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INTRODUCTION

High-Temperature Gas-cooled Reactors (HTGRs) have excellent characteristics in terms of safety and high thermal efficiency, and they are gaining a large interest from the industry as a candidate of Gen-IV reactors for a wide range of applications. The High Temperature gas-cooled Reactor Pebble-bed Module project (HTR-PM) is one of these designs and where helium gas is used to cool the pebble-bed region that consists of spherical fuel elements moderated with graphite. The HTR-PM design is based on the combined experience from the German pebble-bed reactor program from the 1960s through the 1990s and the HTR-10 experience in China during the 2000s [1, 2, 3].

Idaho National Laboratory has a long experience in modeling of HTGRs working in developing neutronics and thermal hydraulics tools for the proper modeling of these reactors. The neutronics code Griffin has the capability to model pebble depletion [4]. While the thermal hydraulics code Pronghorn [5] was developed mainly to model the pebble bed reactors with the porous media assumption.

In this work, an equilibrium core Multiphysics model was developed for the HTR-PM reactor to analyze the depressurized loss of forced cooling accident scenario (DLOFC). This paper is organized as follows: First, a brief description of the reactor and model specifications are provided. Then, the developed Multiphysics model is discussed. Finally, verification results of the steady-state equilibrium core and DLOFC accident are presented followed by a summary of the conclusions.

HTR-PM REACTOR DESCRIPTION

The HTR-PM is a 250.0 MWth two-module reactor with main characteristics including a cylindrical pebble bed region surrounded by radial, lower, and upper graphite reflectors. The radial reflector includes various orifices for the control rod channels and fluid riser channels. The HTR-PM regions with coolant flow path are shown in Fig. 1, and the design specifications are listed in Table I [1, 2]. The HTR-PM reactor layout can be found in Ref. [1].

The pebble bed region has an 11 m height loaded with about 420,000 (at equilibrium state) spherical pebble fuel elements each having a 6 cm diameter. Each pebble consists of TRISO-coated fuel particles dispersed in a graphite matrix with a diameter of 5.0 cm with 7.0 g heavy-metal loading surrounded by a 1 cm graphite shell. The pebbles will reach the maximum design burnup of 100.0 MWd/kgU after approximately 15 passes through the reactor core.

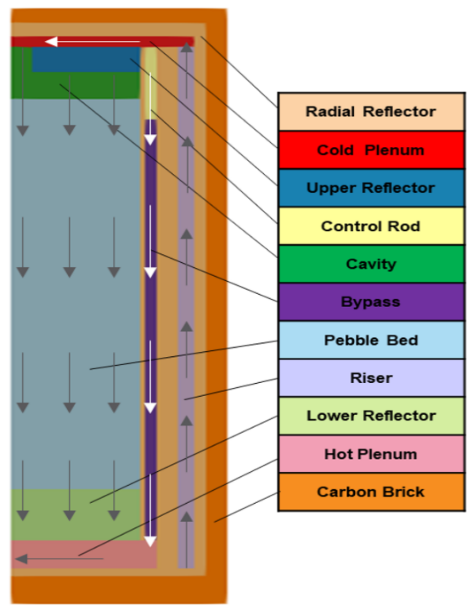

Fig. 1. HTR-PM reactor core geometry considered in the model.

TABLE I. HTR-PM core specifications [1, 2].

Parameter	Value
Core power [MWth]	250.00
Core inlet temperature [K]	523.15
Core outlet temperature [K]	1023.15
Core outlet pressure [MPa]	7.0
Pebble-bed radius [m]	1.50
Pebble-bed height [m]	11.00
Reflector outer radius [m]	2.50
Control rods channels	24
Reactivity Shutdown Channels	4
Barrel outer radius [m]	2.69
Bypass outer radius [m]	1.69
Vessel outer radius [m]	3.00
Number of pebbles	419,384
Pebble types	1 pebble type
Avg. pebble packing fraction	0.61
Avg. number of passes	15
Avg. pebble residence time [days]	70.5

ANALYSIS METHOD AND MODEL DESCRIPTION

In this section, the developed models and Multiphysics coupled system of the HTR-PM problem are described. The Griffin code (MOOSE-based reactor physics application) is used to provide the neutronics solution of the equilibrium core [4]. While the Pronghorn code (MOOSE-based coarse-mesh thermal-hydraulics application) solves the porous medium for fluid flow equations with conjugate heat transfer and heat conduction through a solid medium [5]. Graphite and TRISO particle temperatures in each pebble type are computed using spherical heat conduction models. Both codes are coupled under MOOSE MultiApp [6]. The neutronics model of the HTR-PM was developed in axisymmetric geometry (R-Z), considering different reactor regions, including the control rod and outer core regions. Fig. 2 shows the developed model for neutronics and thermal hydraulics analysis with dimensions and helium flow regions and directions. The description of the various regions and their fractional densities are included in Table II. The neutronics model is limited to the radial reflector (region 8). A detailed description of these models can be found in Ref. [7, 8].

