ISSN 2348-1196 (print)
International Journal of Computer Science and Information Technology Research ISSN 2348-120X (online)
Vol. 8, Issue 4, pp: (50-54), Month: October - December 2020, Available at: www.researchpublish.com

SIMULATION OF COOLANT REYNOLD'S NUMBER AND VELOCITY IN THE CORE OF MINIATURE NEUTRON SOURCE NUCLEAR REACTORS (MNSRs)

Salihu M.[a*], Gambo A.[b], Adamu J.[a2], Bala S [a3]. Abdulrahaman A [a4], Ibrahim A [a5]

[a] Department of Physical Sciences, Niger State Polytechnic, Zungeru, Nigeria

[b] Computer Science Department, Niger State Polytechnic, Zungeru, Nigeria

Corresponding Author's Email: salmedangle@yahoo.co.uk

Abstract: **In the present work, specifications of Ghana Research Reactor 1 (GHARR-1) were adopted for modelling purpose to mimic the MNSR generic design. To ensure consistency in simulated results, GHARR-1 was equally used to experimentally generate data on the inlet and outlet temperatures of the reactor core at varying coolant flow rates for later used to simulate initial conditions of the thermal hydraulic code. The numerical values of flow parameters as contained in GHARR-1 Safety Analysis Report (SAR) were used to further condition the code and also to validate the convergence at the output stage. The model utilized 90% enriched U-235 to produce neutron flux of up to 1x10^{12} cm^{-2}s^{-1} at a corresponding thermal power output of 31 kW. The core coolant modelling and simulation wereachieved using Simulation of Turbulent flow in Arbitrary Regions Computational Continuum Mechanics C++ based (STAR CCM+) Computational Fluid Dynamics (CFD) code. The code successfully analyzed the progression of flow parameters from the inlet to the outlet of the reactor core. Thermal hydraulic studies of the reactor core coolant is necessary for safety of evaluation of a nuclear reactor.**

Keywords: **STAR-CCM+, CFD, GHARR-1, core coolant, flow rates, thermal flux, thermal power, MNSR.**

1. INTRODUCTION

Thermal consideration is major limitation in reactor designs (Mangena, 2016). The MNSRs are relatively low power nuclear reactors that utilized LEU or HEU 235 fuel with an output of 30-34 kW thermal power. The 344 fuel pins, 4 tie and 6 dummy rods that constitute a typical MNSR fuel cage are concentrically arranged in 10 rings. The core has a central guide tube which houses a single cadmium control rod that extends to the bottom of the core. Figure I shows the core configuration of GHARR-1.

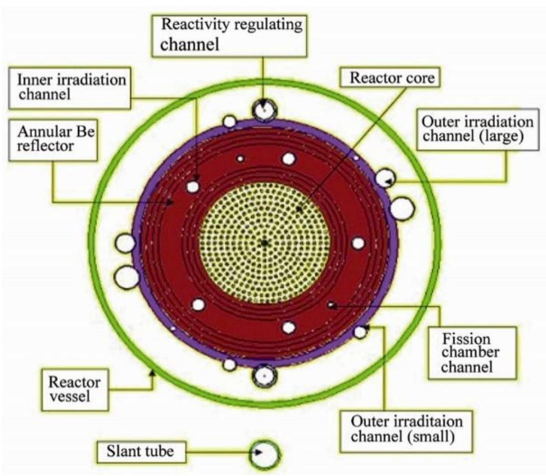

Fig I. MCNP plot of GHARR-1 core configuration

Salihu (2015) and Agbo *et al.* (2015) had reported an extensive analysis of thermal hydraulics of MNSRs. The MNSRs is one of the smallest commercial nuclear reactors in the world with a maximum thermal neutron flux of $1x10^{12}n.cm^{-2}s^{-1}$ (Ahmed *et al.*, 2006; Ampomah-Amoako. 2009). The neutron flux is an important characteristic of nuclear reactors (Nguyen and Do, 2011), studies on GHARR-1 MNSR flux profile is reported by Sogbaji *et al.* (2011). The original prototype of the MNSR was made by the Chinese (Zhou, 1987). The neutron flux is a major factor in experimental analysis of flow parameters (Salihu, 2016). The MNSRs in operation slightly defers from one to another but are of similar design and outputs (SAR-GAEC, 2000; SAR-CERT, 2005).

