TURKISH ENERGY, NUCLEAR AND MINERAL RESEARCH AGENCY
TURKISH JOURNAL OF NUCLEAR SCIENCES
E-ISSN: 2791-7185
https://dergipark.org.tr/tr/pub/tjns

Thermal Hydraulics Safety Analysis at Steady State Operation of Burnup Core of TRIGA Mark II Research Reactor Utilizing PARET/ANL and COOLOD-N2

Md Altaf Hossen[1,*]

[1] **Reactor Physics and Engineering Division, Bangladesh Atomic Energy Commission, Ganakbari, Savar, Dhaka-1349, Bangladesh**

ARTICLE INFO

Article history:
Received November 25, 2024
Available online December 31, 2024

Research Article

Keywords:
- COOLOD-N2
- PARET/ANL
- Thermal hydraulics

ABSTRACT

Steady-state thermal hydraulics analysis of 700 MWD burned core of the TRIGA Mark II research reactor has been studied with the computational codes PARET/ANL and COOLOD-N2. This study aims to ensure the safety of burned TRIGA core by executing steady-state thermal-hydraulics calculations at full power. Fuel centerline temperature, fuel surface temperature, fuel clad temperature, DNB heat flux, and DNBR of the hottest rod, enthalpy, and coolant temperature have been calculated. Safety parameters from both codes align well, consistently maintaining margins significantly below the safety limits. Hence, PARET/ANL data from the simulation can be utilized for the transient study and core management of the reactor.

1. Introduction

Bangladesh Atomic Energy Commission has been operating TRIGA Mark II research reactor since its commission in 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]. The reactor, characterized by being light water-cooled and graphite-reflected, is specifically engineered for continuous operation at a constant power level of 3000 kW (thermal). An exceptional property of the TRIGA reactor lies in its well-demonstrated safety measures, primarily attributed to the substantial prompt negative temperature coefficient of reactivity inherent in its U-ZrH fuel-moderator composition. The reactor core comprises of 100 fuel elements, which include 5 fuel follower control rods and 2 instrumented fuel elements. The concentric hexagonal array design within the core shroud allows for efficient fuel distribution and optimal utilization of space in nuclear reactors. It is capable of functioning in both steady-state and pulsing modes, accommodating natural convection as well as forced convection cooling methods. Notably, the natural convection mode remains efficacious up to 500 KW, following which the transition to forced convection mode becomes imperative.

The prime goals of thermal hydraulics are to efficiently release the heat produced in the fuel without raising fuel temperatures too high or creating steam voids, and without getting too near to the hydrodynamic critical heat flux under steady-state operating circumstances. Since fission neutrons directly generate the reactor core's heat energy, there is a significant link between neutronic and thermal examination of the core. Hence, as the core undergoes burnup or any form of rearrangement, the power peaking factor of fuel rods and neutron flux distribution experience alterations, leading to changes in thermal hydraulics parameters like fuel centerline temperature and DNBR (Departure from Nucleate Boiling defined as the ratio of the critical heat flux to the heat flux achieved in the core). Therefore, it is imperative to analyse thermal hydraulics safety parameters in conjunction with burnup. The main purpose of conducting a thermal-hydraulic core analysis is to verify that the operating temperatures inside the core do not reach the design limit. By assuring that the greatest temperature detected in any fuel rod stays below the core design threshold, it may be fairly believed that the temperatures of the other fuel rods will similarly fall below acceptable ranges. Therefore, significant focus is put on monitoring whether the temperature of the hottest rod stays below the prescribed design threshold. Several algorithms have been used to calculate the thermal-hydraulic characteristics of the TRIGA Mark II research reactor in both steady-state and transient-state operations so far [2,3]. EUREKA has previously performed a simulation of the 700 MWD burned core of the TRIGA thermal hydraulics [4], using the Sudo-Kaminaga correlation for critical heat flux, the Dittus-Boelter correlation for single-phase flow, and the Chen correlation for two-phase flow [5]. General Atomic (GA), the vendor of the TRIGA Mark II reactor, recommended the utilization

*Corresponding author: altaf335@yahoo.com

of the Bernath correlation for DNBR calculation in the thermal hydraulics safety analysis due to its conservative nature [6]. Nevertheless, no thermal hydraulics burn-up analysis has been performed using the PARET/ANL code [7], which encompasses a wide array of correlations, particularly the Bernath correlation. In light of this, PARET/ANL has been utilized for this simulation employing Bernath correlation, with COOLOD-N2 [8] for comparative purposes. This study has simulated key safety parameters, specifically fuel centerline temperature, fuel surface temperature, fuel clad temperature, coolant temperature, enthalpy, DNB heat flux and DNBR to ensure the reactor operates within the safety margin. The ultimate purpose of this research is to utilize the PARET code for transient analysis for the present and modified core configurations.