TABLE II. HTR-PM core regions and fractional densities.

ID	Region	Fractional density
1	Pebble bed	0.61
2	Top reflector	0.70
3	Bottom reflector	0.70
4	Cavity	1.00
5	Hot plenum	0.80
6	Cold plenum Graphite	0.80
7	Side reflector	1.00
8	Carbon brick	1.00
9	Gas gap	1.00
10	Core barrel	1.00
11	Gas gap	1.00
12	Reactor pressure vessel	1.00
61	Riser	0.68
71	Bypass channel -Control rod channel -Empty control rod channel	- 0.72 1.00

The developed HTR-PM Multiphysics model for equilibrium core calculations is depicted in Fig 3. Griffin (mainApp) solves the neutronics problem that includes the steady state neutronics calculations and the streamline depletion-advection problem and establishes the transfer system to exchange coupling variables with subapps. Pronghorn solves the thermal fluid problem to obtain fluid and solid temperatures using the power density profile. The pebble and TRISO temperatures are calculated with a 1D heat conduction problem in spherical geometry, where a subapp is used for each pebble type and burnup group in the active core region. The microscopic multigroup cross sections were prepared using DRAGON [9] code along with ENDF/B-VIII.r0 data set.

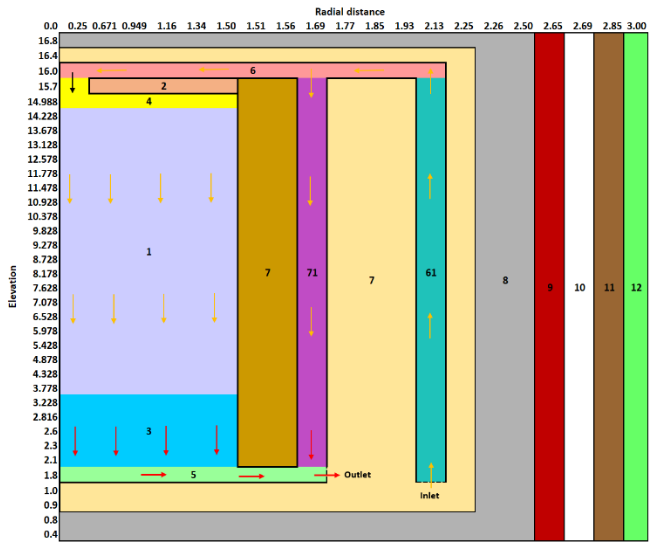

Fig. 2. Geometry of the HTR-PM model. The dimensions are in meters. The fluid flow domain is indicated by arrows.

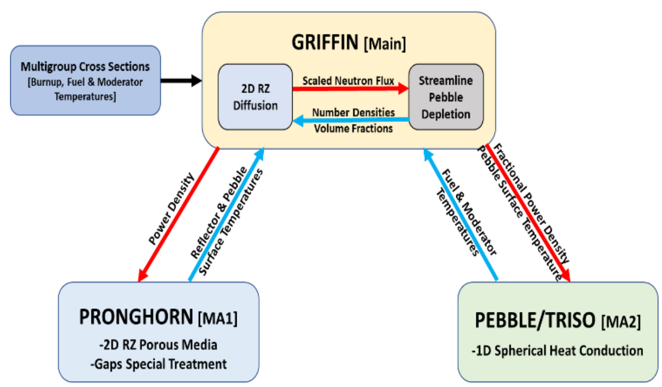

Fig. 3. Coupling scheme of the equilibrium core Multiphysics calculations of HTR-PM.

RESULTS AND ANALYSIS

This section presents the results of the equilibrium core steady-state and transient calculations and compared against available VSOP (very superior old programs) reference solution obtained from the open literature for purpose of verification.

