STUDY OF THERMAL HYDRAULICS PARAMETERS OF TRIGA RESEARCH REACTOR UNDER NATURAL CONVECTION MODE OF COOLANT FLOW USING NCTRIGA COMPUTER CODE

[1]Md Altaf Hossen And [2]Mohammad Jahirul Haque Khan

[1]Reactor Physics and Engineering Division, Institute of Nuclear Science and Technology, Bangladesh Atomic Energy Commission, Ganakbari, Savar, Dhaka-1349, Bangladesh
[2]Director and Chief Scientific Officer, Institute of Nuclear Science and Technology, Bangladesh Atomic Energy Commission, Ganakbari, Savar, Dhaka-1349, Bangladesh
e-mail:[1]altaf335@yahoo.com, [2]jhk_1970@yahoo.com

ABSTRACT

A thermal-hydraulic analysis of the TRIGA Mark II Research Reactor operating at 500 kW under natural convection conditions was conducted using the NCTRIGA code utilizing SRAC2006 neutronics data, and results were benchmarked against MCNP4C calculations. The study focused on evaluating key thermal-hydraulic parameters along the axial length of the hottest fuel rod (C4) to ensure the reactor's safety. The Reynolds number exhibited a consistent increase with axial height. Conversely, heat flux, heat transfer coefficient, and fuel centerline temperature demonstrated a similar trend: increasing from the top of the core, peaking near the midpoint, and subsequently decreasing. Notably, fuel centerline temperatures remained significantly below established safety limits. Fuel surface temperatures remained relatively constant, while coolant temperature demonstrated a slow, incremental increase along the axial length from the top. While minor discrepancies were observed between the SRAC2006 and MCNP4C datasets, the peak values and their locations remained consistent across both.

ABSTRAK

Analisis terma-hidraulik bagi Reaktor Penyelidikan TRIGA Mark II yang beroperasi pada 500 kW di bawah keadaan perolakan semula jadi telah dijalankan menggunakan kod NCTRIGA yang menggunakan data neutronics SRAC2006, dan keputusan telah ditanda aras terhadap pengiraan MCNP4C. Kajian ini memberi tumpuan kepada menilai parameter terma-hidraulik utama sepanjang panjang paksi rod bahan api terpanas (C4) untuk memastikan keselamatan reaktor. Nombor Reynolds menunjukkan peningkatan yang konsisten dengan ketinggian paksi. Sebaliknya, fluks haba, pekali pemindahan haba, dan suhu garis tengah bahan api menunjukkan arah aliran yang sama: meningkat dari atas teras, memuncak berhampiran titik tengah, dan seterusnya menurun. Terutama, suhu garis tengah bahan api kekal jauh di bawah had keselamatan yang ditetapkan. Suhu permukaan bahan api kekal secara relatifnya tetap, manakala suhu penyejuk menunjukkan peningkatan yang perlahan dan bertambah sepanjang paksi dari atas. Walaupun percanggahan kecil diperhatikan antara set data SRAC2006 dan MCNP4C, nilai puncak dan lokasinya kekal konsisten di kedua-duanya.

Keywords: TRIGA, NTRIGA, Thermal hydraulics, Safety

INTRODUCTION

The 3MW TRIGA MARK II research reactor, located in Savar near Dhaka, Bangladesh, was commissioned in late 1986. The reactor is designed for multi-purpose uses, such as training, education, radioisotope production, and various R&D activities in neutron activation analysis, neutron scattering and neutron radiography [1]. This study aims to analyze the essential steady-state thermal-hydraulic parameters of the reactor when it operates with natural convection coolant flow.

