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Thermal–hydraulics analyses for 1/6 prismatic VHTR core and fuel element with and without bypass flow

Boyce W. Travis [a,b], Mohamed S. El-Genk [a,b,c,*]

[a] Institute for Space & Nuclear Power Studies, University of New Mexico, Albuquerque, NM, USA
[b] Chemical & Nuclear Engineering Department, University of New Mexico, Albuquerque, NM, USA
[c] Mechanical Engineering Department, University of New Mexico, Albuquerque, NM, USA

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ABSTRACT

Presented are the results of thermal–hydraulics analyses for a Very High Temperature Reactor (VHTR) 1/6 core and hexagonal fuel element, with and without helium-coolant bypass flow in interstitial gaps and in the control rod channels. Owing to the complexity and massive size of the VHTR, a full core analysis requires extensive and parallelized computation capabilities and a long time (weeks to months) to complete. These demanding requirements are mostly due to the 3-D computational fluid dynamics (CFD) simulation of the helium flow in the 10-m long coolant channels in the reactor core. Results demonstrate the effectiveness of coupling a 1-D helium flow in the channels together with a recently developed and validated convective heat transfer correlation, to a 3-D heat conduction in the graphite and fuel compacts in the core fuel elements. This simplified thermal–hydraulics analysis methodology for VHTR markedly reduces computation time and memory requirements, while maintaining accurate results. The helium bypass flow in interstitial gaps between fuel elements in the VHTR core provides additional cooling of the edge regions, but increases the temperature near the center of the prismatic fuel elements. Results also show that helium "bleed" flow through the control rod channels minimally affects the temperature distribution within the fuel elements. The heat generation in the corner burnable poison rods in the VHTR core fuel elements affects the temperature only in the close vicinity of the rods.

1. Introduction

The development and future deployment of generation-IV, prismatic core, Very High and High Temperature Gas cooled Reactors (VHTRs or HTGRs) require demonstrable and effective computational tools and methodologies for design, operation and safety analyses. These reactors are graphite moderated, helium cooled, and nominally operate at a thermal power of 300–600 MW_{th}, exit temperatures of ~900 to 1273 K and a helium coolant pressure of 4.0–7.07 MPa [1–3]. The primary focus on this work is on the thermal–hydraulics analysis of a prismatic core VHTR (Fig. 1). The annular core of this VHTR is comprised of 102 hexagonal fuel elements stacked 8-m high in three concentric rings. The fuel elements are 0.36 m flat-to-flat, 0.793 m tall and loaded with graphite fuel compacts. The fuel compacts are made of spherical Tristructural-isotropic, TRISO, particles (~1 mm in diameter) dispersed in a nuclear graphite matrix [1–4] (Fig. 2a). The TRISO fuel particles consist of uranium oxy-carbide fuel kernels surrounded by PyC and SiC coatings, designed for full retention of fission products and restraining fuel swelling at the reactor's nominal operating temperatures.

The VHTR fuel elements are surrounded by hexagonal graphite reflector blocks of the same dimensions in the inner five and the outer 2–3 rings of the assembled reactor core. In addition, axial graphite reflector blocks are stacked on the top and at the bottom of the active core (Fig. 1). The fuel compacts, ~1.245 cm in diameter and 4.95 cm tall (Fig. 2b), are loaded into vertical channels arranged in a triangular lattice within the prismatic fuel elements. On average, there are six fuel compact stacks surrounding a helium coolant channel (Fig. 2c) and each channel carries approximately the fission heat generated in two fuel compact stacks. The coolant channels within the stacked fuel elements in the active VHTR core region are ~7.93 m long and most of the channels are 1.5875 cm in diameter. They extend an additional 1.2 and 0.8 m through the top and the bottom graphite reflector blocks. Most fuel elements (Fig. 2c) have an identical configuration and only a few have vertical holes for control rods or reserve shutdown elements (Figs. 2d and 3). Fig. 3 presents a radial cross-section of a prismatic VHTR core. More specifically, it is a layout of a quadrant with the 1/6th core, to be simulated in this paper, highlighted with dashed lines. In this figure, the gray hexagonal assemblies in the annular core

* Corresponding author. Address: University of New Mexico, ISNPS, MSC01-1120, Albuquerque, NM 87131-0001, USA. Tel.: +1 (505) 277 5442; fax: +1 (505) 277 5433.

E-mail address: mgenk@unm.edu (M.S. El-Genk).

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Nomenclature

A	cross-section area of flow channel (m^2)
C_p	specific heat (kJ/kg K)
D	coolant channel diameter (m)
k	thermal conductivity (kW/m K)
h	heat transfer coefficient (W/m^2 K)
Nu	Nusselt number, $h\ D/k$
Pr	Prandtl number, $\mu\ C_p/k$
Re	Reynolds number, $\dot{m}D/A\mu$
T	temperature (K)
z	axial distance from reactor core entrance (m)

Greek letters

μ	dynamic viscosity (Pa s)
ρ	density (kg/m^3)

