In-core fuel management of TRIGA reactor optimising performance and safety

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Abstract: TRIGA Mark II research reactor is under operation in Bangladesh since 1986 and postponed radioisotope production in 2008. The possibility to extend the length of operation cycle of the reactor core by compacting the inner region with fuels replacing the graphite elements has been investigated. This leads to the increment of core excess reactivity at the expense of reactor performance. And confirmation of the safety has been demonstrated by nucleate boiling analysis. The result finds promising invitation for the future analysis with explicit thermal hydraulic safety calculation for the compact core configuration of the reactor.

Keywords: MCNP; TRIGA; in-core fuel management.

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1 Introduction

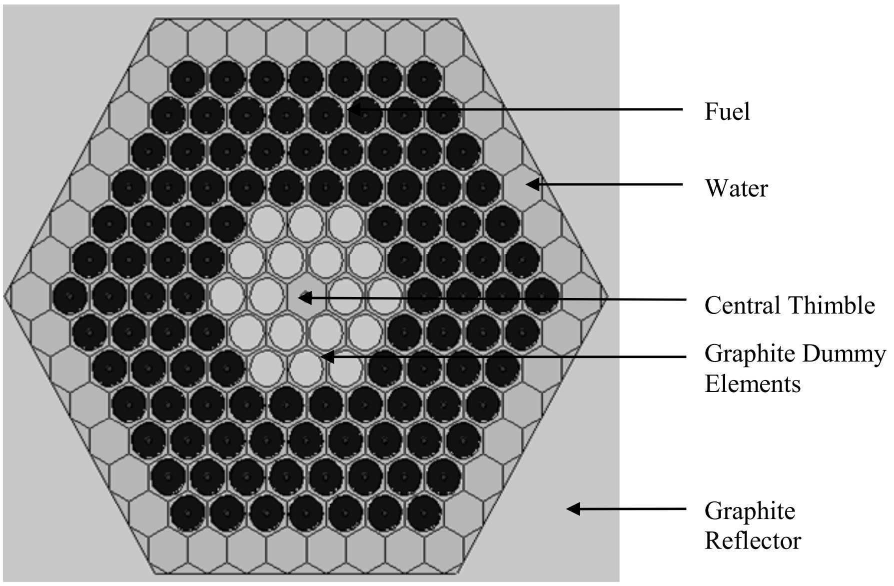
Bangladesh Atomic Energy Commission (BAEC) is operating a 3 MW TRIGA Mark II research reactor. The inner core of the BAEC TRIGA Research Reactor (BTRR) is filled with graphite dummy elements to attain high thermal flux for radioisotope production at its central irradiation channel. Having operated since 1986 for training, research and isotope production, the facility postponed isotope production in 2008 due to some technical issue. Now after 800 MWD operation the reactor has limited its utilisation in order to save fuel consumption. Additional fuel procurement is yet to be decided. In this circumstance, the possibility has been investigated to increase the operation cycle life of the existing core. One option is to compact the core replacing the inner core graphite dummies with fuel elements. The objective is to increase the core excess reactivity at the expense of reactor performance i.e. reducing the thermal flux in the Central Thimble (CT) at the core centre where irradiation was performed for radioisotope production. This will not disturb the core periphery neutron beam application. The main concern is the reactor operation safety because such an in-core fuel management will enhance the radial peaking factor which inturn will affect the Departure of Nucleate Boiling Ratio (DNBR). In Lyric et al. (2013), the power distribution pattern remains same during the whole operation cycle of the small core like TRIGA reactor and hence it is expected that the present analysis, which is done for the fresh core condition for simplicity, is equally applicable for the burnt core.