The TRIGA reactor core, characterized by its hexagonal array of 100 Er-UZrH fuel elements within a shroud, is engineered for both continuous 3000 kW (thermal) and pulsed operation. The solid, homogeneous fuel composition, comprising 20% uranium (enriched to 19.7% ^{235}U) and 0.47% erbium, facilitates a stable and controllable nuclear reaction. Designed for continuous thermal operation at 3000 kW, the reactor also accommodates routine pulsing up to 1.4% δk/k ($2.00). While natural cooling suffices for operation up to 500 kW, higher power levels necessitate a forced downward flow of light water to ensure effective heat transfer, complemented by graphite reflection to optimize neutron economy. Figure 1 below represents TRIGA MARK II reactor core configuration.

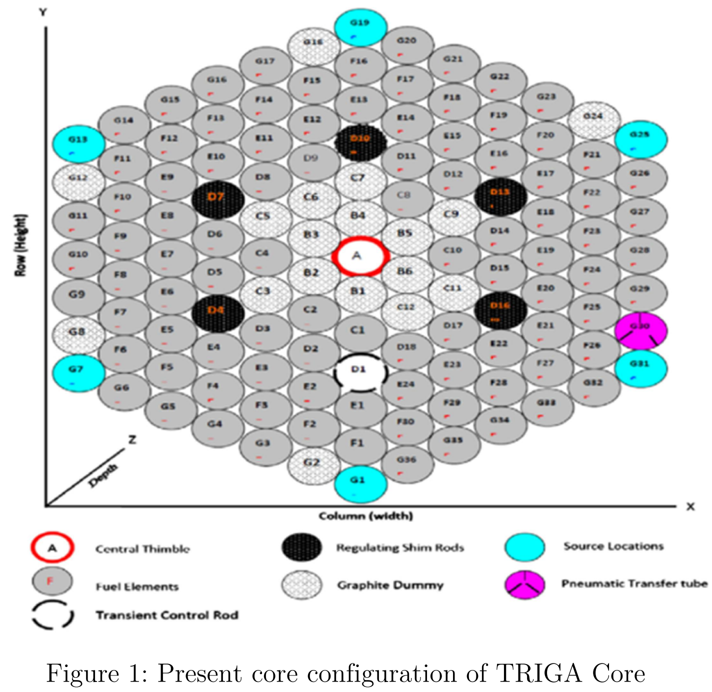

Figure 1: Present core configuration of TRIGA Core

In the natural convection-operating mode, the reactor itself supplies the hydraulics driving force as it transfers heat to the coolant, creating a buoyant head. Counteracting this force are contraction and expansion losses at the entrance and exists to the channel plus the acceleration and potential energy losses and friction losses in the coolant channel itself. Since each flow channel provides its driving force, it is possible to consider flow channels independently. Since the hot channel is the most important one for thermal-hydraulic analysis, the hottest fuel rod and its associated flow channel are considered in this analysis of thermal-hydraulic safety studies. Edge and corner channels are characterized by lower power densities, larger flow areas, and hence, lower heat fluxes and coolant temperatures.

The operational power of natural convection cooling systems is constrained to a maximum of 500 kW. This limitation is predicated on two primary factors: radiation exposure and thermal management. Exceeding this threshold precipitates elevated dose rates at the pool's surface, primarily attributed to the rapid transport of Nitrogen-16 (^{N16}) via natural convection currents. The swift circulation of coolant facilitates the migration of N^{16} to the surface, resulting in increased radiation levels that pose a risk to operational safety. Furthermore, at a power dissipation of 500 kW, the pool water temperature increases at a rate of 20°C per hour (for a 6000-gallon pool). This rapid thermal increase presents significant challenges in maintaining stable operating conditions

Previous study, utilizing the NCTRIGA computer code [2], have analyzed the thermal-hydraulic characteristics of TRIGA research reactors [3]. In this simulation, neutronic parameters, specifically axial power distribution and power peaking factors, derived from the Monte Carlo code MCNP4C [4], were incorporated into the NCTRIGA model to represent the reactor's core characteristics and determine hot spot locations, thereby enhancing the accuracy of the thermal-hydraulic analysis. However, no further study has validated these results. Hence, this essay highlights the necessity for further research using the deterministic code SRAC2006 [5] to validate the earlier neutronics data and thermal hydraulics work done by MCNP4C and NCTRIGA, thereby ensuring the continued safety and operational integrity of these cooling systems.

