Structural Safety Assessment of Graphite Monolith in Heat Pipe Cooled Microreactor (HPMR) Considering Neutron Irradiation Effect

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1. Introduction

The heat pipe cooled microreactor (HPMR) is a type of microreactor that utilizes heat pipes for passive removal from the reactor core. HPMRs offer several advantages, such as compactness, transportability, and enhanced system reliability and safety. Following the successful demonstration of the KRUSTY reactor in 2018 [1], interest in HPMRs has significantly increased compared to other microreactor concepts. Representative terrestrial HPMR designs include LANL's MegaPower reactor [2], Westinghouse's eVinci reactor [3], and space reactor concepts of NASA [4] and the Korea Atomic Energy Research Institute (KAERI) [5] are being developed.

For thermal-spectrum HPMRs that incorporate a moderator or a graphite monolith, the structural integrity of the monolith may be compromised by irradiation-induced phenomena, such as dimensional changes, creep, and degradation of material properties [6]. Therefore, to ensure the structural safety of the core, a multiphysics analysis tool capable of capturing coupled neutronic, thermal, and structural phenomena is essential.

In this study, structural safety assessment of graphite monolith in thermal spectrum HPMR was conducted. This paper introduces the multiphysics analysis tool for the neutronics-thermal-structural analysis of the HPMR core, and structural safety assessment result with neutron irradiation effect based on ASME BPVC criteria.

2. Multiphysics analysis tool

2.1 OpenFOAM

OpenFOAM is an open-source CFD software package that includes a basic solver for thermal-structural analysis [7]. In this study, the solidDisplacementFoam solver from the OpenFOAM version 7 was employed [8]. This solver assumes linear elasticity and evaluates the temperature, strain, and stress fields by solving the following governing equations.

Solid momentum equation

$$\frac{\partial^2(\rho D)}{\partial t^2} = \nabla\cdot[(2\mu+\lambda)\nabla D] + \nabla\cdot[\mu(\nabla D)^{\mathrm{T}} + \lambda I tr(\nabla D) - (\mu+\lambda)\nabla D] + \rho g - \nabla(3K\alpha T) \quad (1)$$

Heat conduction equation

$$\rho C_p \frac{\partial T}{\partial t} = \nabla\cdot(k\nabla T) + q''' \quad (2)$$

2.2 PRAGMA

PRAGMA is a continuous-energy Monte Carlo neutron transport code developed at Seoul National University [9]. By utilizing an event-based tracking algorithm optimized for GPU architectures, it achieves significantly higher performance compared to traditional CPU-based Monte Carlo codes. Although initially designed for pressurized water reactor (PWR) analysis, PRAGMA has since been extended to support unstructured-mesh calculations, enabling high-fidelity modeling for advanced reactor applications..

2.3 ANLHTP

ANLHTP [10] is a one-dimensional(1D) steady-state heat pipe analysis code developed at Argonne National Laboratory (ANL). It is designed to analyze sodium heat pipes, predicting their operating limits—including viscous, boiling, entrainment, and sonic limits—and heat transfer performance through a thermal resistance network. The operating limits considered in ANLHTP include the viscous, boiling, entrainment, and sonic limits. The source code of ANLHTP is publicly available in the open literature, and the code was reconstructed and validated against various experiments [11]. However, the original structure of ANLHTP is not well suited for efficient coupling with OpenFOAM. To enable effective coupling, the code was reimplemented in C++ language, consistent with the programming language used in OpenFOAM [12].

2.4 Code coupling

The coupling of OpenFOAM, PRAGMA, and ANLHTP is implemented using an MPI-based manager–worker method, which ensures robust interoperability even when the codes are executed with different MPI configurations [12]. Within this framework, the manager coordinates the exchange of coupling variables: OpenFOAM provides temperature and density fields to PRAGMA, while PRAGMA returns the computed power distribution and neutron flux. Because OpenFOAM and PRAGMA utilize an identical mesh, no additional mapping is required, facilitating the simulation of localized neutron irradiation effects via cell-wise flux distributions.

