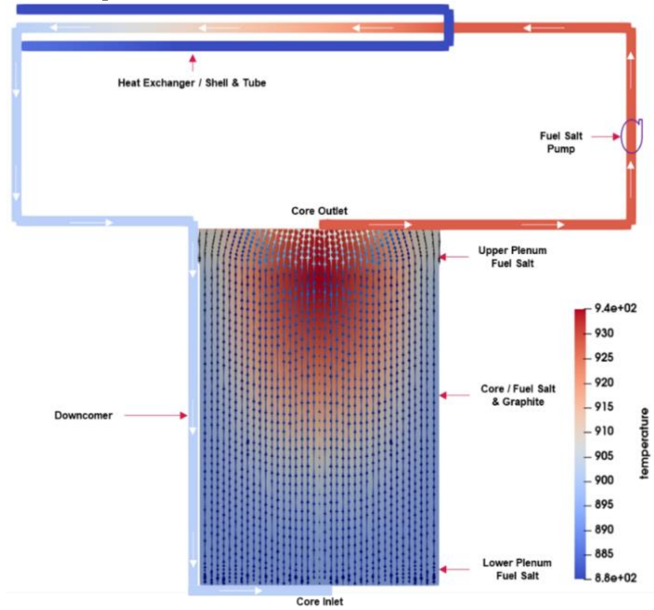

Fig. 3. Simplified full core model of MSRE Experiment [9].

RESULTS AND ANALYSIS

The results of the MSRE initial core steady-state and transient solutions are discussed in this section. The steady state solution focuses on showing the impact of the DNPs on reactivity losses of eigenvalue and effective delayed neutron fraction. The transient tests show unprotected loss of flow (ULOF) scenarios at full and zero power levels to assess the impact of the DNPs during the transient.

Steady State Results

This section presents the steady-state calculation results of the MSRE simplified core. To demonstrate the impact of the DNPs drift on eigenvalue in terms of reactivity several cases were considered with and without thermal feedback, delayed neutron source, and flow of the fuel salt as given in Table III. The reactivity change was calculated relative to the case with the delayed neutron source and without fuel salt flow. The delayed neutron contribution is about 670 pcm reactivity which is the value of the effective delayed neutron fraction of stationary fuel. The drift of the DNPs in the core and its decay in the outer loop results in about 243.0 pcm reactivity losses. The effect of the thermal feedback on the reactivity change with delayed neutrons is minimal. Also, Table IV provides the kinetics parameters of stationary and flowing fuels and the losses in effective delayed neutron fraction of each DNP group. The kinetics parameters were calculated from static calculations and utilizing the multiapp system. The measured value of the DNPs losses for MSRE is 212.0 pcm which is 30 pcm lower than the current value calculated by the developed multiphysics model for MSRE in this work.

The steady state solution of the coupled system for power density, fluid and solid temperatures is shown in Fig. 4 and the distribution of the DNPs in the core region is shown in Fig. 5 for all DNP groups. The power peaks at the core center and small peaks are observed in the upper and lower plena due to massive change in the fuel salt volume. The DNP groups with longer half-life tend to decay mostly in the upper core region and in the outer loop (group 1 and 2), while the groups with shorter half-life decay mostly in the active core region. Note That group 6 and has distribution similar to the prompt neutron source.

Table III. Impact of DNPs drift on eigenvalue and reactivity.

No.	Feedback	DNPC	Flow	k_{eff}	Reactivity [pcm]
1	-	x	-	1.03834	0.0
2	-	-	-	1.03114	672.5
3	-	x	x	1.03574	241.7
4	x	x	-	1.03748	0.0
5	x	-	-	1.03028	673.2
6	x	x	x	1.03487	242.8

Table IV. Kinetic parameters of the MSRE experiment.

