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PRELIMINARY CORE NEUTRONICS DESIGN FOR THE MALAYSIAN MPR BASED ON THE 30MW$_{TH}$ HANARO REACTOR

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ABSTRACT

This paper presents a benchmarking study of the 30 MWth South Korean HANARO reactor to provide reference data for future conceptual studies of a Malaysian Multi-Purpose Research Reactor (MPR). The analysis, performed using the MCNPX 2.7 Monte Carlo radiation transport code, involved constructing a detailed HANARO core model and simulating key neutronic behavior, including criticality, neutron flux distribution, and fuel burnup. Simulation results were rigorously validated against experimental measurements at various irradiation sites, showing strong agreement for thermal neutron flux values and demonstrating the accuracy of the model in capturing core physics, non-uniform power distribution, and the effects of the heavy water reflector on thermal flux. These results establish a robust technical benchmark and a validated modeling approach that can inform preliminary core design considerations for a potential Malaysian MPR, which remains at the conceptual stage with no design decisions finalized. The study confirms that MCNPX 2.7 is a reliable tool for research reactor neutronics analysis and provides essential insights for future design and safety studies.

Keywords: Multi-Purpose Research Reactor (MPR), Neutronics Design, HANARO, MCNPX

INTRODUCTION

The Multi-Purpose Research Reactor (MPR) would serve as a versatile platform for advanced applications, including radioisotope production, high-quality irradiation, and experimental testing in the fields of medicine, agriculture, industry, and environmental research (MOSTI, 2023). The Malaysian Nuclear Agency (Nuklear Malaysia) is positioned to provide the technical foundation for a potential Multi-Purpose Research Reactor (MPR), which remains at the conceptual stage with no final design decisions made. Nuklear Malaysia's efforts focus on neutronics analysis, core modeling, and safety assessment, using validated computational tools and benchmarking against reference reactors such as HANARO. The team has developed detailed models, simulated reactor behavior, and explored fuel and irradiation configurations to generate essential scientific and engineering data. These preparatory studies establish a knowledge base to guide future MPR planning and align with the objectives of the Dasar Teknologi Nuklear Negara (DTNN) 2030.

In this study, several international reactor types were evaluated, including the VVR-TS (Russia), CARR (China), OPAL (Australia), and TRIGA (USA). The HANARO (High-flux Advanced Neutron Application Reactor) in South Korea was chosen as the primary reference due to its proven capabilities and high-power characteristics. The HANARO is an open-pool, light-water cooled reactor with a thermal power of 30 MWth and a maximum thermal neutron flux of reaching up to 5.0×10^{14} n/cm²s in its flux trap. Its design incorporates an inverse neutron trap and a heavy water (D2O) reflector, which is highly efficient at slowing down fast neutrons and boosting the thermal flux in surrounding irradiation channels. The use of low-enriched uranium (LEU) fuel further aligns the design with global non-proliferation standards (Chae, et al., 1996).

This paper presents the preliminary conceptual neutronic design of the proposed Malaysian MPR, using the well-established HANARO reactor as a benchmark. The analysis was performed with the MCNPX 2.7 Monte Carlo radiation transport code. The primary goal is to develop a comprehensive neutronic model and validate its predictions against available operational and experimental HANARO data, providing a reliable technical foundation for Malaysia's future MPR. Without a validated neutronic model, optimizing the core for its intended purposes and ensuring compliance with international safety standards would not be possible. HANARO's extensive dataset serves as an essential reference for model validation. The specific objectives of this study are: (1) to develop a detailed three-dimensional MCNPX model of the HANARO reactor core, including geometry, materials, and irradiation facilities; (2) to simulate and analyze neutronic behavior, including effective multiplication factor (k_{eff}), thermal and fast neutron flux distribution, fuel burnup, and power density distribution; (3) to validate the model against experimental HANARO data; and (4) to provide a benchmarked technical basis to guide the design of the Malaysian MPR.

METHODOLOGY

Monte Carlo radiation transport code MCNPX 2.7 is well-suited for validation and equipped with several versatile tool for criticality, flux, burnup, and safety studies. MCNPX 2.7, developed by Los Alamos National Laboratory, supports neutron, photon, electron, and coupled transport, and uses extensive nuclear data libraries (ENDF7). It is proven and benchmarked for research reactor studies and is accepted by the international community (IAEA, OECD/NEA) (Goorlet et al. 2011).