OpenFOAM and ANLHTP exchange temperature and heat flux at the wick–vapor interface inside the heat pipe. This approach enables OpenFOAM to solve three-dimensional heat conduction and account for axial heat transfer along the heat pipe wall which cannot be considered in one-dimensional ANLHTP model. The schematic of manager-worker system of coupled code is shown in Figure 1.

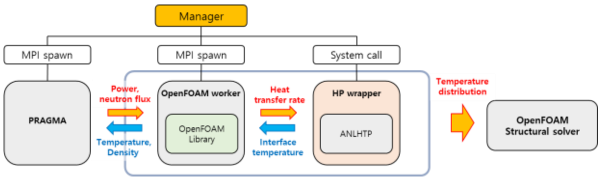

Fig. 1. Schematic of manager-worker system for the PRAGMA-OpenFOAM-ANLHTP coupled code.

3. Safety acceptance criteria of graphite structure

3.1 Safety acceptance criteria from ASME BPVC

Graphite structures are addressed in subsection HHA of the ASME BPVC Section Ⅲ Division 5. The ASME BPVC does not prescribe an explicit temperature limit for graphite structures. Instead, it stipulates that the applicable temperature range should be defined by the manufacturer's quality-assurance envelope, which must explicitly account for temperature and neutron irradiation effects [13]. Generally, graphite is reported to have a practical temperature limit exceeding 2000 °C in non-oxidizing environments. In HPMR cores, fuel temperatures typically exceed those of the monolith. For TRISO fuel—commonly used with graphite monolith concepts—the temperature limits are approximately 1250 °C during normal operation and 1600 °C under design-basis accident conditions [14]. Consequently, rather than applying an explicit temperature limit for the graphite monolith, the TRISO fuel temperature limits were adopted as the governing thermal criteria.

According to the ASME BPVC, graphite components are evaluated using a brittle, probabilistic approach. Structural safety is assessed in terms of the probability of failure (POF), with failure defined by an equivalent stress. In the simplified assessment process, the maximum allowable stress S_g is determined using a two-parameter Weibull distribution obtained from specimen test data, as expressed by Eqn. (3).

$$S_g = S_{c,95\%}[-\ln(1 - POF)]^{\frac{1}{m_{95\%}}} \tag{3}$$

3.2 Neutron irradiation effect on graphite

Graphite is known to be strongly affected by neutron irradiation. According to subsection HHA-3142.3 of ASME BPVC Section Ⅲ Division 5, the structural evaluation of graphite components must account for irradiation-induced dimensional changes, creep, and changes in material properties. The additional stresses arising from neutron irradiation can be incorporated by superposing them onto the non-irradiated stress field. In this study, irradiation-induced dimensional change, irradiation creep, and changes in thermal conductivity, coefficient of thermal expansion, and Young's modulus of graphite are considered.

4. Multiphysics analysis of rHPMR

4.1 Modeling of rHPMR

Within the multiphysics core analysis program led by Idaho National Laboratory (INL), the 5 MW(th) realistic Heat Pipe-Cooled Microreactor (rHPMR) was established as a benchmark problem for the CRAB multiphysics framework [15]. The rHPMR core configuration is illustrated in Figure 2. Each hexagonal graphite assembly accommodates 72 TRISO fuel compacts and 19 sodium heat pipes, with the entire core comprising 18 of these assemblies.

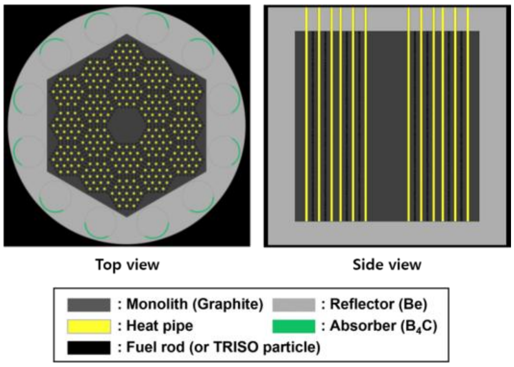

Fig. 2. Schematic of rHPMR core.

The rHPMR has a total thermal power of 5 MW and incorporates 342 heat pipes within the core. Since certain detailed specifications for the graphite and heat pipes were not fully provided in the reference, IG-110 graphite was assumed as the monolith material in this study. Additionally, a virtual heat pipe model was established based on the operating limits reported in the INL documentation. Figure 3 illustrates the temperature-dependent operating limits and the heat removal performance under nominal operating conditions.

