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Research Article

Study of Thermal Hydraulics Parameters of TRIGA Research Reactor under Natural Convection Mode of Coolant Flow using NCTRIGA Computer Code

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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 the 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 reactor safety. The Reynolds number increased consistently 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.

Keywords: TRIGA, NCTRIGA, Thermal hydraulics, Safety

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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% 235U) 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 the reactivity of 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. Fig. 1 represents the TRIGA MARK II reactor core configuration.

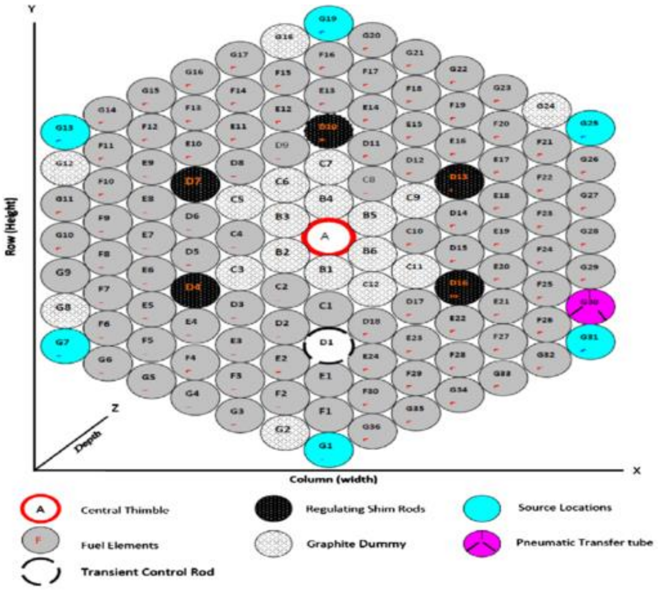

Fig.1 *Present core configuration of TRIGA Core*

In the natural convection-operating mode, the reactor itself supplies the hydraulic 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 exit 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 the 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 (16N) via natural convection currents. The swift circulation of coolant facilitates the migration of ^{16}N 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 studies on the thermal-hydraulic behaviour of the TRIGA Mark II reactor under natural convection cooling using the NCTRIGA code [2] have been reported in the literature [3, 4].
In addition, neutronic analysis using the SRAC code system has also been investigated in earlier work [5].In this simulation, neutronic parameters, specifically axial power distribution and power peaking factors, taken from the MCNP modeling report of the TRIGA Mark II reactor [6] and the Monte Carlo code MCNP4C [7], were incorporated into the NCTRIGA model to represent the reactor's core characteristics and determine the 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 [8] 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.

2.Materials and Method

2.1 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 [9]. Validation of NCTRIGA was conducted at Argonne National Laboratory using empirical data obtained from General Atomic of the 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/s)	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, recalculates the heat transfer coefficient and indicates this adjustment with an asterisk. These features collectively improve the code's predictive accuracy and efficiency.

2.2 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 the number of fuel elements, thereby obviating the need for comprehensive core-wide data. The code requires fundamental data pertaining to the fuel geometry, material characteristics, axial power distribution, and power reactor, specifically concerning the fuel rod under analysis. As illustrated in Fig. 2, the code incorporates a geometric representation of the channel, facilitating accurate modeling 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 Fig. 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 [10], compared with1.8540 for MCNP4C. Additionally, the code calculated axial power distributions and the axial peak-to–average-power ratio of at the C4 rod was found to be 1.2204 [10], compared to 1.21 that obtained from MCNP4C [6]. Axial power peaking factor calculated by SRAC2006 and MCNP4C is shown in Table 2 and Fig. 2. These neutronic parameters are integrated within NCTRIGA to compute the reactor's thermal-hydraulic parameters.

Table 2. Comparison of axial power peaking factor

Axial Position (cm)	SRAC2006	MCNP4C
0	0.6670	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	1.1810	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	0.7450	0.7454
38.00	0.6928	0.6932