The focus of the methodology is involved developing a detailed three-dimensional model of the reactor core. The model was meticulously constructed to represent the reactor's key components, including the pool (H_2O, R=200cm), reflector (D_2O, R=100cm, H=120cm), and fuel assemblies (H=70cm). The fuel assemblies were modelled as both 36-rod hexagonal and 18-rod cylindrical types standards (Chae, et al., 1996). The model also included various irradiation holes such as for the Neutron Transmutation Doping (NTD).

Figure 1 shows the MCNPX model illustrated in two perspectives: a top-down view (a) and a side view (b). The top-down view highlights the cylindrical geometry of the reactor core and its surrounding components. The outermost red region represents water in the pool, labelled as *Pool (H_2O)*. Concentric within this pool is a blue cylindrical region, corresponding to the reflector that surrounds the core. The reactor core itself, composed of numerous several coloured cylinders that represent different fuel elements and core components. Several white dots are also visible within the reflector, which correspond to irradiation channels.

To provide a clear overview of the physical and operational parameters used in the model, a summary of the core and fuel design characteristics is presented in Table 1. This table details key parameters such as the thermal power, coolant, reflector, fuel material, and the geometry of the core. It also lists the dimensions of the fuel assemblies and the cycle length of the reactor's operation. In the reactor core model, the fuel is made up of two types of rods: standard and reduced. The standard rods, with a diameter of about 6.35 mm, are positioned in the inner sections of the fuel assemblies. Surrounding them are the reduced rods, which have a slightly smaller diameter of about 5.49 mm. This design helps improve neutron moderation in the outer regions of the core and contributes to a more even power distribution across the assemblies.

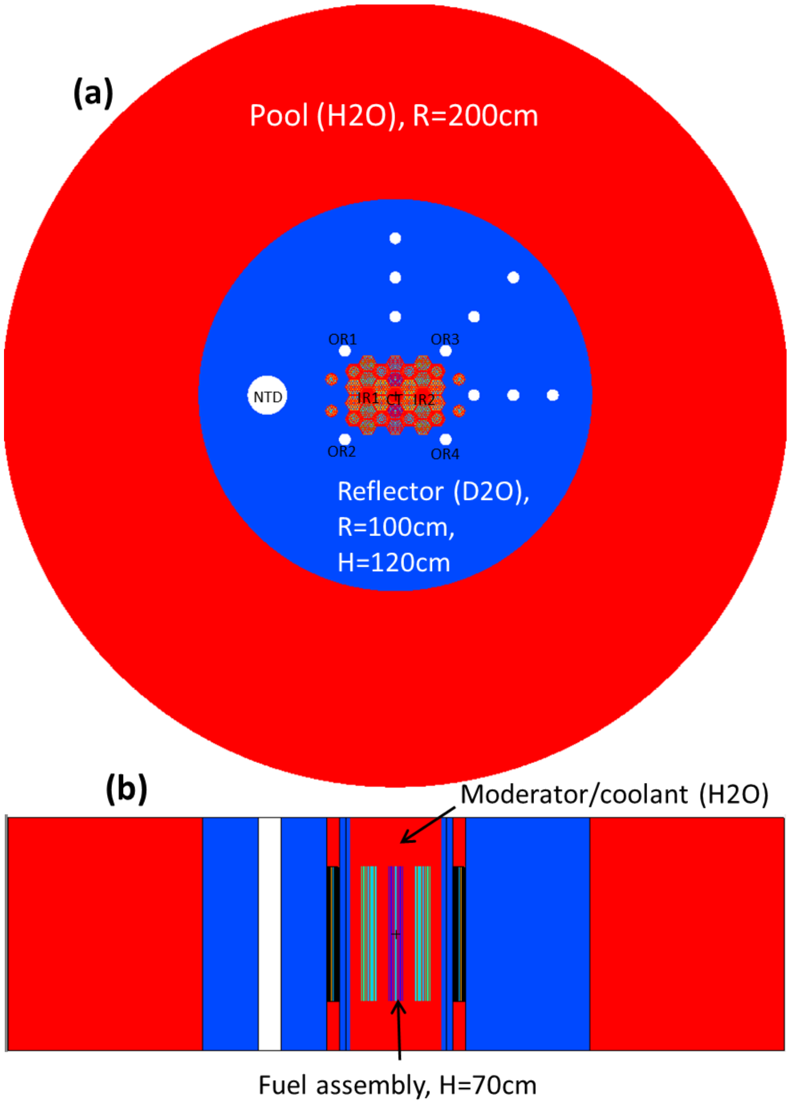

Figure 1. MCNPX model based on the HANARO reactor core shown in (a) top and (b) side views.