2 Neutronic tools and modelling

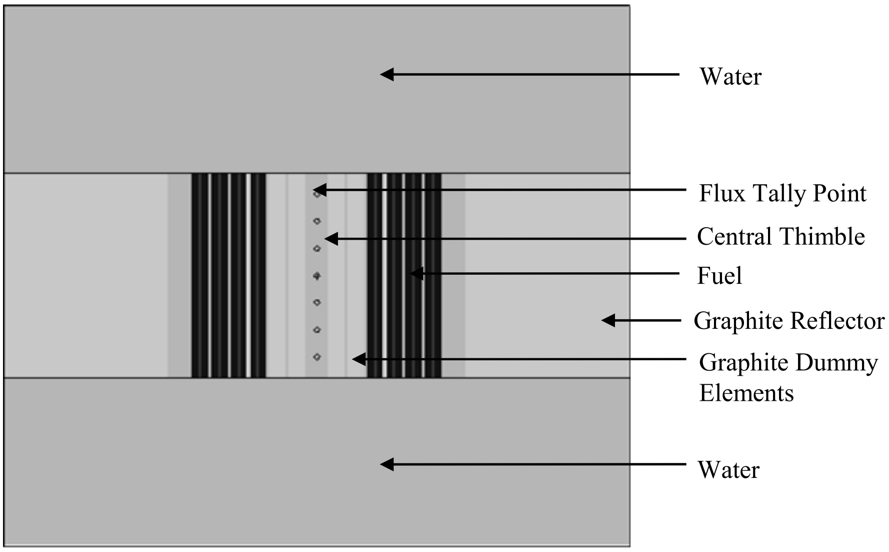
MCNP5, version 1.60 (X-5 Monte Carlo Team, 2005) with nuclear data library based on ENDF/B-VII (Chadwick et al., 2006) has been used for the neutronic simulation. This work is based on the MCNP model of TRIGA core developed in Yasmeen and Mahmood (2017). MCNP provides accurate probabilistic transport solution and verified for TRIGA reactor analysis (Delfin-Loya et al., 2016). Two different control rod free configurations, each with equal number of fuel pins have been considered in the present study. Detailed cross-sectional views of the MCNP models of the TRIGA core under investigation are shown in Figures 1 and 2. Sequentially from the centre location, labelling the concentric hexagonal rings as A, B, C and so on the reference core comprised graphite dummy elements at B and C ring around the water-filled CT. The graphite elements are surrounded by fuel elements in the D, E, F and G rings. In the second configuration the core has been compacted by shifting the fuel pins to the inner positions replacing the graphite dummy elements. In this simulation the core is surrounded by graphite with water in between. Top and bottom of the active core height is modelled with water. Perfect absorber boundary condition has been applied surrounding the whole system. The arbitrary dimensions of the core externals are immaterial as long as our concern is the relative change in reactivity and local flux.

The simulation has been done with 11×10^2 cycles with 1×10^4 neutron histories per cycle and 1×10^2 initial cycles were skipped for quick convergence. The performance parameter has been quantified with the *f* value of the reactor i.e. the ratio of the thermal to higher energy neutron flux; up to 4×10^{-7} MeV flux has been considered as thermal. The power peaking factor that quantifies the safety is the ratio of the peak to average value of the fuel pin power. Transiting from the reference to compact core, there is no change in the vertical direction and hence it is enough to consider only the radial peaking. Thermal neutron flux has been estimated with the track length option (F4 tally of MCNP) at 5 cm interval starting from –15 cm to +15 cm axially in the central thimble.

Figure 1 Reference core MCNP model (a) Horizontal cross-section (b) Vertical cross-section

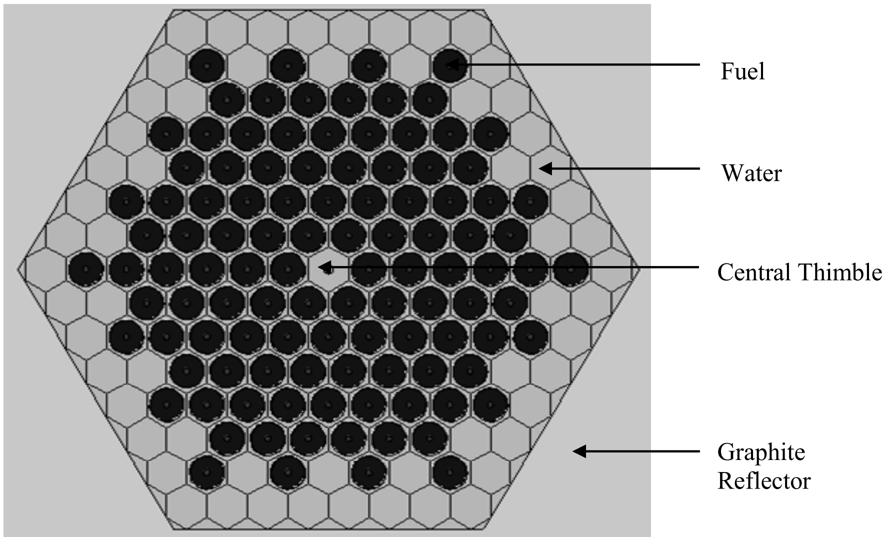


(a)