Group	λ [s^{-1}]	β_{eff} [pcm]		
		Stationary	Flowing	Losses
1	0.013	23.7	0.6	23.0
2	0.033	121.8	37.4	84.4
3	0.121	116.3	53.5	62.8
4	0.303	259.2	184.6	74.6
5	0.849	106.8	104.8	2.0
6	2.853	44.7	48.1	-3.4
Total	-	672.5	429.0	243.5
Mean Generation Time [ms]			0.259	

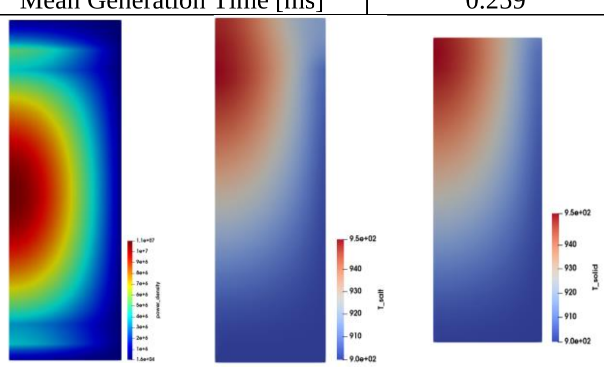
Fig. 4. MSRE steady-state solutions power density, fluid and solid temperatures (left to right).

ULOF Transient Scenario

The ULOF is one of the most important design basis accidents for MSRs. The reactivity feedback of MSRs has three main mechanisms (1) the temperature effects through Doppler, (2) salt expansion or density changes, and (3) The flow of the fuel results in DNP losses and any flow changes result in reactivity changes relative to the salt flow rate. At full power operation, the fuel temperature feedback is the dominant feedback mechanism while at low power operation, the flow rate change is the main feedback mechanism. To demonstrate different feedback mechanisms and the impact of the DNPs during transient analysis ULOF for the MSRE core was simulated at full and zero power levels. The ULOF accident scenario considered in this work assumes a complete loss of the primary cooling pump while the secondary loop system remained operational. Prior to the transient starts, the steady-state conditions were established by running a null transient. Then, the primary pump head was linearly reduced to reach the desired value. Beyond that point there were no changes in system input parameters.

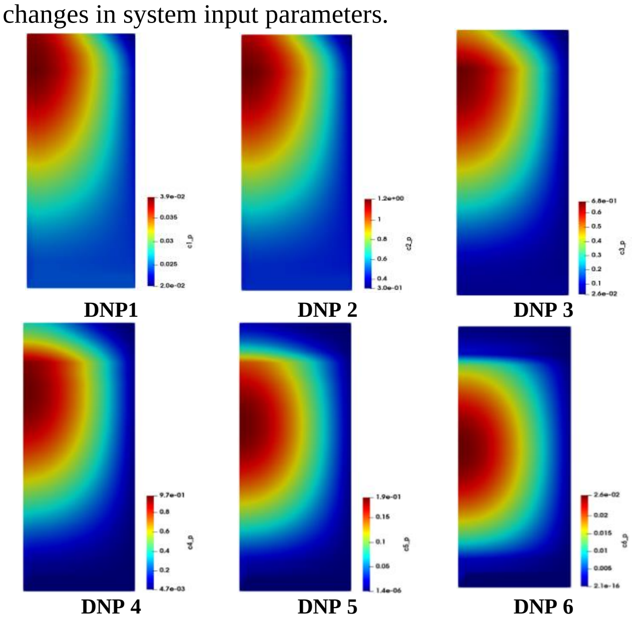

Fig. 5. MSRE steady-state delayed neutron precursor distributions in core region.

The ULOF results at full power level are shown in Fig. 6. To understand the power and temperature evolution, the transient was divided into three regions. (1) null transient region: where there was no change in input parameters of the system to confirm steady state conditions are well established. (2) Power oscillation region: in this part of the transient the fuel flow rate was linearly reduced to 1% if its nominal value in 30 s, following that, the fuel salt temperature increases which introduces a negative reactivity due to Doppler and salt expansion. On the other hand, fuel salt velocity decreases which introduces a positive reactivity due to the reduction in delayed neutron losses. The negative temperature feedback is the dominant feedback mechanism, and the power starts decreasing. Once the fuel salt is at the natural circulation level the unheated salt starts flowing back into the core at slower rate and the power increases again due to positive feedback of the lower salt temperature. An oscillatory temperature behavior is observed due to the heat transferred to the graphite and extracted by the secondary system. Eventually, the power fluctuations decrease when the average temperature stabilizes as the fuel salt and graphite reach thermal equilibrium (region 3) and power stabilizes at new level as the temperature fluctuations vanish which is about 45% of full power value.