2. Calculation Method

In thermal-hydraulic analyses, the term steady state in a reactor denotes a state where the reactor sustains a consistent power level that remains constant throughout its operation. The PARET/ANL and COOLOD-N2 code possess the capability to compute the heat transfer phenomena from the fuel element to the coolant when the reactor operates under steady-state conditions. The heat generation within the fuel core along the radial axis is assumed to be uniform, and a one-dimensional heat transfer models are employed in this codes. In this process, the distribution of temperatures along the axial direction of the fuel rods is determined based on the local bulk temperature of the coolant and axial peaking factors. Therefore, these two codes have been employed to calculate the steady state parameters of thermal hydraulics of the reactor. Figure 1 illustrates the configuration of the existing core of the reactor. Also, Table 1 represents significant thermal-hydraulic parameters of the reactor.

The power peaking factors of the 700 MWD burned core, used in this thermal hydraulics calculation, were determined using the Monte Carlo code MVP, incorporating the cross-section data library JENDL3.3 [9]. The calculation yielded the hottest rod factor of 1.668. Only peaking factor has been revised for simulating 700 MWD as no thermal properties have been reexamined here. Also physical parameters and operating condition for steady state operation for both BOC and 700MWD are same as the core arrangement is still the same. One other data that will be changed is reactivity and associated control rod insertion rate. However that will be applied in transient operation.

The axial peak-to-average power ratio at the hottest rod within the TRIGA core is presented in Figure 2. The thermal-hydraulic calculations were carried out with a water inlet temperature of 40.6˚C and an inlet pressure of 160.6 kPa, corresponding to the static pressure of water across the reactor channels. As per the final safety analysis report, the rate of mass flow for coolant circulation in the downward direction stands at 13248

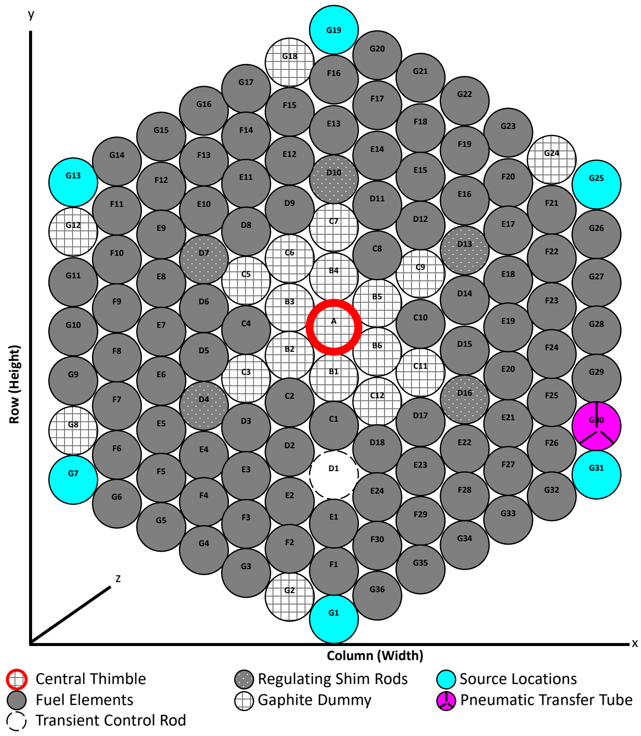

Figure 1. Present Core arrangement of TRIGA mark II research reactor.

liters per minute [10].

The PARET/ANL code was chosen due to general applicability, its simplicity of coding, and rapid execution. It is designed for use in predicting the course and consequences of non-destructive reactivity accidents in small reactor cores. It is a

Table 1. TRIGA fuel specifications.

