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Pulse mode analysis of TRIGA Burned Core using thermal hydraulics code PARET/ANL

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ABSTRACT

A transient-state thermal hydraulics analysis of the TRIGA burned core of the TRIGA Mark II research reactor was conducted using PARET/ANL. This study focused on calculating safety parameters due to large reactivity insertions at low operating power. The peak power and energy for the burned core were computed, whose values were found to be lower than those of the beginning of the cycle (BOC) core. Moreover, the safety parameters of DNBR and clad temperature remained well within the margins of the Safety Analysis Report (SAR). No nucleate boiling was observed in the hottest fuel. These findings indicate that the burned core can be safely utilized for pulse mode operation.

1. Introduction

TRIGA Mark II research reactor in Bangladesh was commissioned in Bangladesh in 1986. The reactor is designed for multi-purpose use, such as training, education, radioisotope production and various R&D activities including neutron activation analysis, neutron scattering, and neutron radiography [1]. Having a water-cooled and graphite-reflected core, the reactor has the capacity to operate continuously at 3 MW (thermal). The unique feature of the reactor is its substantial prompt negative temperature coefficient that can control the reactor before its engineering control system is active. The TRIGA reactor comprises 100 fuels, including 5 fuel follower control rods and 2 instrumental fuel elements. The fuel is arranged in a hexagonal array in the reactor core shroud that distributes fuel efficiently and utilizes space optimally. TRIGA reactors offer operational flexibility by functioning in both steady and transient states, notably featuring a unique pulse mode, and while they rely on natural convection for cooling up to 500 kW, forced convection is employed for operation at higher power levels.

Thermal hydraulics research primarily focuses on efficiently removing heat from the fuel under various operating, burnup, and core arrangement conditions to prevent excessive fuel temperatures, steam void formation, and approaching the hydrodynamic critical heat flux.

The unique and extreme operating feature of TRIGA reactors is pulse mode that generates immense flux for research and training when a large amount of reactivity is achieved for a short period. Even at low energy operation, usually around 100 watts, accidental initiation of large reactivity, such as from control rod blockage, can induce pulsing. While the negative temperature coefficient of the reactor inherently controls the power, rapid generation of significant power in pulse mode operation can still impose substantial thermal and mechanical stress on the fuel and core.

After 38 years of operational life, the TRIGA reactor is no longer capable of operating in full power; It is now run at low power (100 watts) to support research only. However, even at this reduced power, there is a risk of large pulsing if sufficient reactivity is suddenly introduced. Furthermore, the extensive burnup of fuel elements from fission reactions has altered the axial and radial fuel composition as the core has already experienced 800 MWDs (Mega Watt Days) of operation life. This change directly impacts the power peaking factor and heat transfer coefficients. As the core continues to experience burnup, thermal hydraulics parameters like fuel temperature and DNBR (Departure from Nucleate Boiling Ratio) - defined as the ratio of the critical heat flux to the heat flux achieved in the core - could exceed safety limits. Therefore, it is essential to study these safety parameters specifically for pulse mode operation under current burnup conditions.

Historically, several codes have been used so far to calculate the thermal-hydraulic characteristics of the Beginning of Cycle (BOC) core of the TRIGA Mark

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II research reactor [2, 3]. Altaf et al. [4] conducted a thermal hydraulics study of the burned core using EUREKA-2/RR. However, EUREKA does not include Bernath correlation, which is suggested by General Atomic [5] to conduct the DNBR calculation, a crucial safety parameter. Moreover, no thermal hydraulics transient study of burned core of Bangladesh Atomic Energy Commission (BAEC) TRIGA has been performed using the PARET/ANL code, which encompasses a wide array of correlations, particularly the Bernath correlation. Therefore, safety parameters due to the insertion of a large amount of reactivity was studied in this simulation to ensure the reactor operates within the safety margin.

2. Calculation Method

The PARET/ANL [6] code possess the capability to simulate heat transfer phenomena from the fuel element to the coolant when the reactor is operating. Therefore, the PARET/ANL code was employed to calculate the transient state parameters of thermal hydraulics of the reactor. Figure 1 illustrates the configuration of the existing core of the reactor. 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.

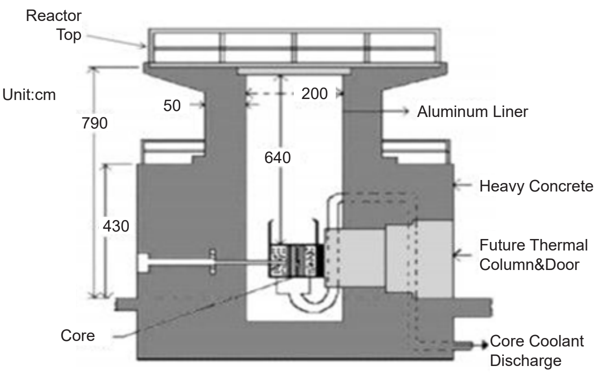

Figure 1. Cutaway view of TRIGA reactor

It is a coupled neutronic-hydrodynamic 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 15 fuel elements and associated coolant channels. In our modeling, the whole core was divided into two channels, keeping the hottest rod and associated coolant in one channel and other fuel rods and coolant in the rest. All channels were divided into 10 equal nodes. Table 1 represents thermal-hydraulic operating parameters of the TRIGA.