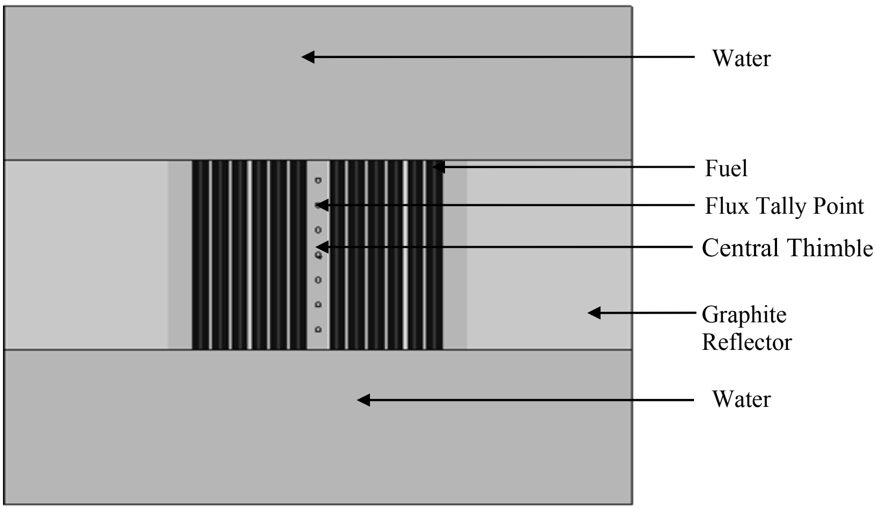


(b)

Figure 2 Compact core MCNP model (a) Horizontal cross-section (b) Vertical cross-section



(a)



(b)

3 Results and discussions

MCNP calculation shows the increased effective multiplication (k_{eff}) value (see Table 1) for the compact core as was expected due to the fuel contribution at the inner core. Lowering of thermal flux at the CT is for less moderation (see Figure 3) in the graphite free compact core. The statistical uncertainty is shown in 1 standard deviation in Figure 3. The calculation shows 65% reduction of the reactor performance (see Table 2). The core excess reactivity increased by 6$ (Table 1); this indicates about 1200 MWD extension of core operating life (Lyric et al., 2013). There will be margin to allow this extended burnup after 23.84% maximum burnup of U235 for the present core configuration of BTRR which is within the safety limit, i.e. 50% for TRIGA fuel (Lyric et al., 2013). Also, the constraint of 3.3$ shutdown margin (Huda et al., 2004) may be covered by not using 100 fuels all together in the core keeping out for successive use.

Table 1 Multiplication factor and excess reactivity

Configuration	$k_{eff} \pm \Delta^{a}$	*Excess reactivity($)*
Reference core	1.10717 ± 0.00028	13.83
Compact core	1.16123 ± 0.00029	19.83

Note: [a]The statistical uncertainty is shown in 1 standard deviation.