Parameters	Design Vaue
Fuel Element (rod type)	20% w/o U-ZrH, 19.7% enriched
Total Number of fuels in the core	100
Cladding	Stainless Steel 304L
Reflector	Graphite
Inlet Temperature ˚C (Full Power)	40.6
Radius of Zr rod (cm)	0.3175
Fuel radius (cm)	1.82245
Clad outer radius (cm)	1.87706
Gap width (cm)	0.00381
Active fuel length (cm)	38.1
Flow area (cm^2)	5.3326
Hydraulic Diameter (cm)	1.80594
Pressure (Pa)	1.60654×10^5
Friction Loss Coefficient	0.07
Pressure Loss Coefficient	1.81(Inlet), 2.12 (Outlet)
Pitch (cm)	4.5716
Mass flow rate, kg/m^{2}s	3.2089×10^3
Coolant Velocity (cm/sec)	287.58

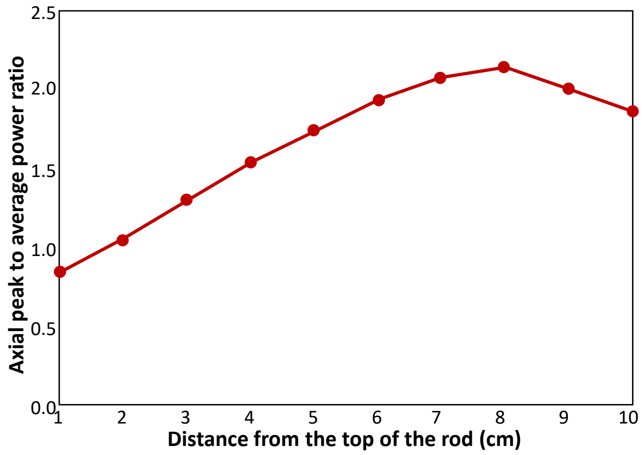

Figure 2. Axial peak to average power ratio.

coupled neutronic-hydro-dynamic heat transfer code employing point kinetics, one-dimensional hydrodynamics, and one-dimensional heat transfer. The kinetics equations of the point reactor ensure the dynamic behavior of power within the reactor through computational analysis. The time-dependent temperatures within the fuel element are computed using a one-dimensional heat conduction equation solved in axial sections. The resolution of these equations is accomplished by estimating the reactivity feedback from the initial moment until the specific point of interest. The feedback resulting from the expansion of fuel rods, the density effects of the moderator, and the fuel temperature effect collectively contribute to the overall reactivity feedback. The PARET/ANL model consists of a water-cooled core represented by a maximum of fifteen fuel elements and associated coolant channels. In our modelling the whole core was divided into two channels, keeping the hottest rod and associated coolant in one channel and other fuel and coolants in another channel. All channels were divided into 10 equal nodes. The Bernath correlation was chosen for the calculation of DNBR in this instance, as it yields the minimum value in accordance with the recommendations put forth by GA.

The COOLOD-N2 code is utilized for research reactors employing both plate-type and rod-type fuels. It can calculate fuel temperatures for both forced convection and natural convection cooling modes. The heat transfer coefficient and the DNB heat flux are determined using the Heat Transfer Package and Lund Correlation within this code. Power distribution at ten equal nodes along only the hottest rod was utilized in its modeling.

3. Results And Discussions

3.1. Axial Temperature Profiles

Steady-state thermal hydraulics parameters such as fuel centerline temperature, fuel surface temperature, fuel clad temperature along the axial length of 38.1 cm of the fuel, and coolant temperature have been studied in this simulation utilizing both PARET/ANL and COOLOD-N2. Data were taken at the center of the 10 equal axial nodes. All temperature profiles, such as fuel centerline temperature, fuel surface temperature, clad temperature, and coolant temperature, displayed an ascending trend from the top of the rod where it touches the peak just below the axial center, then gradually decreased towards the rod's end for both codes, as shown in Figure 3.