Equilibrium Core Results

Before performing transient analysis of the HTR-PM, the developed steady-state Multiphysics model of the equilibrium core is verified against the VSOP results considering eigenvalue, peak power, power density, average and maximum fuel temperatures, temperature coefficients, and kinetics parameters as provided in Table III considering a discharge burnup of 90 MWd/kgU. In general, the compared parameters are in good agreement with the reference VSOP solutions, and the discrepancies are mainly attributed to differences in the discretization methods, multigroup cross section generation, and lack of information on the VSOP models. The VSOP eigenvalue was not reported, so it was assumed critical. The peak power density and power ratio are in very good agreement with the VSOP results. Differences of respectively 40 and 20 K were observed in the maximum and average fuel temperatures, mainly due to differences in the discretization methods and the material properties. Fig. 4 shows the power density, peak power, temperature profiles, and velocity fields of the equilibrium core.

TABLE III. Nuclear parameters of equilibrium core, with a discharge burnup ~90 MWd/kgU.

Parameter		**Griffin**	**VSOP**
Eigenvalue		0.9958	1.0000
Power Peak		2.00	2.04
Peak Power Density		6.4	6.56
Aveg. Burnup [MWd/kgU]		53.62	-
Max, Fuel temp. [K]		1124.6	1163.2
Avg. Fuel Temp. [K]		899.0	873.2
Avg. Moderator Temp. [K]		886.0	-
Avg. Fluid Temp.[K]		653.0	-
Temp. coef. [pcm/K]	Fuel	-4.32	-4.36
	Moderator	-1.92	-0.94
	Reflector	0.72	1.49
Kinetics Parameters	β_{eff} [pcm]	519.0	549.0
	Lambda [ms]	1.3	1.1

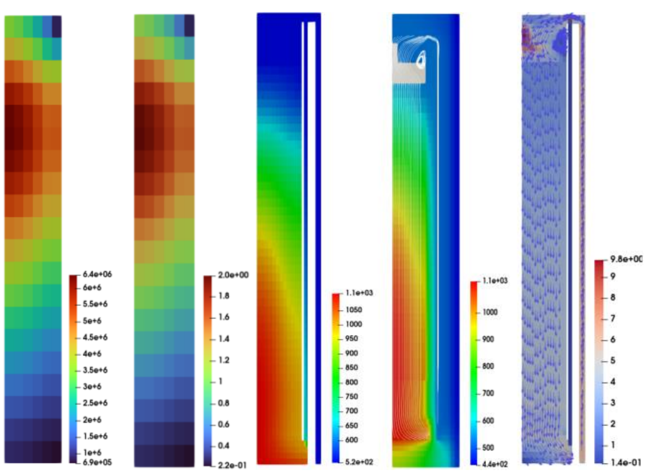

Fig. 4. Equilibrium core steady state solution power density, peak power, fluid and solid temperatures, and velocity field (left to right).

DLOFC Transient Scenario

The DLOFC accident scenario is a challenging design basis accident due to the assumption that the coolant has completely leaked out of the reactor pressure vessel. In this accident scenario, a large break in the main coolant flow pipe was assumed, resulting in a fast reduction of the helium inventory in the system followed by a depressurization of the reactor. The natural circulation effect has a minimal contribution to the heat extraction from the fuel due to the low coolant density at atmospheric pressure. Instead, the heat will be extracted from the fuel via conduction and radiation, through the reflector, barrel, reactor pressure vessel, and reactor cavity cooling system. In the DLOFC accident, the reactor power is not the main concern as it decreases to a very low level at the beginning of the transient, while it is more important to make sure that the reactor is able to remove the heat efficiently and monitor that the maximum fuel temperature to ensure it is not exceeding the safety limits.

In this work, the DLOFC transient simulation was initiated by reducing the mass flow rate of the coolant from its nominal value to zero over 13.0 s. At the same time, the system pressure was reduced from 7.0 MPa to 0.101 MPa (atmospheric pressure), assuming these changes will behave linearly. The simulated DLOFC transient is a protected transient, therefore, after the initiation of the accident, the control rods were fully inserted to shut down the reactor after completing the flow rate and pressure ramps. Beyond that, there were no changes to the system's main parameters, and the simulation was performed for up to 140 hours. The sequence of events for the simulated DLOFC accident with the developed Multiphysics model of the equilibrium core is listed in Table IV.

TABLE IV. DLOFC sequence of events.