Figure 3 Thermal flux profile

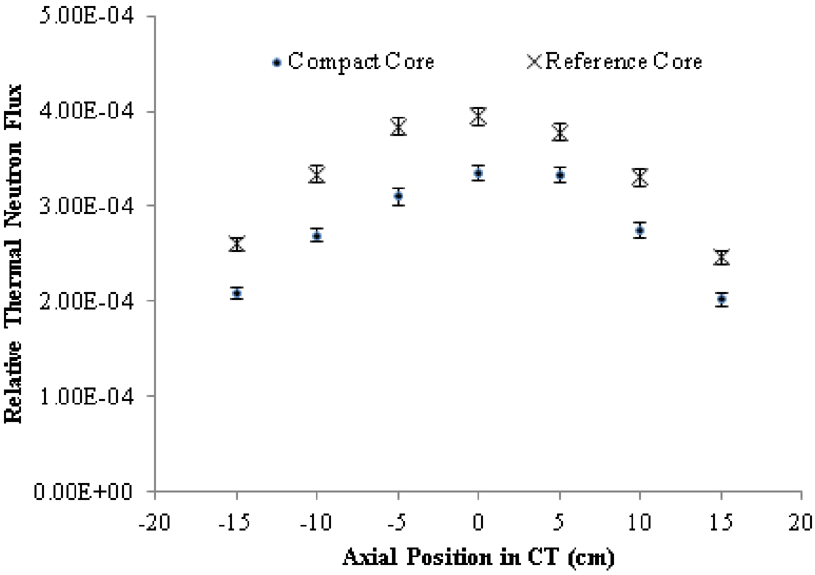


Table 2 *f*-value at the core centre

Configuration	*Thermal flux/Higher energy flux*
Reference core	2.32
Compact core	0.80

Radial power profiles at fuels in different rings of both cores are presented in Figure 4. Naturally the power peak shifted to the B ring in the compact core with an increased value of hot rod power production (see Table 3) compared to the reference core. Radial peaking factor has been found to be increased by a value of 0.13. This will increase the total peaking factor 1.09 times the present value of BTRR core. According to Huda and

Rahman (2004),. such an increase of peaking factor will reduce the minimum DNB ratio to 2.785 from the design value 2.8, which is greater than unity and still greater than the minimum DNBR value 2.3 corresponding to 120% over power (Huda and Rahman, 2004). Hence, the reactor core will remain within the safety margin of operation. The demonstrated possibility in this paper will be confirmed in the future for the exact (present) core configuration of BTRR along with detail DNB analysis with some suitable thermo-hydraulic analysis code.

Figure 4 Radial power distribution

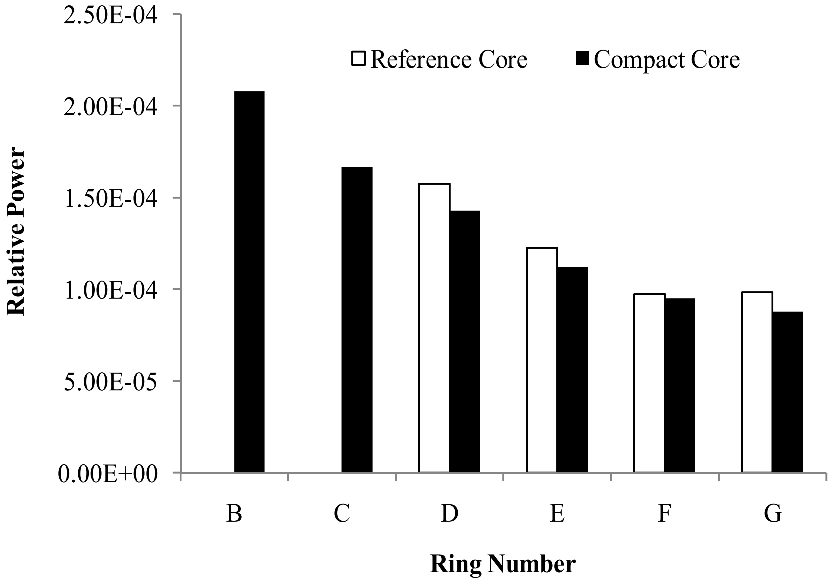


Table 3 Power peaking factors

Configuration	*Power peaking factor*
Reference core	1.50
Compact core	1.63

4 Conclusion

The feasibility of TRIGA core compactification optimising the performance and safety has been demonstrated in this study. Because it is not used for isotope production, sacrificing the performance and ensuring the safety, operation life can be extended with the existing fuel in the core. The present investigation ensures the safety margin due to compactification. This work can be expanded with explicit thermal hydraulic safety calculation. And also the result will be confirmed in respect to the present core configuration of BTRR in the future. This will open the provision of reactor utilisation for extended period of research activities.

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