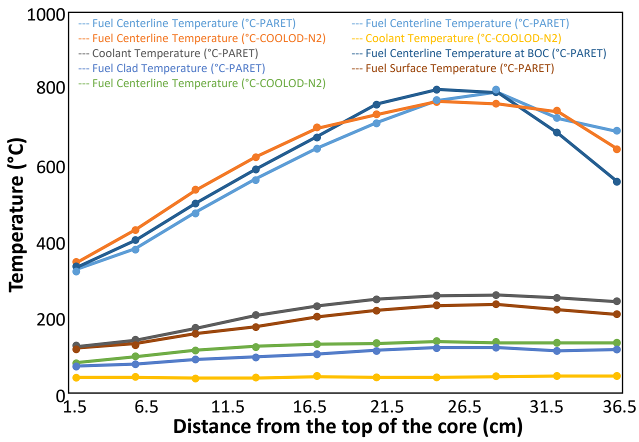

Figure 3. Temperature of the hottest channel at 750 MWD burned core

The analysis revealed that in the case of the PARET/ANL code, the fuel centerline temperature varied from 324.88°C at the topmost point to 678.48°C at the down most point, reaching its pinnacle at 770.03°C at position 28.57 cm from the top, while for COOLOD-N2, it ranges from 329.02°C to 636.68°C with a peak of 763.71°C at the same position. It was observed in another calculation that for the Beginning of Cycle (BOC) core, the PARET/ANL simulation spanned from 330.99°C to 552.26°C with a peak of 793.41°C at 24.76 cm from the peak. Hence, as the power peaking factor at burnup decreased, fuel centerline temperature profile also followed it.

The fuel surface temperature exhibited a similar pattern; it ascended from the upper region, reaching its maximum near the peak centerline temperature position, then gradually decreased towards the bottom. In the context of the PARET/ANL code, the temperature initiated at 114.50°C and culminated at 211.87°C, with a peak of 236.90°C. On the other hand, the fuel surface temperature in the COOLOD-N2 code ranged from 112.7°C to 236.15°C, with a peak of 254.69°C.

In another calculation, in the case of the PARET/ANL code, the clad temperature began at 73.63°C at the upper node, showing an upward trend until it reached the peak of 126.67°C at 24.76 cm, then decreased to 116.58°C at the lower node. Similarly, the COOLOD-N2 temperature profile mirrored the PARET/ANL results, starting at 83.09°C at the top, peaking at 133.52°C, and then declining to 127.05°C at the bottom. Hence, concerning both the maximum fuel surface temperature and the peak clad temperature, COOLOD-N2 exhibits an overestimation when compared to PARET.

Subsequently, the coolant temperature commenced at 40.58°C at the top, exhibiting a nearly consistent increase until it reached 47.38°C at the end for PARET/ANL, while starting at 40.76°C at the top and rising to 46.59°C at the bottom for COOLOD-N2.

It is observed that among the four profiles, the fuel centerline temperature profile stood out the most, with PAREL/ANL showing an overestimation of the peak centerline temperature in comparison to COOLOD-N2. Only a limited amount of nucleate boiling is observed between the axial positions of 24.76 and 28.57 cm from the top, which is consistent with the Beginning of Cycle (BOC) outcomes [11]. However, peak temperatures from both codes are well below the fuel swelling temperature limit of 950°C as per SAR and are comparatively lower than the peak value for the BOC core, implying the burned core is safer than the BOC core.

It is also noted that all profile peaks were found just below the axial center. Ideally, flux distribution in fuels supposed to be in cosine shape over the axial length as neutrons in axial central gets moderated more than in the peripheral region. However, as coolant passes through the channel from the top, it takes heat from the fuel and get heated. As coolant comes down, its temperature increases. More importantly, it faces more heated fuel. As a consequence, the capacity of coolant to transfer heat from fuel reduces over axial length from the top. Therefore the right side of the cosine shape, due to lower part of the fuel, lifted slightly leaving the peak just down the axial center.

3.2. DNB Heat Flux and DNB Ratio

DNB heat flux and DNBR stand out as crucial parameters. To prevent the most adverse combination of mechanical and coolant conditions within the core, it is imperative to maximize the value of DNBR from unity as outlined in the Safety Analysis Report (SAR). DNB heat flux and DNBR calculation have been performed utilizing both codes [Figure 4 and Figure 5].