STAR-CCM+ CFD

STAR-CCM + CFD code is a simulation package based on numerical algorithms and used to solve problems involving fluid flow, heat transfer, turbulence effects and also for stability analysis (CD ADAPCO, 2011; Agbodemegbe, 2015). Details of design procedure adopted for the present study was published (Salihu, 2015)

Understanding the behavior of flow parameters as fluid develops in the core of a nuclear reactor from inlet to outlet is important for operational and safety predictions particularly as the thermal behavior of the reactor core coolant is related to the fuel temperature. Coolant flow behavior is used in assessing the integrity of the core during design consideration.

HEAT TRANSFER IN MNSR

Natural convection, a mechanism for heat removal where fluid motion result from difference in densities arising from temperature gradients as coolant transits from the bottom (inlet) to the top (outlet) of the core, provides the cooling mechanism for MNSR given rise to a single phase flow under normal operating conditions. In GHARR-1, the core is cooled and moderated by light water reactor and as power increases the flow develops from laminar to turbulent (SAR-GAEC, 2000; SAR-CERT, 2005).

EXPERIMENTAL

In the present study, Ghana Research Reactor-1 was used to generate the experimental data which included the inlet and outlet temperatures for 5 kW to 30kW at 5 kW step increase. For each power level considered, mean values of the corresponding inlet and outlet temperatures were recorded and used as inlet temperatures for the simulation conducted at that power level. Details of the experiment al procedure and results are published in Salihu, (2016).

MODELLING AND SIMULATION

For the adopted model, the original dimensions of the core coolant were proportionally scaled down to a tenth from 230 mm to 23 mm. This was necessitated by computational memory limitations and the need to reduce the duration for which convergence of the solution was attained.

In order to account for the varying heat fluxes imposed on the surface of the modelled fluid and also be able to measure bulk and average values of flow parameters of interest, the modelled geometry was segmented into 21 radial segments which was subsequently meshed using models published in Salihu *et al.* (2016)

PHYSICS

At the inlet, mass flow inlet and inlet temperatures were specified. Further, free stream was allowed at the interface between adjoining segments. The walls were imposed varying heat fluxes based on segmental power peaking factor values while the outlet was assigned pressure**.** Summary of the initial conditions and Physics models adopted were as published by Salihu *et al.* (2016).

2. RESULTS AND DISCUSSION

The result of the simulation was in addition to been monitored for convergence, validated using the experimental data. This was done by comparing the outlet temperatures obtained from the simulation to that obtained experimentally; Table I.

ISSN 2348-1196 (print)
International Journal of Computer Science and Information Technology Research ISSN 2348-120X (online)
Vol. 8, Issue 4, pp: (50-54), Month: October - December 2020, Available at: www.researchpublish.com

TABLE I: Experimental and Calculated Results

Power (kW)	T_{in} (^oC)	Exp. T_{out} (^oC)	Calc. T_{out} (^oC)	Difference
5	38.5	42.85	42.22	0.63
10	38.75	46.5	46.31	0.19
15	40.27	50.44	51.87	-1.43
20	43.5	54.76	56.62	-1.86
25	41.24	55.5	58.44	-2.94
30	43.17	58.11	59.14	-1.03

The standard deviations for the Inlet temperature T_{in} (^oC), experimental outlet temperature Exp. T_{out} (^oC) and the calculated outlet temperature Calc. T_{out} (^oC) are ±0.74, ±2.14 and 16.94 respectively.