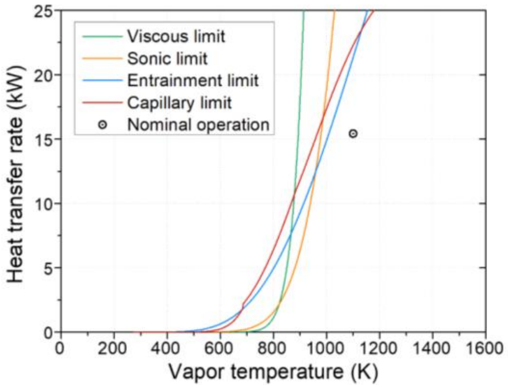

Fig. 3. Operation range and limit of virtual sodium heat pipe.

4.2 Multiphysics analysis without neutron irradiation

First, a steady-state multiphysics analysis was conducted for nominal operation state. For boundary conditions, ANLHTP was coupled for heat pipe boundary and adiabatic conditions were applied to other boundaries. The power and temperature distributions presented in Figure 4. Radially, power peaking occurs in the central and peripheral fuel regions, where the moderating and reflecting effects of the graphite and reflector are most pronounced. Under nominal operating conditions, the maximum temperature of the TRISO fuel compacts was found to significantly exceed the design limit of 1250 °C.

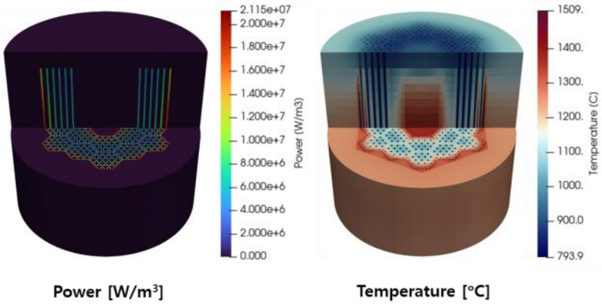

Fig. 4. Multiphysics analysis results of the rHPMR core.

The steady-state thermal analysis results were subsequently utilized as input for the structural analysis. Assuming a horizontal reactor orientation, normal direction constraint was applied to the surface in contact with the floor, while all other surfaces were modeled to allow free expansion. The structural analysis result is shown in Figure 5. The maximum stress is 26.6 MPa near the bottom surface, which exceeds the maximum allowable stress of IG-110 graphite, 12.29 MPa.

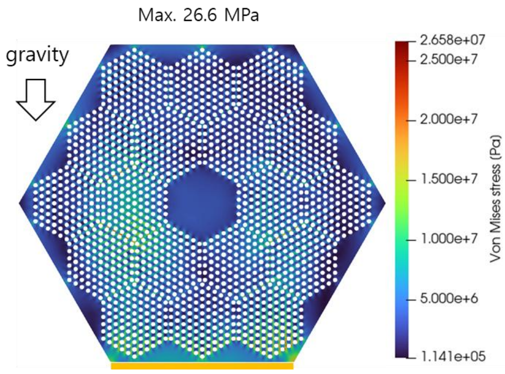

Fig. 5. Structural analysis results for rHPMR core.

4.3 Multiphysics analysis with neutron irradiation (EOL)

To evaluate the effects of neutron irradiation, the neutron fluence was calculated based on the multiphysics analysis results presented in Section 4.2. Figure 6 illustrates the energy-dependent neutron flux for obtained from the steady-state analysis. The neutron fluence at the End of Life (EOL) was determined by integrating neutron flux over the rHPMR's operational lifetime. The EOL was set at 1,400 days in accordance with the reference [15]. Consequently, irradiation-induced dimensional changes, creep, and changes in material properties based on the references [16,17] were quantified and incorporated into the analysis. The multiphysics analysis results and structural analysis result considering neutron irradiation effect are presented in Figure 7–Figure 9.