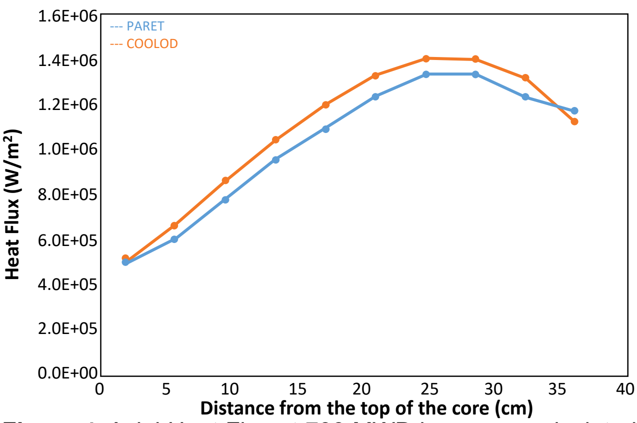

Figure 4. Axial Heat Flux at 700 MWD burn core calculated by COOLOD-N2 and PARET/ANL codes at 3 MW power.

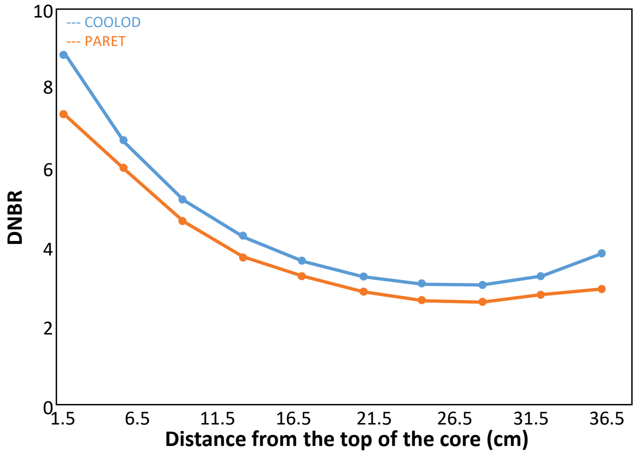

Figure 5. DNBR profile at 700 MWD burn core calculated by COOLOD-N2 and PARET/ANL codes at 3 MW power.

The DNB heat flux varies from 5.01×10^5 to 1.17×10^6 W/m^2 from the uppermost to the lowermost points, peaking at 1.35×10^6 W/m^2 at 28.57 cm for PARET. In comparison, it ranges from 5.05×10^5 W/m^2 to 1.11×10^6 W/m^2 with a maximum of 1.41×10^6 W/m^2 at the same position for COOLOD-N2. The DNBR spectrum for PARET/ANL was computed to be 7.39 at the topmost point, progressively decreasing until reaching its minimum value of 2.61 at 24.76 cm. It then rises to 2.96 at the lowest point. On the other hand, for COOLOD-N2, the DNBR ranges from 8.93 to 3.85, with a peak of 3.04.

As the DNB heat flux aligns with the enthalpy in the hottest rod, it escalates in the axial direction from top to bottom, culminating at the center for both codes. Similarly, the DNBR displays an inverse pattern as it represents the ratio of the DNB heat flux to the critical heat flux, with the minimum value corresponding to the maximum DNB heat flux. Consequently, PARET/ANL tends to overestimate the DNB heat flux compared to COOLOD-N2, while the opposite holds for the DNBR. It is also bigger than SAR accepted minimum value. Hence, our minimum DNBR values from both codes remarkably surpass that of EUREKA-2R and SAR, underscoring the core's safety at an operating power of 3MW.

3.3. Enthalpy

Also, the enthalpy within the fuel rod was determined solely using PARET/ANL, presented in Figure 6, as COOLOD-N2 does not offer this capability. Throughout the core's length, a consistent increasing trend is observed, commencing at 169.52 Kj/Kg and culminating at 197.94 Kj/Kg.