Table 1: Design and Operation Parameters [2]

Parameter	Value
	Reactor
Type	Open-tank-in-pool, 30 MW
Coolant/ Moderator	Light water (H2O)
Reflector	Heavy water (D2O)
Control Rods	Hafnium
	Core
Geometry	Hybrid type, 32 fuel assemblies
Reflector Dimensions	2.0 m diameter, 1.2 m height
Fuel Assembly Pitch	8.01 cm
	Fuel Rods
Fuel Material	Uranium silicide (U_3Si) dispersion
Enrichment	19.75% U−235
Cladding	Aluminum alloy (AA-1060)
Fuel Length	700 mm
Fuel Meat Diameter	6.35 mm (standard), 5.486 mm (reduced)
	Operation
Core Residence Time	~175 full-power days (~6 cycles)
Average Burnup	>50% U−235
Operating Cycle	28 days operation, 14 days maintenance

The referred HANARO reactor core provides high neutron fluxes across different irradiation sites. In the central trap (CT), the peak thermal neutron flux reaches 4.39×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, with a corresponding fast flux of 2.10×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$. The inner irradiation region (IR) records slightly lower values, with thermal and fast fluxes of 3.93×10^{14} and 1.95×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, respectively. At the neutron transmutation doping (NTD) facility, the thermal flux is 6.20×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$ while the fast flux is much lower, at 1.11×10^{11} $n\cdot cm^{-2}\cdot s^{-1}$. In the outer irradiation region (OR), the thermal flux is 3.06×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, accompanied by a fast flux of 2.13×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$ (standards (Chae, et al., 1996; Kim, et al., 2016). These reference flux values serve as benchmark experimental data that will be used to compare and validate the MCNPX model results.

For the core model U_3Si–Al dispersion fuel meat, the bulk density is about 5.25 g/cm^3, with uranium contributing 3.15 g/cm^3. Assuming 19.75% enrichment in U-235 (balance U-238, neglecting U-234), the composition per cubic centimetre can be expressed as follows: U-235 accounts for 0.622 g (1.59×10^{21} atoms/cm^3), U-238 for 2.53 g (6.40×10^{21} atoms/cm^3), silicon from the U_3Si phase for 0.124 g (2.66×10^{21} atoms/cm^3), and aluminium for 1.98 g (4.42×10^{22} atoms/cm^3). In terms of atom fractions, these correspond to 0.029 for U-235, 0.117 for U-238, 0.048 for Si, and 0.806 for Al, summing to unity within rounding. This representation provides the necessary basis for MCNP fuel definition using either atom fractions or atomic densities.

Figure 2 provides a detailed, zoomed-in view of the core model developed, highlighting the geometry and key components of the core. In the active core region (a), the central multi-colored section represents the reactor core, which contains both hexagonal and circular channels that house the fuel assemblies. The hexagonal fuel assemblies, together with the cylindrical assemblies, are arranged in a honeycomb-like configuration. White hexagonal spaces are reserved for experimental facilities, while the surrounding blue region denotes the heavy water reflector.

Panel (b) focuses on the details of the fuel assemblies and rods. On the left is a vertical cross-section of a fuel assembly showing individual rods with an active length of approximately 700 mm, while the right side provides a horizontal cross-section illustrating rod arrangements. The hexagonal assembly contains 36 rods in a hexagonal lattice, and the cylindrical assembly has 18 rods arranged in two concentric rings. The red background corresponds to the light water coolant, which flows through the assemblies to remove heat from the core.