MATERIALS AND METHOD

Brief Description of NCTRIGA Code

NCTRIGA is a one-dimensional thermal-hydraulic code designed for the analysis of fuel rod channels. It computes temperatures at the fuel centerline, fuel surface, and within the coolant at discrete nodes along a single fuel rod. Notably, it also predicts the coolant flow rate induced by natural convection. The code leverages power distribution data derived from neutronic analyses, in conjunction with geometric and material composition data of the channel under consideration, to perform temperature calculations and flow rate predictions. NCTRIGA is based on the NATCON code developed at Argonne National Laboratory, with modifications implemented to specifically model reactors utilizing TRIGA-type fuel [6]. Validation of NCTRIGA was conducted at Argonne National Laboratory using empirical data obtained by General Atomic of USA. The data presented in Table 1 demonstrates the code's capacity to generate plausible temperature and flow rate estimations when applied to TRIGA cores.

Table 1: Comparison of NCTRIGA results to experimental data

Power, MW	No. of Elements	Source	Flow rate kg/sec	Error %	Outlet Temp (°C)	Error %
1.0	91	GA	8457	-7.0	70.2	-10.0
		NCTRIGA	7890		67.2	
1.5	91	GA	9555	-3.2	76.6	-2.6
		NCTRIGA	9259	-0.3	75.6	
2.0	101	GA	11080		86.1	-14.0
		NCTRIGA	11051		80.1	

Several modifications enhance NCTRIGA's flexibility. User-defined overrides for thermal conductivity and axial power distribution broaden its applicability. The code's validity extends beyond the incipient boiling point, provided the coolant remains subcooled, crucial for plate-type fuel reactors. For TRIGA reactors, which exhibit self-regulation, the code resets the wall temperature to the incipient boiling temperature upon exceedance, recalculating the heat transfer coefficient and indicating this adjustment with an asterisk. These features collectively improve the code's predictive accuracy and efficiency.

NCTRIGA Modelling of TRIGA

NCTRIGA serves as a tool for determining steady-state thermal-hydraulic parameters essential for establishing limiting safety system settings within a reactor. Characterized by its concise input requirements, NCTRIGA models a single fuel rod, necessitating only the total core power and number of fuel elements, thereby obviating the need for comprehensive core-wide data. The code requires fundamental data pertaining to fuel geometry, material characteristics, axial power distribution, and power reactor, specifically concerning the fuel rod under analysis. As illustrated in Figure 2, the code incorporates a geometric representation of the channel, facilitating accurate modelling of thermal-hydraulic behaviour.

NCTRIGA, a nuclear reactor analysis code, necessitates the input of power distribution at the ends of each axial region, as shown in Figure 3. The hot rod factor, representing the ratio of the power of the hottest rod to the average power of the core, was calculated to be 1.854980 for SRAC2006 [7], in comparison to 1.8540 for MCNP4C. Additionally, the code calculated axial power distributions where the axial peak to average ratio of power at the C4 rod was found to be 1.2204 [7], compared to that for MCNP4C of 1.21 [3]. Axial power peaking factors calculated by SRAC2006 and MCNP4C is shown in Table 2. These neutronic parameters are integrated within NCTRIGA to compute the reactor's thermal-hydraulic parameters.