4. Conclusions

The thermal-hydraulic safety analysis of the 3 MW TRIGA Mark-II research reactor at Savar, Dhaka, Bangladesh has been evaluated using COOLOD-N2 and PARET/ANL. The computational outcomes of both cores were compared, revealing a high degree

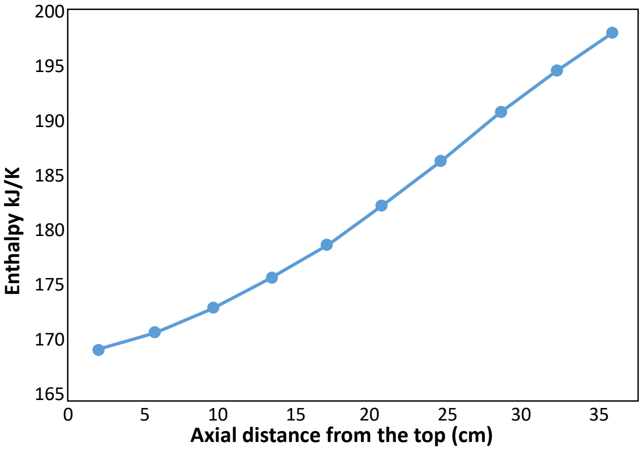

Figure 7. Enthalpy calculated by PARET/ANL.

of consistency between the two. Based on these findings, it can be inferred that the PARET/ANL and COOLOD-N2 codes exhibit a commendable capability in accurately predicting steady-state temperature profiles, enthalpy, DNB heat flux, and DNBR. Moreover, it is deduced that all safety parameters maintain margins well below the safety limits. Thus, the burned core of 3 MW TRIGA Mark-II research reactor is safer to operate compared to the BOC core. Therefore, the utilization of the PARET/ANL code is recommended for conducting transient analyses of the burned core, as well as for any modifications to the TRIGA core design.

References

[1] Hossain, S. M., Zulquarnain, M. A., Kamal, I. and Islam, M.N. Current Status and Perspectives of Nuclear Reactor Based Research in Bangladesh, IAEA, Vienna, (2011), 7-14. IAEA-TECDOC-1659

[2] Mizanur Rahman M. M., Akond M. A. R., Basher M K., and Huda M. Q., Steady State Thermal Hydraulic Analysis of TRIGA Research Reactor, World Journal of Nuclear Science and Technology, (2014), 81-87. DOI: 10.4236/wjnst.2014.42013

[3] Huda M. Q. and Rahman M., Thermo-Hydrodynamic Design and Safety Parameter Studies of the TRIGA MARK II Research Reactor, Annals of Nuclear Energy, 31, (2004) 1101-1118. https://doi.org/10.1016/j.anucene.2004.02.001

[4] Altaf M. H. and Badrun N. H., Thermal Hydraulic Analysis of 3 MW TRIGA Research Reactor of Bangladesh Considering Different Cycles of Burnup, Atom Indonesia, 40 (2014) 107-112. DOI:10.17146/aij.2014.328

[5] Sudo, Y., Ikawa, H. and Kaminaga, M, Development of Heat Transfer Package for Core Thermal Hydraulics Design and Analysis of Upgradded JRR-3. Proceedings of the International Meeting of Reduced Enrichment for Research and Test Reactors, Petten, (1985) 363-372.

[6] General Atomics, 10 MW TRIGA-LEU Fuel and Reactor Design Description. General Atomics, San Diego (1979).

[7] Obenchain, C.F., PARET—A Program for the Analysis of Reactor Transients. IDO-17282, Idaho Atomic Energy, 1969.

[8] Kaminaga, M., COOLOD-N2: A Computer Code, for the Analyses of Steady-State Thermal Hydraulics in Research Reactors, JAERI Report. Japan Atomic Energy Research Institute, Tokai-mura, Ibaraki, 1994.

[9] Mahmood M. S. et al., Individual Fuel Element Burnup of BAEC TRIGA Core, Technical Report, INST-RPED-RARD-01/009 (2012).

[10] FSAR, Final Safety Analysis Report for 3 MW TRIGA MARK-II Research Reactor at AERE, Savar, Dhaka, Bangladesh. BAEC, Dhaka (2006).

[11] Huda, M. Q., Bhuiyan, S. I., Chakroborty, T.K., Sarker, M.M. and Mondal, M.A.W. Thermal Hydraulic Analysis of the 3 MW TRIGA MARK-II Research", Reactor. Nuclear Technology, 135, "(2001), 51-66. https://doi.org/10.13182/NT01-A3205.