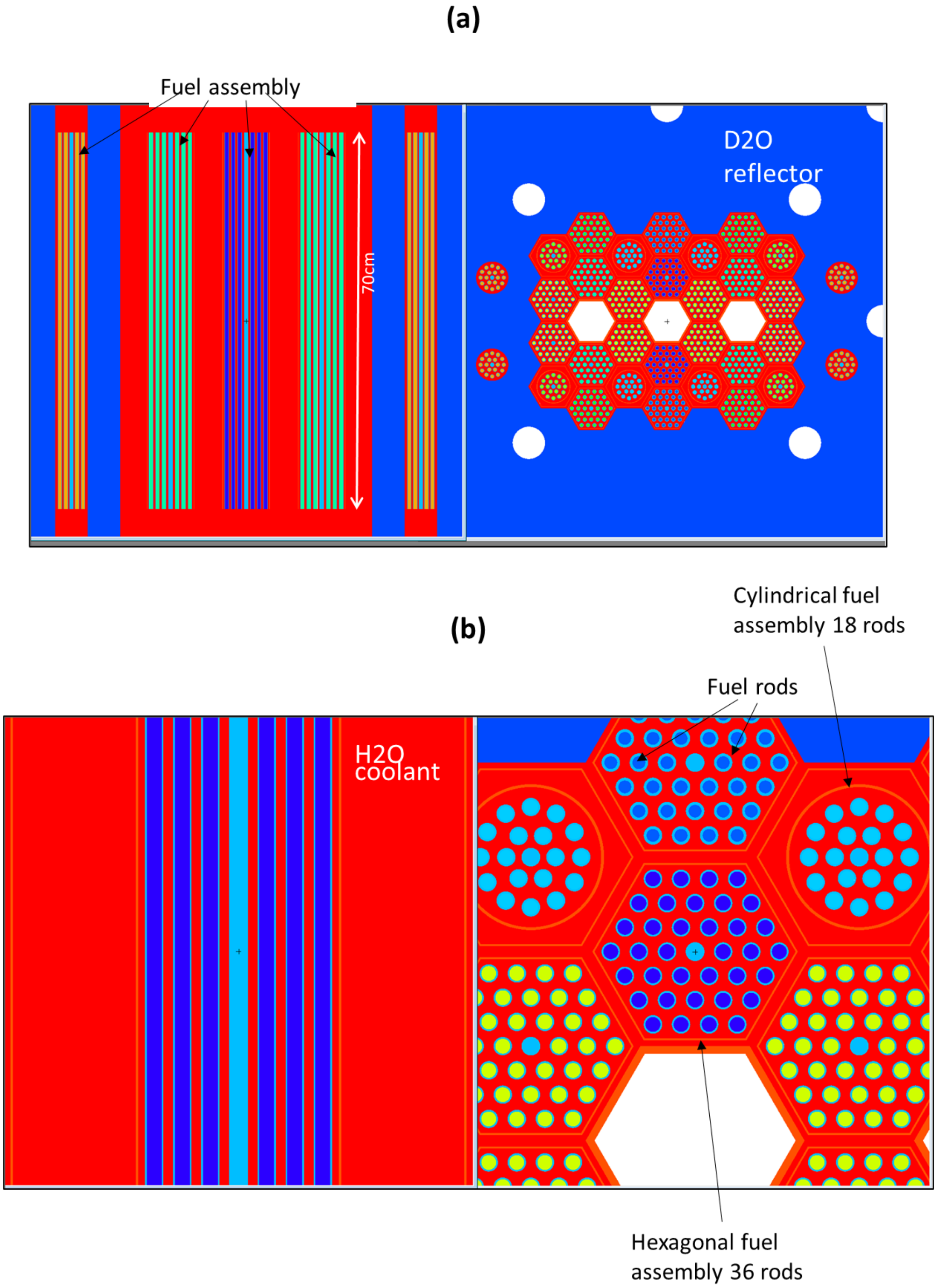

Figure 2. MCNP model of the core, fuel assemblies and rods.

A series of key neutronic parameters were calculated to characterize the reactor's behavior. The k_{eff} was determined using MCNPX's KCODE mode, which simulates successive neutron generations. The neutron flux (ϕ) tallies were obtained via the track length estimator (F4 tally), which measures the total path length of neutrons in a cell and normalizes it by the cell volume and number of source particles. These calculations provided detailed maps of neutron flux across the core, reflector, and irradiation channels.

To obtain absolute quantities and relate the MCNP results to the reactor's actual power level, a specific scaling procedure was implemented. Initialy, the average energy release per fission (E_f) was computed, which is approximately 200 MeV or 3.204×10^{-11} J. This value was used to calculate the fission rate (R_f) necessary to achieve the target power (P) of 30 MWth using the formula R_f=P/E_f. The MCNP tallies, which are initially given per one source neutron, were then scaled to match this required fission rate to obtain absolute quantities like absolute flux and absolute power density (Los Alamos National Laboratory, 2011). This method ensures that the simulation accurately reflects the physical conditions of a 30 MWth reactor. All MCNPX calculations were performed using the 'KCODE' card and an additional 'BURN' card for burnup analysis. To ensure statistical reliability, the error for k_{eff} results was kept below 0.00030, while the error for both the flux and mesh tallies was maintained below 5% by adjusting the particle history and the number of cycles.