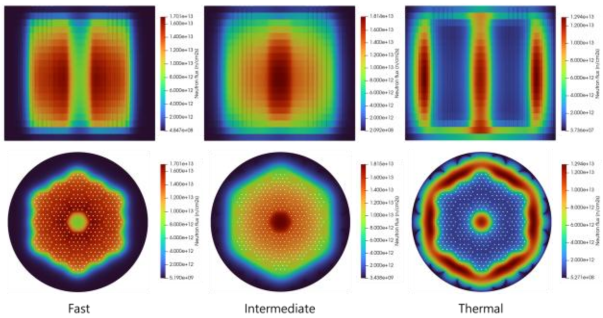

Fig. 6. Neutron flux distributions of rHPMR core.

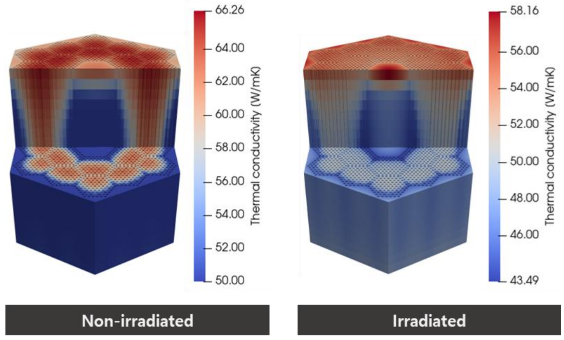

Fig. 7. Thermal conductivity change with neutron irradiation.

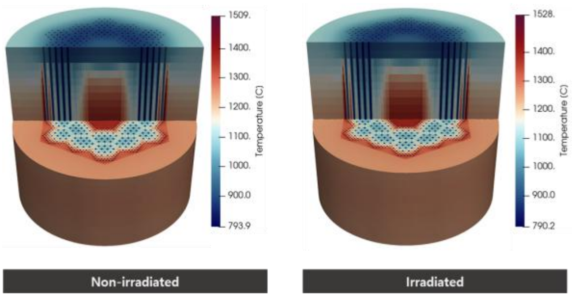

Fig. 8. Temperature distribution for rHPMR core.

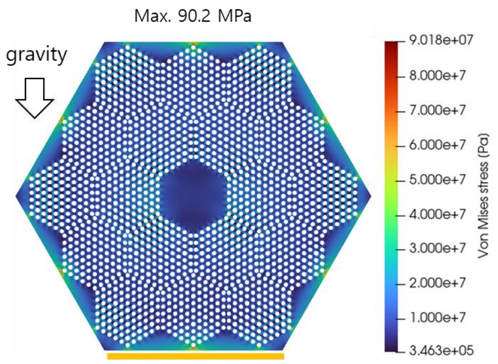

Fig. 9. Structural analysis results for rHPMR core.

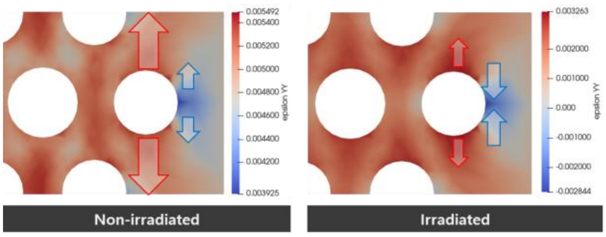

Fig. 10. Local y-direction strain distribution change after neutron irradiation

Neutron irradiation led to a degradation of the thermal conductivity of graphite, which subsequently resulted in elevated core temperature. While substantial increase in stress was observed, it was confirmed that this was primarily driven by irradiation-induced dimensional changes rather than the effects of temperature increase or temperature gradients. Figure 10 shows local y-direction strain where maximum stress appeared. While only the thermal expansion effects are observed under non-irradiated conditions, after neutron irradiation, the strain was decreased and even contraction of graphite appeared. This strain difference within the narrow regions causes a significant increase in stress. Because of these localized dimensional change, the local neutron irradiation effect of graphite should be considered in analysis of HPMR core.

5. Conclusions

In this study, a coupled neutronics–thermal–structural multiphysics analysis tool was developed to evaluate the structural safety of HPMR cores. The framework integrates the OpenFOAM thermal-structural solver with the Monte Carlo neutron transport code PRAGMA and the 1D heat pipe analysis code ANLHTP. In this process, a manager–worker method was implemented to facilitate robust coupling between these codes, even under heterogeneous MPI configurations.