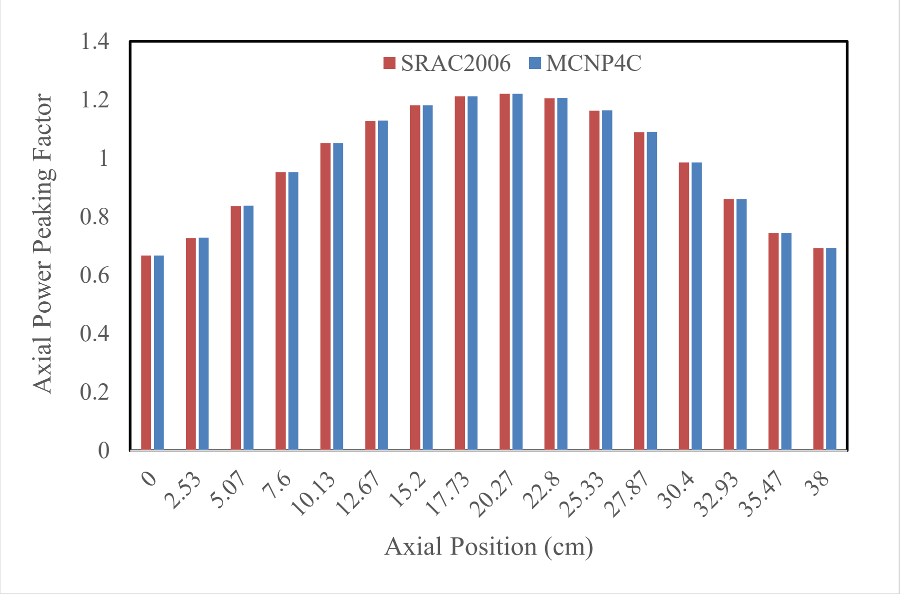

***Fig. 2** Comparison of axial power peaking factor*

The code uses a geometric representation of the channel to model thermal-hydraulic behavior accurately. The radial configuration of the fuel channel, as modeled within NCTRIGA, exhibits a cylindrical fuel rod having a radius of 0.01823 meters. This rod is surrounded by its conductance, which, within the model, lacks a defined physical thickness. The pitch-to-diameter ratio of 1.21776 provides the required data in determining the water channel dimensions. The axial and radial geometric arrangement is visually depicted in Figs. 3-5, and the main reactor parameters used as input to the code are summarized in Table 3, which are based on the standard TRIGA Mark II reactor data reported in previous NCTRIGA analyses [3,4]. This simplified radial geometry is crucial for the efficient thermal-hydraulic analysis within the NCTRIGA framework.

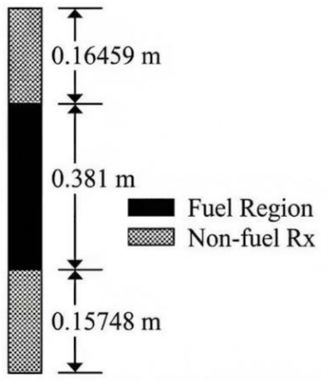

Figure 3: Axial geometry of TRIGA fuel

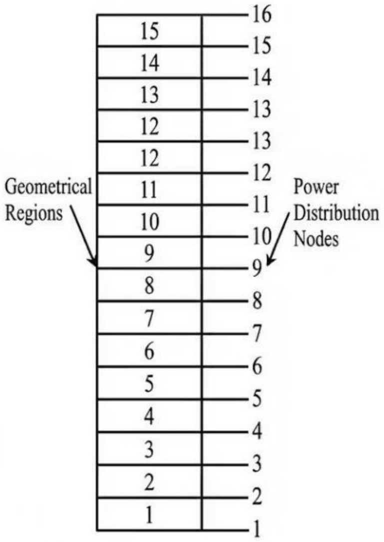

Figure 4: Axial regions for NCTRIGA

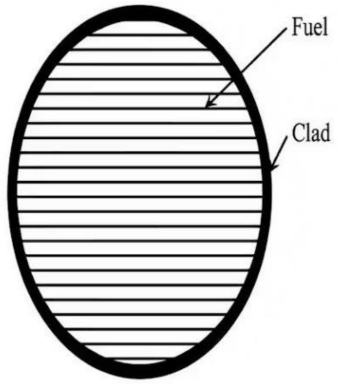

Figure 5: Radial geometry for NCTRIGA

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 (°C)	40.6
Active Fuel meter	0.381
Pitch (m)	0.04577
Fuel Radius (m)	0.01823
Clad Outer Radius (m)	0.01877
Coolant Channel Height (m)	0.7031
Pool Depth (m)	7.1031

Accurate prediction of fuel temperature and related physical properties is 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, the key physical and material properties, including mass flow rate, 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.

3. 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 the maximum fuel temperature due to the predicted onset of surface boiling in the hottest fuel elements. Furthermore, the potential blockage of the grid plate openings by foreign objects is mitigated by the open design below the grid plates, facilitating lateral cross-flow. This design ensures that blocked elements experience minimal cooling loss, with the primary consequence being a slight increase in the 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 the~~se~~ results were compared to those obtained with the MCNP4C code~~s~~. Calculated Reynolds Numbers are presented in Fig. 6. 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. Although the Reynolds number exceeds the classical transition value ($Re \approx 4000$), the flow remains in the natural convection regime. The elevated values reflect buoyancy-driven flow along the hottest fuel rod rather than forced turbulence, with the SRAC2006-based calculations slightly higher due to marginally higher axial power, which produces stronger local buoyancy. These differences are small (<2%) and do not affect the overall thermal-hydraulic safety assessment, confirming that both datasets provide consistent and reliable trends for reactor analysis.