In the present work, simulations were performed at 5kW to 30kW power for hot and average channels. When compared, the trends of any given pair, the hot and average channels does not differ significantly. The mass flow rate (MFR) were conditioned separately for 0.15, 0.20 and 0.23. Mass flow rates had been previously used as a control parameter to evaluate steady state thermal hydraulic behavior of MNSR core under natural convection cooling. The plots of the corresponding velocities were then obtained as the coolant progresses from the inlet to the outlet. The trend of this segmental variation in the fluids velocity at the centre is presented in Figure II.

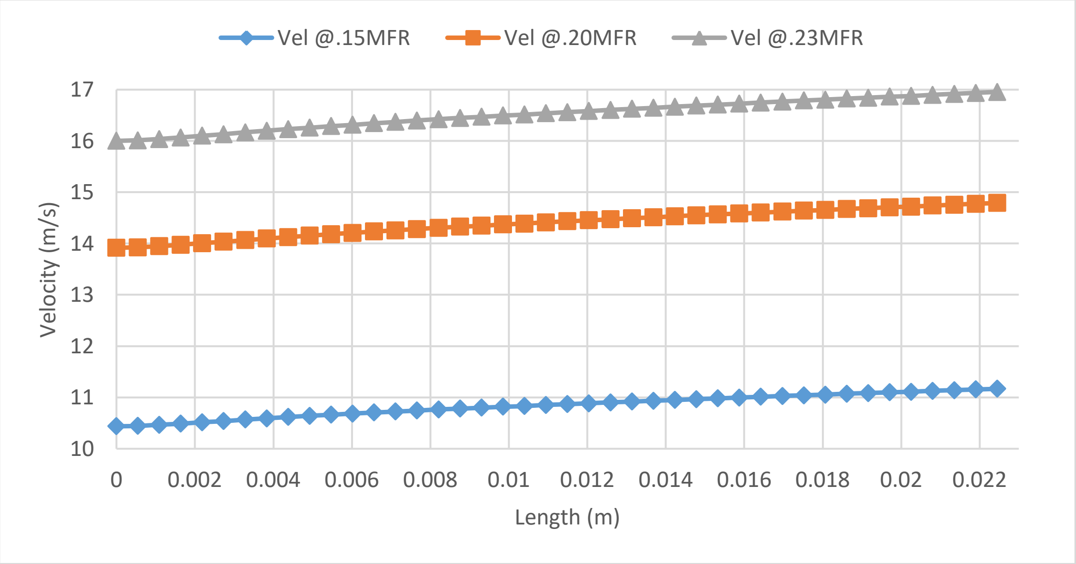

Fig. II. Plot of Segmental Velocity with MFR Control

Trends of Figure II showed that the domain average velocity increases consistently along the channel for a given MFR. Furthermore, that the velocity also varies proportionally with the mass flow rate.

Figure II though a measure of centerline velocity, is in tandem with the volume average velocity taken per segment progressively through the fluid as shown in Figure III.