Table 2: Comparison of axial power peaking factors

Axial Position(cm)	SRAC2006	MCNP4C
0	0.667	0.6674
2.53	0.7278	0.7282
5.07	0.8373	0.8377
7.60	0.9525	0.9530
10.13	1.0521	1.0527
12.67	1.1283	1.1289
15.20	1.181	1.1816
17.73	1.2116	1.2122
20.27	1.2204	1.2210
22.80	1.2057	1.2063
25.33	1.1633	1.1639
27.87	1.0897	1.0903
30.40	0.9853	0.9858
32.93	0.8609	0.8614
35.47	0.745	0.7454
38.00	0.6928	0.6932

The code uses a geometric representation of the channel to accurately model thermal-hydraulic behaviour. The radial configuration of the fuel channel, as modelled within NCTRIGA, exhibits a cylindrical fuel rod of 0.0182245 meter of radius. This rod is surrounded by its conductance, which, within the model, lacks a defined physical thickness. The pitch-to-diameter ratio of 1.217755426 provides the required data for the determination of the water channel dimensions. The radial geometric arrangement is visually depicted in Figure 4, with the parameters essential for input preparation summarized in Table 3. This simplified radial geometry is crucial for efficient thermal-hydraulic analysis within the NCTRIGA framework.

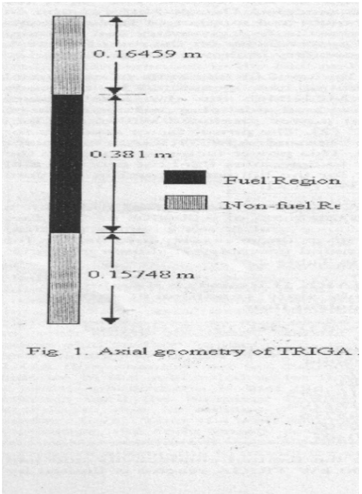

Figure 2: Axial geometry of TRIGA fuel

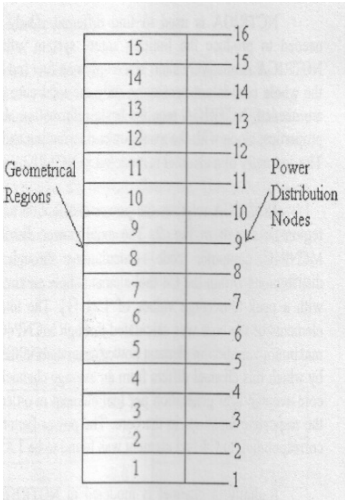

Figure 3: Axial regions for NCTRIGA

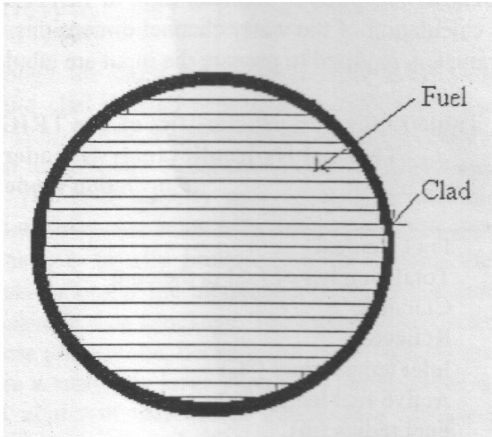

Figure 4: Radial geometry for NCTRIGA

Table 3: Some characteristics of the TRIGA Mark II Research Reactor for thermal-hydraulic analysis under steady state conditions at natural convection mode of coolant flow

Fuel Element	20 w/o, U-ZrH, 19.7% enriched
Total Number of Fuel In the Core	100
Cladding	Stainless Steel 304L
Reflector	Graphite
Inlet Temperature (^{o}C)	40.6
Active Fuel Length	0.381
Pitch (m)	0.0457716
Fuel Radius (m)	0.0182245
Clad Outer Radius (m)	0.0187706
Coolant Channel Height (m)	0.70307
Pool Depth (m)	7.10307

Accurate prediction of fuel temperature and related physical properties holds paramount importance in the operation of the NCTRIGA reactor. The steady-state power rating of the reactor is inherently constrained by the maximum allowable temperatures of both the fuel and cladding, in addition to the critical heat flux. A study was conducted to compare the effectiveness of SRAC2006 data against previous findings obtained using MCNP4C data. To ensure a direct comparison, key physio-material properties, including mass velocity, coolant velocity, fuel thermal conductivity, and gap conductance, were maintained at constant values, consistent with those utilized in the prior NCTRIGA study. This approach allows for a focused evaluation of the impact of the different datasets on the accuracy of fuel temperature predictions.