Table 1. TRIGA fuel specifications.

Parameters	Design Value
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^2s	3.2089×10^3
Coolant Velocity (cm/sec)	287.58

The power peaking factors of fuel rods and axial peak-to-average ratio of the hottest rod of the 700 MWD burned core, used in this thermal hydraulics calculation, were determined using the Monte Carlo code MVP [7]. 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. Physical parameters and operating conditions for transient state operation for 700MWD has been kept as same as the BOC core as the core arrangement is still the same. 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 (FSAR), the rate of mass flow for coolant circulation in the downward direction stands at 13248 liters per minute as per FSAR [8]. For the current study, the thermal hydraulics safety parametric study due to reactivity insertion of 1.996$ and 2.24$ at operating power of 100 watts for 0.1 sec with 0.015 delay time has been selected so that the results can be compared with already available data for the same condition of the BOC core conducted by Huda M Q [9]. Bernath correlation has been selected for DNBR calculation in this study.

3. Results and Discussions

3.1 Reactivity Insertion of 1.996$

Transients of reactor power, energy, minimum DNBR at the hottest spot of the hottest fuel, and clad temperature at the surface of the hottest fuel for reactivity insertion of 1.996$ was studied in this simulation utilizing PARET/ANL. As depicted in Figure 2, reactor power began to rise upon reactivity insertion. Although

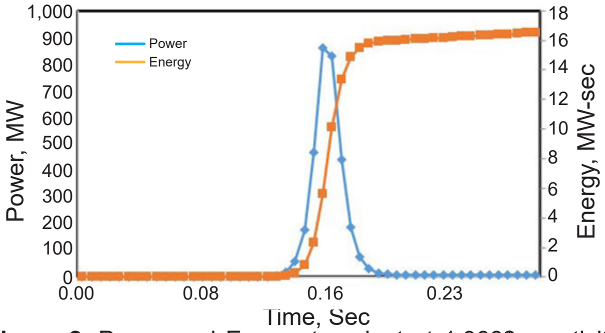

Figure 2. Power and Energy transient at 1.966$ reactivity insertion

expected to continue until the 0.1 second insertion time elapsed, a 0.015 second delay resulted in a peak power of 859.44 MW at 0.115 seconds. Subsequently, negative feedback reactivity caused power to return to its initial level by 0.18 seconds. The peak power for the same reactivity was calculated to be 873 MW in the BOC core by Huda M. Q. [9]. The power pulse's full-width half maxima was 15.8 milliseconds, further indicating the burned core is safer for pulse mode operation compared to the BOC core. Concurrently, core energy increased over time, initially rising slowly with power before accelerating significantly after 0.1 seconds, mirroring the power curve. It reached 16.5 MW-sec at 0.20 seconds before stabilizing.

DNBR stands out as a crucial parameter in the safety analysis. 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. DNBR calculation over 0.40 seconds starting from arbitrary zero seconds for the reactivity insertion of 1.996$ is shown in Figure 3. The minimum DNBR over the transient time was found to be 1.59 over the simulation time, which is bigger than the SAR accepted minimum value of unity, ensuring the safety of the core at this operating condition.

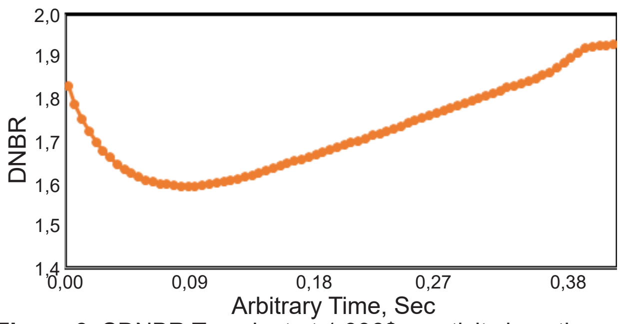

Figure 3. SDNBR Transient at 1.996$ reactivity insertion

In another study, the cladding temperature of the hottest rod for the same reactivity insertion and operating power was computed. Figure 4 shows the cladding temperature over 5 seconds from the insertion of reactivity of 1.996$. It is evident from the figure that the cladding temperature increases sharply after the insertion and reaches 133.64 °C peak temperature, compared to 144.54 °C for ERUEKA-2/RR for 2$ [10]. After that, the temperature decreases gradually until dropping down to 82.89°C at 5 seconds. No nucleate boiling has been found in the output file for the hottest fuel rod.