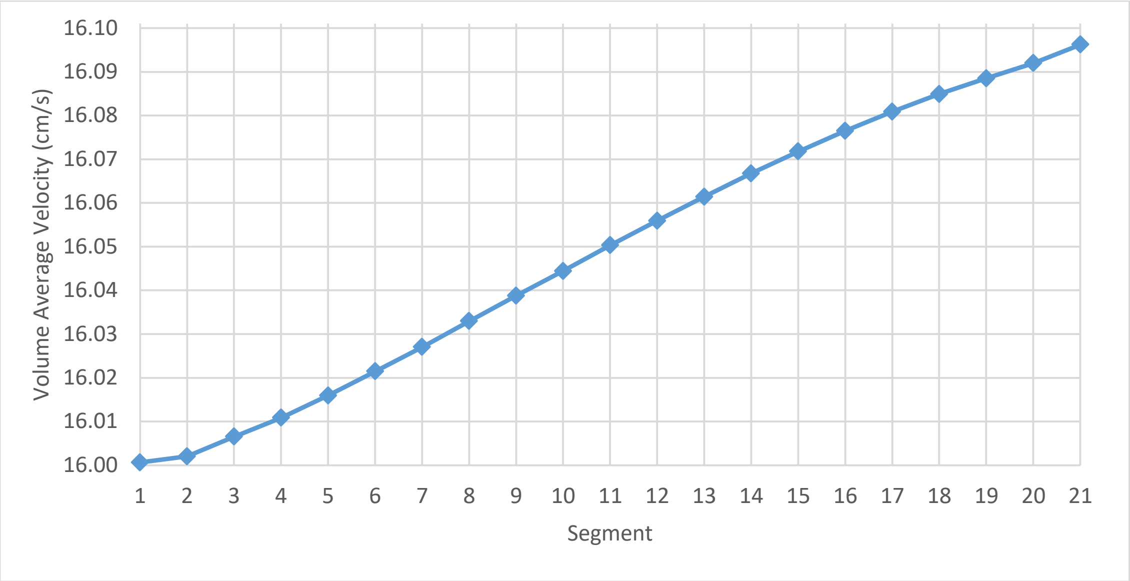

Fig. III. Volume average velocity

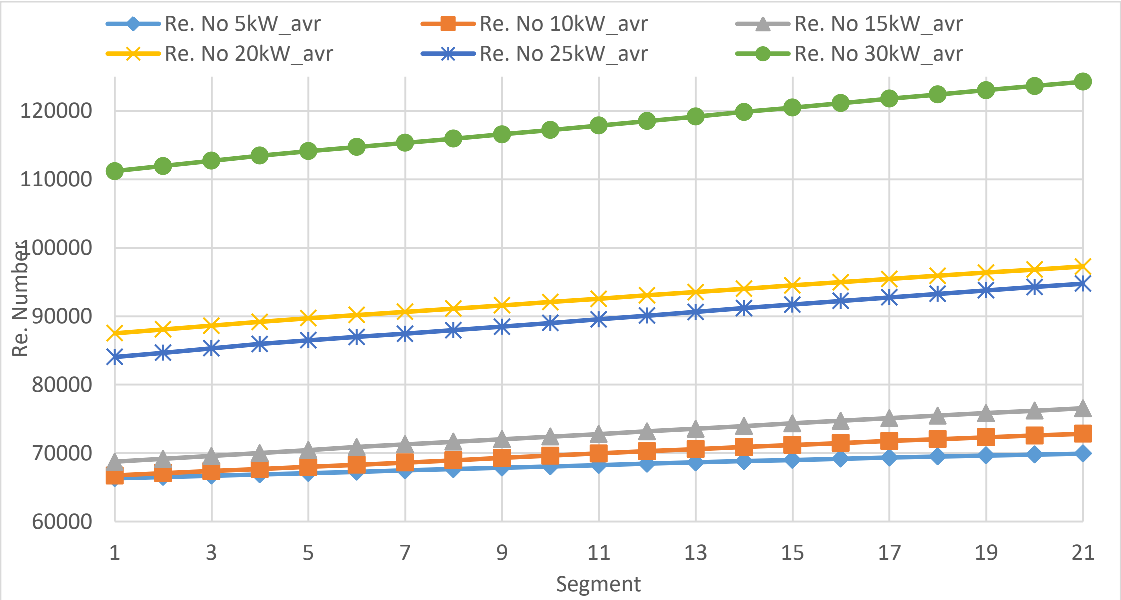

Fig. IV: Segmental Reynold's number

The simulation predicted that the Re. Number increases significantly with increases in power where its value almost doubled between simulations conducted at 5 kW and 30 kW respectively.

3. CONCLUSION

STAR-CCM+ was used to model, simulate and study the core coolant and fluid flow pattern as it affects Coolant Velocity and corresponding Reynolds Number in the core of MNSR. The results obtained were in conformity with those previously measures and calculated as contained in GHARR-1 Safety Analysis Report. STAR_CCM+ CFD Code successfully predicted the distribution of the flow parameters of interest base modelling specifications of MNSRs.