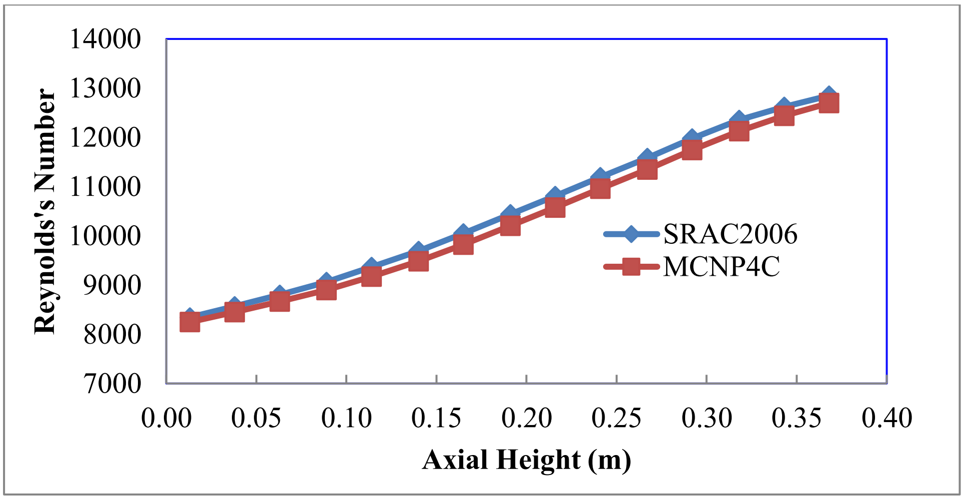

Fig.6 *Axial distribution of Reynold's Number*

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·K, with a peak near 4026 W/m²·K, while MCNP4C data displayed a slightly broader range, spanning from 1740.4 to 3423.2 W/m²·K, with a peak value of 4009 W/m²·K. 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 Fig. 7.

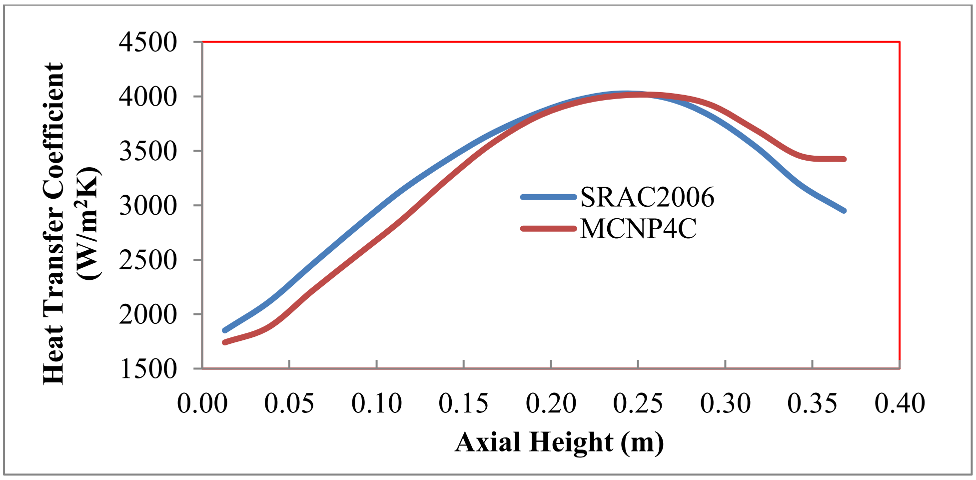

Fig.7 *Heat transfer coefficient along axial direction of the hottest rod*

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 x 10^5 to 1.49 × 10^5 W/m², whereas MCNP4C covered a wider range, spanning from 1.35 × 10^5 to 1.74 × 10^5 W/m², yet both reached peak heat flux values around 2.52 × 10^5 W/m², showing concordance in ~~the~~ thermal behavior. Fig. 8 shows the heat-flux distribution simulated by SRAC2006 and MCNP4C data.