RESULTS AND DISCUSSION

The thermal-hydraulic design of the reactor core incorporates considerations for extreme operational scenarios. An inlet temperature of 40.6°C, representing the highest anticipated ambient temperature, has a negligible impact on maximum fuel temperature due to the predicted onset of surface boiling in the hottest fuel elements. Furthermore, the potential blockage of grid plate openings by foreign objects is mitigated by the open design below the grid plates, facilitating lateral crossflow. This design ensures that blocked elements experience minimal cooling loss, with the primary consequence being a slight increase in core pressure drop. These design features contribute to the robustness and safety of the reactor's cooling system.

The Reynolds number, a key dimensionless parameter, is vital for distinguishing fluid flow regimes in thermal-hydraulic systems, particularly within nuclear reactors. Accurate determination of this number is fundamental to ensuring reliable safety analyses and performance predictions. This research employed the NCTRIGA code to calculate the Reynolds number using neutronics data from SRAC2006, and these results were compared to those obtained with MCNP4C codes. Calculated Reynolds Numbers are presented in Figure 5. Findings revealed a consistent axial increase in Reynolds number across both datasets, though SRAC2006 data tended to slightly overestimate values relative to MCNP4C throughout the reactor core.

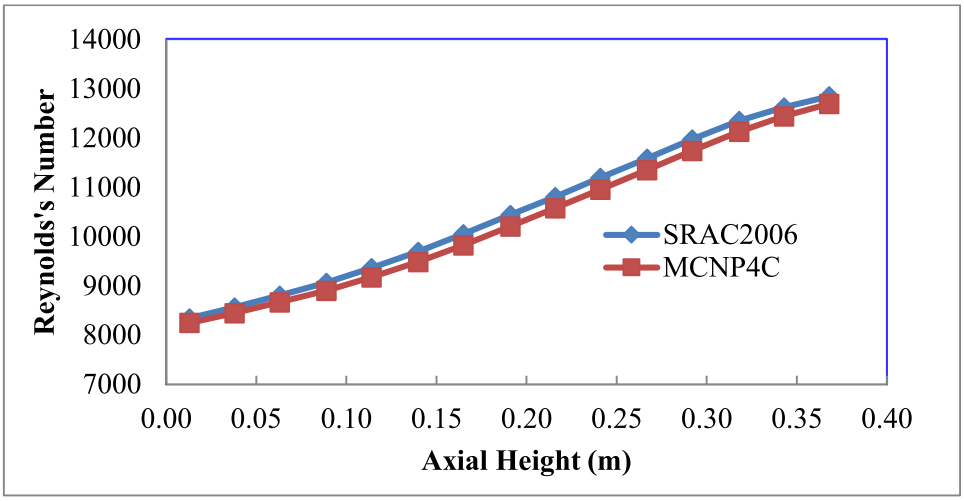

Figure 5: Axial distribution of Reynold's Number

Further analysis of the heat transfer coefficient along a nuclear reactor rod showed comparable axial profiles between the two codes. SRAC2006 results ranged from approximately 1850 to 2950 W/m², with a peak near 4026 W/m², while MCNP4C data displayed a slightly broader range, spanning from 1740.4 to 3423.2 W/m², with a peak value of 4009 W/m². Both profiles peaked just below the rod's midpoint and exhibited nearly identical curve shapes, indicating strong agreement in heat transfer predictions. Heat transfer coefficient calculated by NCTRIGA for two data sets are represented in Figure 6.