ACKNOWLEDGEMENTS

Developers of STAR-CCM+ (CD-ADAPCO), Niger State Polytechnic Zungeru Nigeria, SNAS University of Ghana, Ghana Atomic Energy Commission, Rijau LGA Community.

REFERENCES

[1] Agbo, S. A., Ahmed, Y. A., Ewa, I. O. B., Abubakar, M., & Anas, M. S. (2015). An experimental testing of coolant flow rate and velocity in the core of Nigeria Research Reactor-1. *International Journal of Nuclear Energy Science and Technology*, *9*(2), 171-185.

[2] Agbodemegbe, V. Y., Cheng, X., Akaho, E. H. K., & Allotey, F. K. A. (2015). Correlation for cross-flow resistance coefficient using STAR-CCM+ simulation data for flow of water through rod bundle supported by spacer grid with split-type mixing vane. *Nuclear Engineering and Design*, *285*, 134-149.

[3] Ahmed, Y. A., Ewa, I. O. B., Umar, I. M., Bezboruah, T., Johri, M., & Akaho, E. H. K. (2006). *The low power miniature neutron source reactors: Design, Safety, and Applications. The Abduls Salam International Centre for Theoretical Physics*. IC/2006/020

[4] Akaho E.H.K., Maakuu B.T., Anim-Sampong S., Emi-Reynolds G., Boadu H.O., Osae E.K., Akoto S. B., Dodoo-Amoo D.N.A. (2003). Ghana Research Reactor-1 Safety Analysis Report. GAEC-NNRI-RT-90

[5] Ampomah-Amoako, E., Akaho, E. H. K., Anim-Sampong, S., & Nyarko, B. J. B. (2009). Transient analysis of Ghana Research Reactor-1 using PARET/ANL thermal–hydraulic code. *Nuclear Engineering and Design*, *239*(11), 2479-2483.

[6] Centre for Energy Research and Training (CERT) (2005) *Final Safety Analysis Report of Nigeria*

[7] Mangena, S. (2016). *Thermal Hydraulic and Safety Analysis of Heat Transfer and Distribution in the Ghana Research Reactor-1 (GHARR-1) core using STAR-CCM+ CFD Code* (Doctoral dissertation, University of Ghana).

[8] Mohammed, S. (2015). *Investigation of heat transfer and distribution in the core of Ghana Research Reactor-1 (GHARR-1) using STAR-CCM+ CFD Code* (Doctoral dissertation, University of Ghana).

[9] Mohammed, S. (2015). *Investigation of heat transfer and distribution in the core of Ghana Research Reactor-1 (GHARR-1) using STAR-CCM+ CFD Code* (Doctoral dissertation, University of Ghana).

[10] Mohammed, S., Agbodemegbe, V. Y., Debrah, S. K., & Ampomah-Amoako, E. HEAT TRANSFER AND DISTRIBUTION IN THE CORE OF GHANA RESEARCH REACTOR-1 (GHARR-1) USING STAR-CCM+ CFD CODE. *ARCA*, 110.

[11] Nguyen, H. H., & Do Quang, B. (2011). Investigating The Neutron Flux Distribution Of The Miniature Neutron Source Reactor MNSR Type. In *Proceedings of the 9 th National Conference on Nuclear Science and Technology*.

[12] *Research Reactor-1 (NIRR-1/ SAR)*, CERT Technical Report-CERT/NIRR-1/FSAR-01.

[13] Salihu M., Musa B. L. (2016). A Study on Pressure Distribution in Miniature Neutron Source Reactor (MNSR) Core Coolant**.** *International Journal of Thermal Technologies*

[14] Sogbadji, R. B. M., Abrefah, R. G., Ampomah-Amoako, E., Agbemava, S. E., & Nyarko, B. J. B. (2011). Neutron energy spectrum flux profile of Ghana's miniature neutron source reactor core. *Annals of Nuclear Energy*, *38*(8), 1787-1792.

[15] STAR-CCM+ User Guide version 6.04.014 CD_ADAPCO 2011.