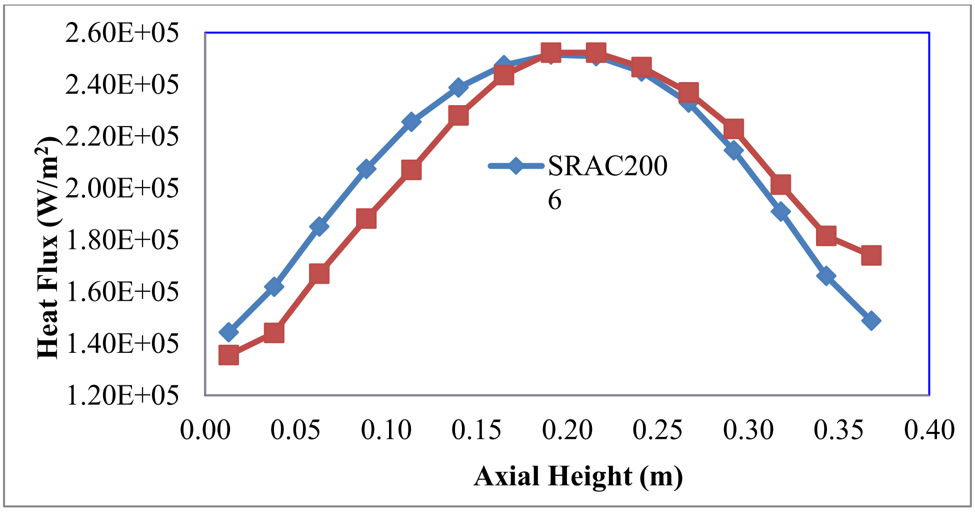

Fig.8 *Heat flux along the axial direction of the hottest rod*

Finally, the temperature profiles of the hottest fuel rod (C4) were examined in Fig. 9. 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 Fig. 9 that fuel centerline temperatures remained significantly below the established safety limits [11]. 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 the SRAC2006 data and 68.16 °C for the MCNP4C data. Axial temperature distribution is presented in Fig. 9.

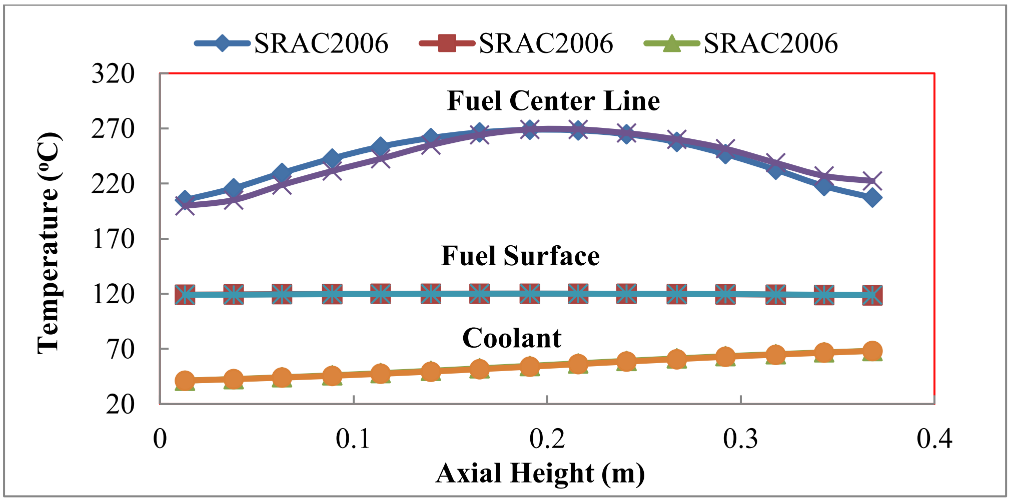

Fig.9 *Axial temperature distribution*

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.

4. Conclusion

The present study demonstrates that the thermal-hydraulic parameters of the TRIGA Mark II research reactor operating at 500 kW under natural convection conditions can be reliably predicted using the NCTRIGA code with neutronics input from both SRAC2006 and MCNP4C. A quantitative comparison shows very close agreement between the two datasets. The axial power peaking factor differed by less than 0.1%, while the maximum fuel centerline temperature predicted by SRAC2006 (268.72 °C) and MCNP4C (269.15 °C) differed by less than 0.2%. Similarly, peak heat transfer coefficients (4026 $W/m^2 \cdot K$ for SRAC2006 and 4009 $W/m^2 \cdot K$ for MCNP4C) and coolant outlet temperatures (~68 °C for both cases) indicate deviations within a small margin, confirming the consistency of the two neutronics inputs.

These results demonstrate that SRAC2006-based neutronics data can reproduce the thermal-hydraulic behavior of the hottest fuel channel with accuracy comparable to the widely used MCNP4C results reported in earlier thermal-hydraulic studies of the TRIGA Mark II reactor [3,4]. From a practical point of view, this agreement supports the use of SRAC2006 together with NCTRIGA for routine reactor safety analysis, operational evaluation, and future thermal-hydraulic studies of the TRIGA reactor under natural convection mode. The confirmed consistency between deterministic and Monte Carlo neutronics inputs increases confidence in safety margin predictions. It provides a reliable basis for future reactor analysis and possible core modifications, consistent with previous neutronic and thermal-hydraulic investigations reported in the literature [3, 4, 5].

Acknowledgement

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Conflict of Interest

All the authors declare no conflict of interest.

Ethical Review and Approval

We, the authors, confirm that this manuscript, submitted for consideration and possible publication in this journal, does not require ethics committee review or approval

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