Finally, the burnup of the fuel was simulated to understand the long-term behavior of the core over its operational cycle. The analysis considered the typical HANARO operational cycle of 28 days on followed by 14 days off for maintenance and refuelling (Shin, 2023). The simulations captured the effects of fuel depletion and the accumulation of fission products, highlighting the temporary reactivity reduction due to xenon poisoning at the start of each operational cycle. This time-dependent behavior provides essential information for fuel management and predicting the reactor core’s end-of-life. Nevertheless, it should be noted that the observed power and reactivity variations are driven solely by changes in the burnup input in the MCNPX burn card, as actual control rod movements were not included in the model.

RESULTS AND DISCUSSION

Multiplication factor and burnup

Figure 3 visually represents the simulated operational characteristics of the MCNPX model over an irradiation campaign spanning approximately 250 days, detailing the reactor power history, neutron multiplication factor (k_{eff}), average core burnup, and U-235 depletion. This simulation provides a dynamic representation of the reactor's performance under realistic operating conditions, benchmarked against the established HANARO research reactor. The burnup calculations were performed using the MCNPX Tier-3 depletion option, which tracks the full inventory including fission products such as Xenon and Samarium. The time-dependent power history for a typical cycle was modeled using variable step durations: 1-day steps at the beginning of the cycle to capture rapid reactivity changes, 2-day and 4-day steps during intermediate periods, and 7-day steps for the remaining cycle to efficiently model long-term isotopic evolution. This sequence of time steps was repeated for a total of six operational cycles, allowing accurate simulation of fuel depletion, fission product buildup, and associated reactivity effects over extended operation.

Plot (a) illustrates the prescribed power schedule at operating mode of HANARO, it is a crucial input in this study that precisely mimics the 28-day full-power operation interspersed with 14-day zero-power intervals. This stepwise power profile is essential for accurately simulating transient phenomena, particularly the build-up and decay of fission product poisons such as Xe-135, power-dependent thermal-hydraulic reactivity feedbacks, and the cumulative fuel depletion. Plot (b), depicting the evolution of k_{eff}, exhibits two primary features: a monotonic downward trend and superimposed sawtooth oscillations. The long-term decline is a direct consequence of the progressive destruction of fissile material (U-235) and the accumulation of non-fissile fission products (e.g., samarium-149 and stable isotopes), both of which introduce negative reactivity. The characteristic sawtooth oscillations are generated by the Xe-135 transient; during the 28-day power-up period, Xe-135 builds up to its equilibrium concentration, suppressing k_{eff}. During the subsequent 14-day zero-power shutdown, the Xe-135 rapidly decays, leading to the sharp increase (or 'sawtooth') in k_{eff} at the start of each new cycle.

The long-term performance indicators, Plots (c) and (d), provide critical metrics for fuel management. Plot (c) presents the average core burnup, expressed in GWd/MTU (Gigawatt-days per Metric Ton of Uranium). The plot shows a clear incremental increase only during the 28-day full-power phases, plateauing during the shutdown periods. The final predicted average burnup of approximately 98–99 GWd/MTU aligns remarkably well with reported discharge burnup data from HANARO (Chae et al., 2018). This correspondence strongly suggests that the MCNPX model accurately simulates the rate of energy extraction from the fuel. Plot (d) shows the depletion of U-235 over time, which is intrinsically linked to burnup accumulation. The simulated final depletion level approaches ~60%, which is quantitatively comparable to the 59.1% U-235 depletion values reported from extended irradiation experiments in the reference reactor (Chae et al., 2018). Interestingly, the simulation demonstrates that k_{eff} gradually decreases to nearly 1.0 after about six operating cycles (approximately 175 full-power days, 6×28 days), which correlates well with HANARO's typical fuel residence time before full discharge.