Subscripts/superscripts

b	helium coolant bulk
w	coolant channel wall

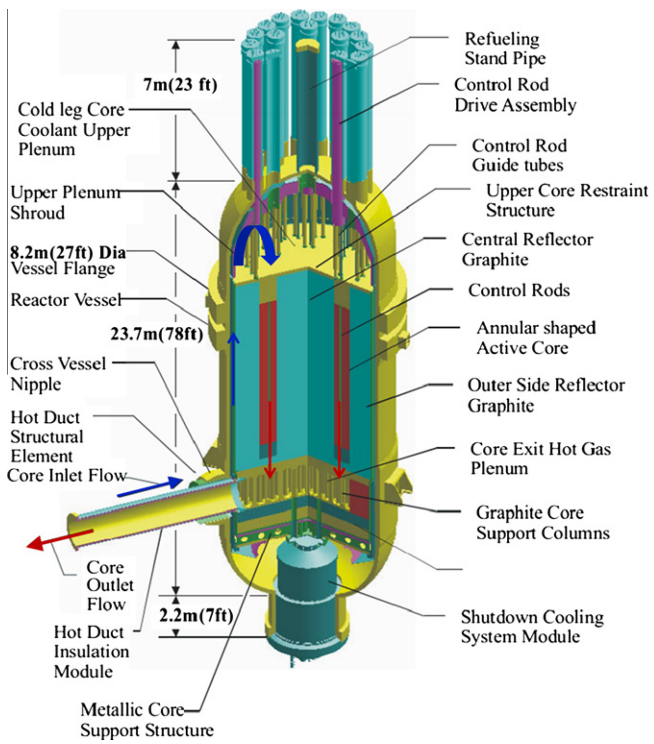

Fig. 1. A longitudinal cut-away cross-sectional view of a prismatic VHTR core [1–4].

region are the fuel elements (Fig. 2c) and the lighter shade hexagonal assemblies are those of the inner and outer graphite reflectors.

Ideally, a VHTR full core thermal–hydraulics analysis is desired to investigate the performance and optimize the designs of prismatic HTGRs or VHTRs. However, owing to the complexity and massive size of the core in these reactors, such an analysis requires extensive and massively parallelized computation capabilities and a long time (weeks to months) to complete. These requirements are primarily due to the 3-D computation fluid dynamics (CFD) simulation of the helium gas flow in the ~10-m long cooling channels in the 102 hexagonal fuel elements of the active core and in the top and bottom axial graphite reflector blocks. The helium bypass flow in the interstitial spaces between the fuel elements in the VHTR further complicates the helium flow distribution and the thermal–hydraulics analysis of the core. These spaces or small gaps would be initially present due to manufacturing tolerances of the hexagonal fuel elements and the reflector blocks. The width of the interstitial spaces also changes in the different sections of the core with operation time of the reactor due to thermal expansion and irradiation swelling. Although the exact dimensions are not well characterized, the bypass flow interstitial spaces could be up to 5 mm wide, or even larger, depending on the reactor design and operation conditions.

Bypass flow in a prismatic VHTR core raises concerns with regards to the helium coolant flow distribution and the potential for developing hotspots in the fuel elements. Little work has been reported on the bypass flow distribution and its effect on the thermal performance of the prismatic VHTR core, thus it remains a subject for future investigations [5–9]. The helium bypass flow is that which does not traverse the coolant channels in the reactor core, but flows through the interstitial spaces between the vertically stacked prismatic fuel elements. A helium "bleed" flow outside the core coolant channels is that entering from the upper plenum through orifices into vertical channels to cool the control rods in some of the core fuel elements. This and the bypass helium flow exit to the lower plenum. The control rod channels' diameter is much larger (10.16 cm) than that of a regular helium coolant channel (1.5875 cm). Although the helium temperature into the control rod channels is comparable to that entering the coolant channels in the VHTR core, the exit temperature is significantly lower. The helium flow through the control rod channels varies with the reactor design, but is typically limited to ~3% of the total helium coolant flow for the reactor [10].

Most reported thermal–hydraulics analyses for the prismatic VHTRs either neglected the helium interstitial bypass flow, or assumed an average bypass flow that is 5–10% of the reactor's total helium coolant flow [4,5]. To the best of the authors' knowledge, the most extensive study of the effect of the helium interstitial bypass flow on the thermal performance of a prismatic VHTR is that of Sato et al. [8]. They analyzed a 1/12th section of a 7.93 m tall stack of prismatic fuel elements under different bypass flow conditions. Their analysis solved for the helium flow distributions in the core coolant channels and the bypass flow passages. Although fairly robust, the method and approach used by Sato et al. [8] could not be easily expanded to the analysis of a larger section of the VHTR core due to the demanding numerical meshing and computation requirements. In addition to the massive VHTR structure, accounting for the helium bypass flow further complicates the thermal–hydraulics analysis and increases the magnitude of the already very demanding computational and analysis task. Thus, there is a need to develop and validate numerical analysis approaches that would simplify the computational task while maintaining accurate results.

Recent results have demonstrated the practicality of replacing a full 3-D, CFD simulation of the helium gas flow in a 10 m tall coolant channel of a prismatic VHTR core with a simplified methodology that thermally couples a 1-D helium flow in the coolant channels to a 3-D heat conduction within the graphite and fuel compacts of the hexagonal fuel elements. In addition to decreasing the memory requirements and the number of elements or cells in the implemented numerical mesh grid [5], this methodology decreases the total computation time to ~2.5% of that required for a full 3-D analysis, without compromising the accuracy of the results (<2% difference) [9].

The simplified thermal–hydraulics analysis methodology provides good predictions of the global parameters such as the 3-D