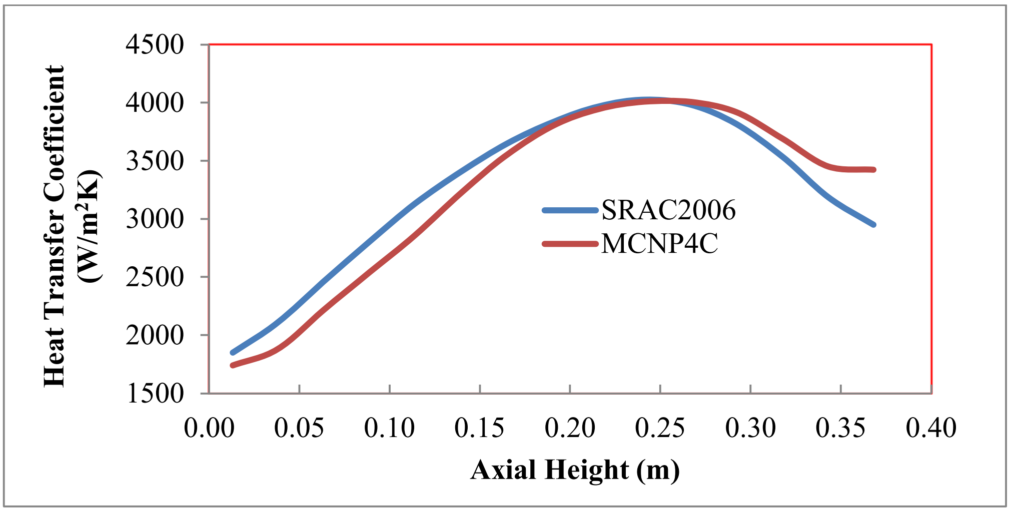

Figure 6: Heat transfer coefficient along axial direction of the hottest rod

The axial heat flux distribution of the hottest fuel rod (C4) followed a cosine pattern for both neutronics data sets. SRAC2006 values varied narrowly around 1.44 to 1.49 × 10^5, whereas MCNP4C covered a wider range, spanning from 1.35 × 10^5 to 1.74 × 10^5, yet both reached peak heat flux values around 2.52 × 10^5, showing concordance in thermal behavior. Figure 7 shows heat flux distribution simulated by SRAC2006 and MCNP4C data.

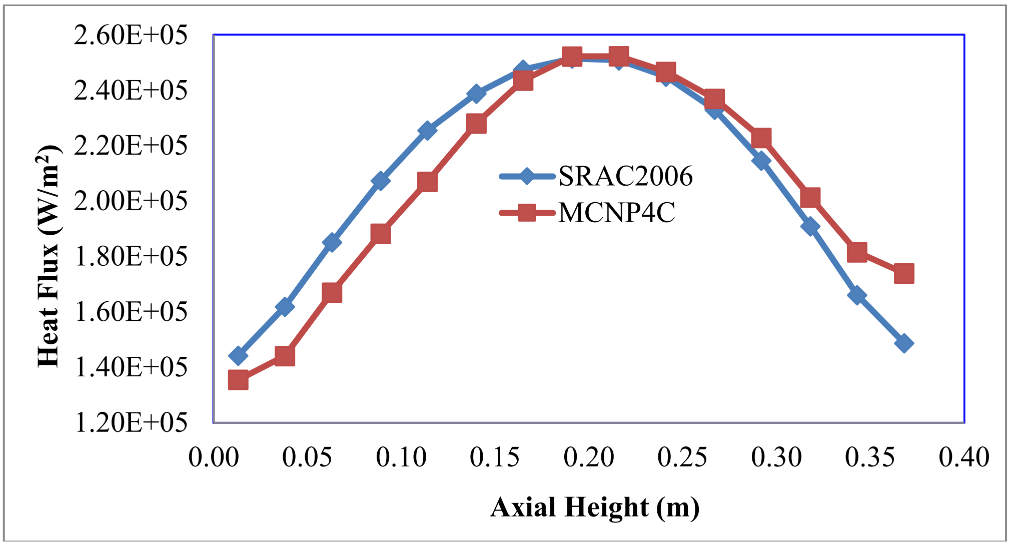

Figure 7: Heat flux along the axial direction of the hottest rod

Finally, temperature profiles of the hottest fuel rod (C4) were examined. The fuel temperature displayed a cosine distribution, peaking centrally with negligible differences between SRAC2006 (268.72 °C) and MCNP4C (269.15 °C). It is evident from the figure that fuel centerline temperatures remained significantly below established safety limits [8]. Fuel surface temperatures remained stable axially, with maximum values near 120.13 °C for both datasets. Coolant temperature rose steadily along the axial length, starting around 40.6 °C and reaching approximately 68 °C for SRAC2006 and 68.16 °C for MCNP4C data. Axial temperature distribution is presented in Figure 8.