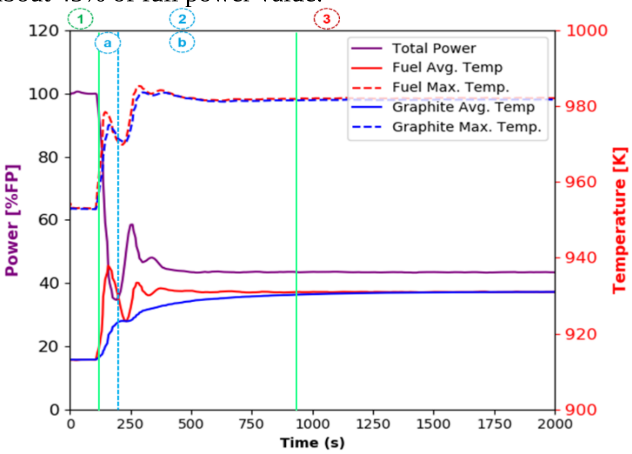

Fig. 6. Power and temperature evolution during a ULOF at full power level.

The ULOF results at zero power level is shown in Fig. 7. Similar to the full power level, (1) null transient was maintained with no change in input parameters while the reactor is at low power level about 0.01% of its full power. (2) Power oscillation region: in this part of the transient the fuel flow rate was linearly reduced to 1% if its nominal value in 30 s. following that, the fuel salt velocity decreases which introduces positive reactivity due to the reduction in delayed neutron losses. Also, the fuel salt temperature increases which introduces negative reactivity due to Doppler and salt expansion. The positive velocity feedback is the dominant feedback mechanism, and the power starts increasing and reaches its peak value at about 14% of its full power value which is 1400 times its initial value. As a result, the fuel salt starts heating up as power increases which introduces a negative temperature feedback that became the dominate feedback mechanism after about 700 s and the power starts decreasing. The oscillatory temperature behavior is observed due to the heat transferred to the graphite and extracted by the secondary system. Eventually, the power fluctuations decrease as average temperature stabilizes as the fuel salt and the graphite are in (3) thermal equilibrium and power stabilizes at new level as the temperature fluctuations vanish which is about 5% of full power value.

SUMMARY

In this work, a multiphysics full core model was developed to simulate the impact of the DNPs drift on steady state and transient analyses. A decrease in the effective delayed neutron fraction of 243 pcm was observed relative to the stationary fuel case. Also, the ULOF accident scenario was simulated at full power and zero power case to show the impact of different feedback mechanisms and their progression during the simulated transient. In the full power case, the negative temperature feedback dominates the positive feedback of the reduced delayed neutron losses. Further validation tests of this MSRE model will be investigated in the future.

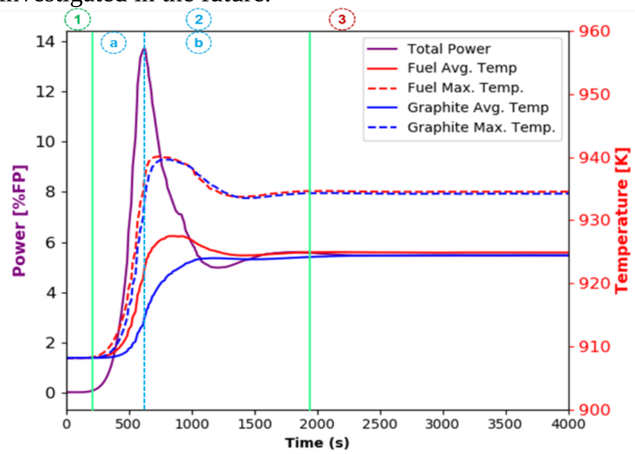

Fig. 7. Power and temperature evolution during a ULOF at zero power level.

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