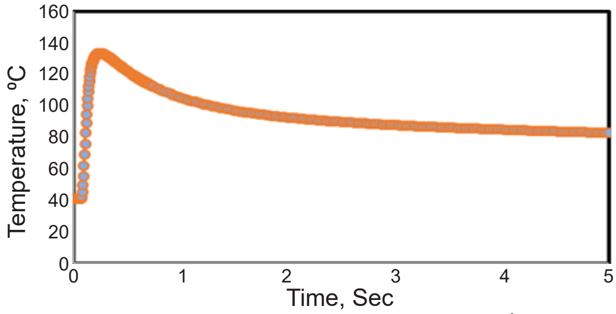

Figure 4. Temperature transient with 1.996$ reactivity insertion

3.2 Reactivity Insertion of 2.24 $

Transients of reactor power, energy, minimum DNBR at the hottest spot of the hottest fuel, and cladding temperature at the surface of the hottest fuel for the reactivity insertion of 2.24$ was studied in another simulation. As depicted in Figure 5, reactor power began to rise upon reactivity insertion. Although

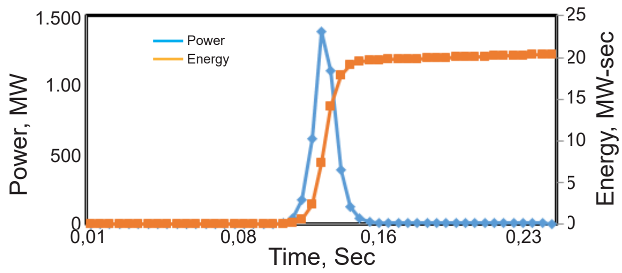

Figure 5. Power and Energy transient at 2.24$ reactivity insertion

expected to continue until the 0.1-second insertion time elapsed, a 0.015-second delay resulted in a peak power of 1385 MW at 0.15 sec. Subsequently, negative feedback reactivity caused power to return to its initial level in 0.19 seconds. The peak power for the same reactivity was calculated to be 1629 MW in the BOC core by Huda M. Q. [9]. Hence, the burned core found to be safer for pulse mode operation compared to the BOC core. The full-width half maxima of the power pulse was found to be 16.2 milliseconds, compared to 15.8 milliseconds for 1.996$.

Meanwhile, the energy of the core increased steadily over time. It began rising with the increase of power from the beginning of reactivity insertion, and then surged after 0.1 seconds, reaching 20.4 MW-sec at 0.22 seconds before stabilizing. It is worth noting that both the power and energy for the 2.24$ reactivity insertion were greater than those for the 1.996$ insertion.

DNBR calculation over 0.42 seconds after arbitrary zero seconds for reactivity insertion of 2.24$ is shown in Figure 6. The minimum DNBR over the transient time was found to be 1.21 over the simulation time, which is less than 1.59 for a reactivity insertion of 1.996$. However, this value is bigger than the SAR accepted minimum value of unity, meaning the core operates safely for a reactivity insertion of 2.24$ during 100 watts operating power.

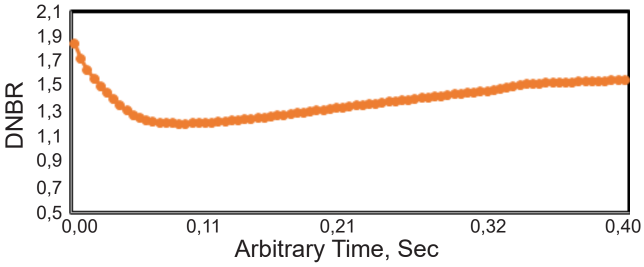

Figure 6. DNBR Transient at 2.24$ reactivity insertion

Lastly, the clad temperature for this reactivity insertion and operating power was conducted. Figure 7 shows the cladding temperature over 5 seconds from the insertion of reactivity of 2.24$. One can observe from the figure that the cladding temperature increases sharply after the insertion and reaches 139.56°C peak temperature, compared to 133.71°C for 1.996$. After that, the temperature decreases gradually till it drops down to 84°C at 5 seconds, compared to 82.89°C for 1.996$. No nucleate boiling has been observed in fuel.

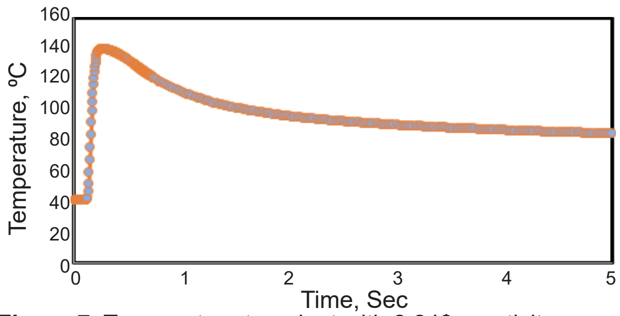

Figure 7. Temperature transient with 2.24$ reactivity insertion

4. Conclusion

The reactivity-initiated transient analysis of the 3 MW TRIGA Mark-II research reactor at Savar, Dhaka, Bangladesh, has been evaluated using PARET/ANL. The study compares the thermal hydraulics parameters of the reactor's burned core with those of its Beginning of Cycle (BOC) core. The results indicate that safety parameters for the burned core were consistently lower and no nucleate boiling was observed in any region of the hottest fuel. These findings suggest that the burned core of the 3 MW TRIGA Mark-II research reactor is safer to operate than the BOC core. Furthermore, among the two reactivity insertions in the burned core, the reactor was found to be safer for low reactivity insertion. Based on these observations, the burn-TRIGA core can be recommended for pulse mode operation.

5. Acknowledgment

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