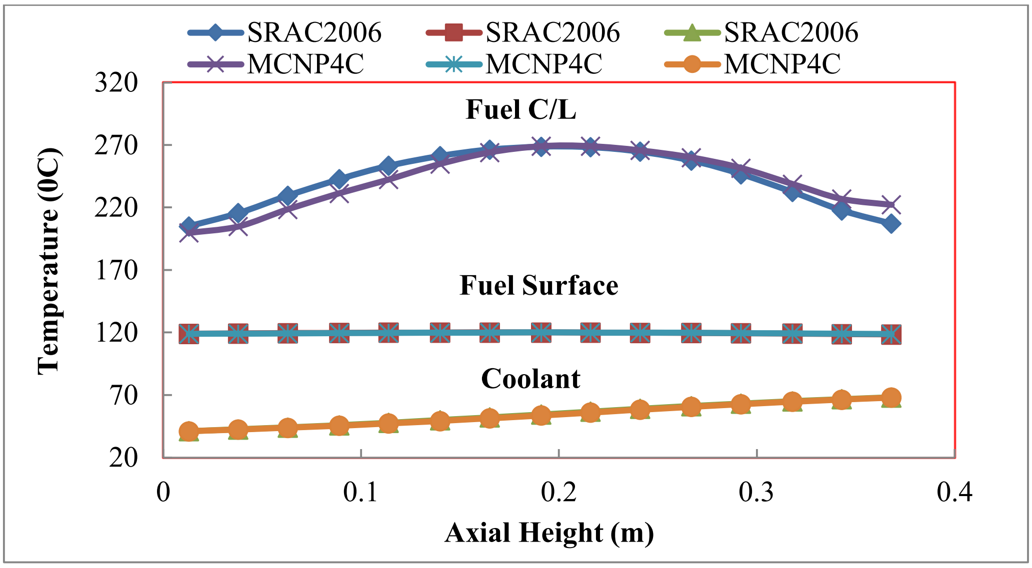

Figure 8: Axial temperature distribution

In summary, the comparative study underscores that despite minor variations, SRAC2006 and MCNP4C neutronics data yield highly consistent results for Reynolds number, heat transfer coefficient, heat flux, and temperature distributions, reinforcing confidence in their application for nuclear reactor thermal-hydraulic assessments.

CONCLUSION

The utilization of MCNP4C-NCTRIGA codes offers an effective approach for steady-state thermal-hydraulic analysis of the TRIGA MARK II research reactor, specifically under natural convection coolant flow conditions. The Reynolds number serves as a vital dimensionless parameter in defining fluid flow regimes within thermal-hydraulic systems, thereby underpinning reliable safety assessments and performance evaluations of nuclear reactors. A comparative study employing the NCTRIGA code integrated with neutronics data from SRAC2006 and MCNP4C codes revealed consistent axial trends in the Reynolds number, though SRAC2006 slightly overestimated values relative to MCNP4C. Furthermore, analyses of heat transfer coefficients, heat flux, and temperature distributions in the hottest reactor fuel rod demonstrated strong agreement between both datasets, with minor variations in peak values and spatial profiles. These findings validate the efficacy of SRAC2006 neutronics data in predicting key thermal-hydraulic parameters comparable to the established MCNP4C results. Hence, the combination of the SRAC2006 data set and NCTRIGA can be considered to be reliable for studying thermal hydraulics parameters.

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