The developed tool was applied to the rHPMR, which features a graphite monolith structure. Based on ASME BPVC Section Ⅲ Division 5, stress-based acceptance criteria and irradiation effects were established and assessed. The steady-state analysis results indicated that core temperatures and stresses exceeded the acceptance criteria under nominal conditions. Furthermore, incorporating neutron irradiation-induced dimensional changes, creep, and property changes led to a marked increase in both temperature and stress levels.

In this study, multiphysics analysis was limited to steady-state analysis because the PRAGMA and the ANLHTP do not currently support transient analysis. However, transient analysis is essential for evaluating accident scenarios such as heat pipe failure, which can trigger cascade failure of heat pipes. Also, it can simulate transient behaviors of core parameters during off-normal events. In future work, transient analysis capability will be developed and validated against experimental data such as KRUSTY reactor experiment, and heat pipe performance experiments.

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REFERENCES

[1] D. I. Poston et al., “KRUSTY Reactor Design”, Nuclear Technology, volume 206, S13-S30 (2020).
[2] P. R. McClure et al., “Design of Megawatt Power Level Heat Pipe Reactors,” Los Alamos, New Mexico, USA, LA-UR-15-28840 (2015).
[3] M. M. Swartz et al., “Westinghouse eVinci™ Heat Pipe Micro Reactor Technology Development,” Proceedings of the 2021 28th International Conference on Nuclear Engineering, Virtual, Online, August 4-6, Vol. 85246, p. V001T04A018 (2021).
[4] M. A. Gibson et al., “NASA's Kilopower Reactor Development and the Path to Higher Power Missions,” in IEEE Aerospace Conference, 20170002010.
[5] B. H. Park et al., “Key Technologies for Korean Space Heat Pipe Reactor,” Nuclear Technology 211, S1–S11 (2024).

[6] T. D. Burchell et al., "Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs) - Volume 5: Graphite PIRTs", NUREG/CR-6944, Vol.5 (2007).

[7] The OpenFOAM Foundation Ltd., "OpenFOAM and The OpenFOAM Foundation,", 2026, accessed May 1, https://openfoam.org

[8] The OpenFOAM Foundation Ltd., "C++ Source Code Guide," 2019, accessed March 1, https://cpp.openfoam.org/v7/solidDisplacementFoam_8C.html

[9] N. Choi, K. M. Kim and H. G. Joo, "Optimization of Neutron Tracking Algorithms for GPU-based Continuous Energy Monte Carlo Calculation," Annals of Nuclear Energy, 162, no. 108508 (2021).

[10] G. A. McLennan, "ANL/HTP: A Computer Code for The Simulation of Heat Pipe Operation," ANL-83-108 (1983), https://digital.library.unt.edu/ark:/67531/metadc283626.

[11] C. H. Lee, Y. S. Jung, H. K. Cho, "Micro Reactor Simulation Using the PROTEUS Suite in FY19", ANL/NSE-19/33 (2019).

[12] J. Im et al., "Multiphysics Analysis System for Heat Pipe Cooled Micro Reactors Employing PRAGMA-OpenFOAM-ANLHTP," Nuclear Science and Engineering, DOI: 10.1080/00295639.2022.2143209 (2023).

[13] ASME, Boiler and Pressure Vessel Code 2015 Edition - Section Ⅲ Division 5 – High Temperature Reactors (2015).

[14] F. C. Trejo et al., "Understanding the 1600°C Fuel Temperature Limit of TRISO Coated Fuel Particles," MRS Online Proceedings Library 1769 (2015).

[15] N. J. Choi et al., "Advancements in Multiphysics Microdepletion Analysis of and eVinci-like Microreactor Leveraging OpenMC-CRAB Workflow", INL/RPT-24-77346 (2024).

[16] T. Shibata et al., Draft of Standard for Graphite Core Components in High Temperature Gas-cooled Reactor, JAEA-Research 2009-042 (2010).

[17] P. Bajpai et al., Development of Graphite Thermal and Mechanical Modeling Capabilities in Grizzly, INL/RPT-24-78905 (2024).