Time [s]	Event
< 0	-Equilibrium steady state completed
0	-Start of accident
0 – 13	-Start linearly reducing outlet pressure to 0.1 MPa over 13 s - Start linearly reducing mass flow rate to 0 kg/s over 13 s
13	-Mass flow rate and pressure ramps completed -Initiate SCRAM
13 - 16	-Fully insert control rods
16	-SCRAM completed Power level is determined by decay heat until end time
360,000	-Simulation end time

During the DLOFC transient simulation, coupled neutronics and thermal-hydraulics calculations were performed at each time step to obtain the time-dependent power density distribution in the pebble-bed region considering decay heat produced by the decay of the fission products. The decay heat is the main driving heat source during the transient as the prompt power goes to zero following the control rod insertion. The evolution of the reactor's total power and the maximum and average fuel temperatures during the DLOFC accident are shown in Fig. 5. The reactor power starts decreasing at the beginning of the transient due to the negative thermal feedback as the fuel and moderator temperatures increase following the helium flow rate reduction in the system. After 15 s, a large reduction in the reactor power can be noticed as the control rods are fully inserted to shut down the reactor. Beyond that point, the prompt power goes to zero while the remaining reactor power is just the decay heat component of the fuel.

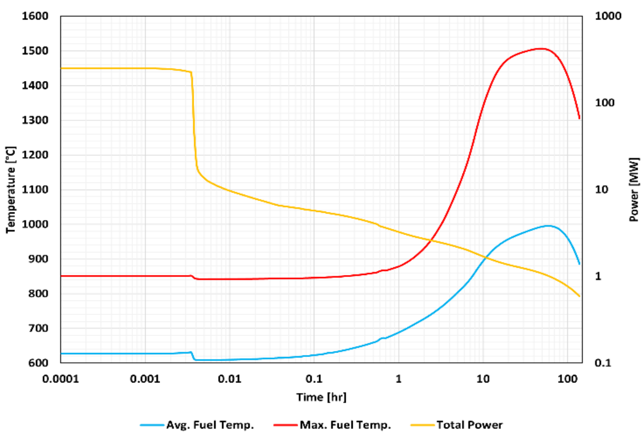

Fig. 5. Power and temperature evolution during a DLOFC.

The fuel maximum and average temperatures during DLOFC are compared to the VSOP results as shown in Fig. 6. In general, the temperature matches the VSOP solution for the first 40 hours of the transient and starts to deviate from the reference solution due to differences in the thermal-hydraulics models, thermophysical properties, heat extraction mechanisms, and problem boundary conditions.

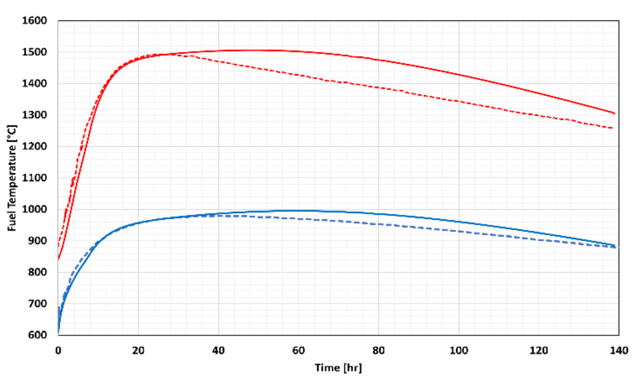

Fig. 6. Comparison of average and maximum fuel temperatures during a DLOFC.

The fuel temperature reaches its maximum value after 20 hours from the start of the transient. During steady-state operation, the fluid and solid temperatures peak at the center-bottom of the pebble bed. The reflector temperature is higher at the bottom region, rather than the radial and top reflectors, which are much cooler. Once the transient is started, the maximum fluid and solid temperatures start moving axially and toward the top of the core, which is caused by the decay heat distribution that peaks towards the top of the core. Beyond 10 hours of the transient, the temperature distributions changes mainly in the radial direction.

SUMMARY

In this work, the protected DLOFC accident scenario of the HTR-PM equilibrium core was simulated using the Multiphysics coupled code system (Griffin-Pronghorn). The steady state and transient solutions of the equilibrium core were assessed against reference solutions obtained from open literature for the purpose of verification. A good agreement was observed in general for all solutions, and several sources of differences were identified between the current and reference models. The model shows the applicability of the developed tools to handle HTGR for equilibrium core calculations for steady-state and transient analyses. Further improvements and verification tests will be performed in the future.

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