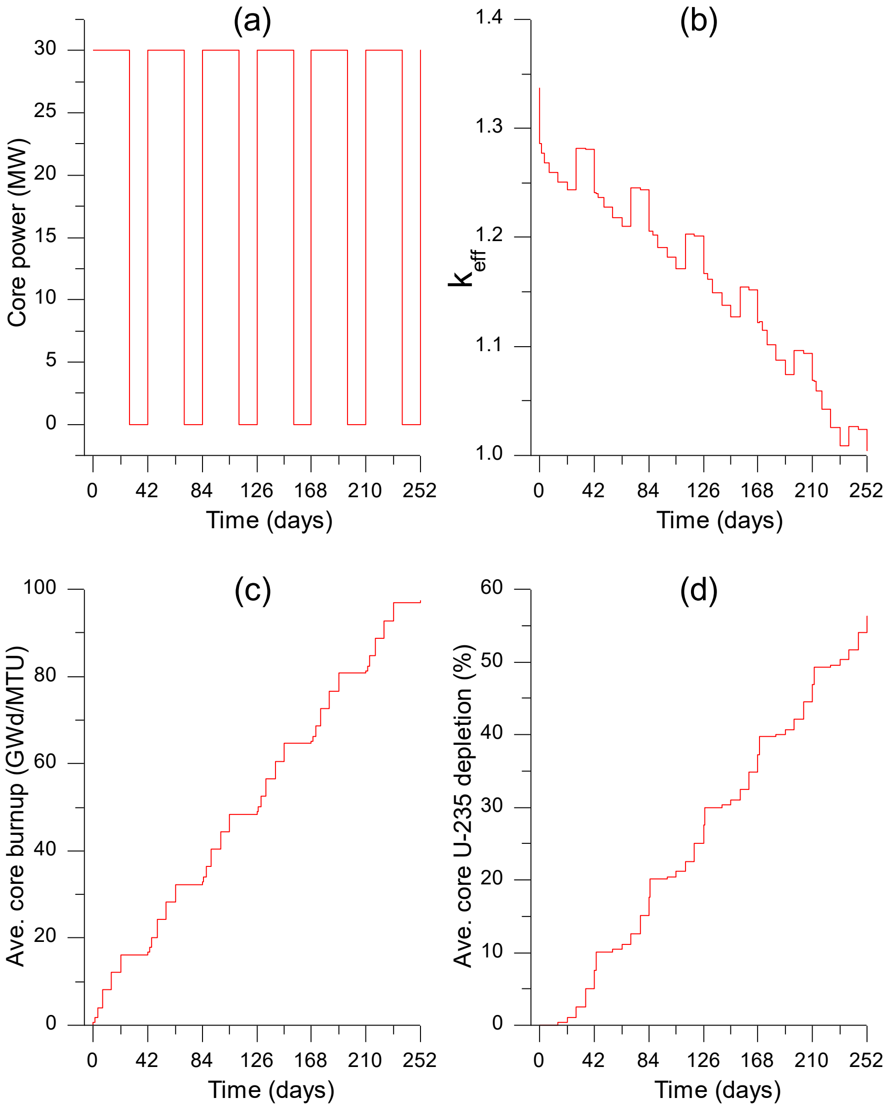

Figure 3. Simulated HANARO operation showing (a) power history, (b) k_{eff} trend, (c) burnup progression, and (d) U-235 depletion.

Neutron flux distribution

Comparison of neutron flux distributions is crucial because flux represents the spatially and energy-resolved parameter that directly controls irradiation conditions, material testing environments, and isotopic evolution. While k_{eff} and burnup validate overall reactor behavior, flux validation confirms the model can accurately reproduce the local neutron field in both magnitude and spectrum, which is vital for reliably predicting reaction rates, radionuclide inventories, and the effective use of irradiation facilities. Figure 4 presents this validation, detailing a comparison of neutron fluxes predicted by the MCNPX model against measured HANARO data at four key irradiation facilities—CT, IR, NTD, and OR—across two energy ranges: thermal (<0.625 eV) and fast (>0.82 MeV) (Chae et al., 1996). For the thermal flux distribution, the MCNPX simulation shows excellent agreement in the core-adjacent regions (CT and IR), which exhibit the highest flux levels (~10^{14} n·cm^{2}.s^{-1}), with differences of less than 3%. However, larger deviations occur in the moderated regions: the calculated thermal flux overestimates the NTD data by nearly 50%, while it underestimates the OR data by over

26%. For the fast neutron flux distribution, a similarly good agreement is observed in the high-flux regions (CT and IR), though the model's precision is slightly lower, with differences ranging from -10% to nearly -30%. In the moderated regions, the discrepancies are more pronounced, as the modeled fast flux at the NTD location, approximately five times higher approximately 6.23E11 in the model and 1.11E11 in the HANARO. The larger deviations can be attributed to uncertainties in the irradiation facility model, particularly in the geometric representation and the exact positioning of the irradiation channels. In most facilities, such as the NTD, the precise design details and actual locations of the channels are not publicly available. Consequently, the MCNPX model must rely on assumed or approximate positions and simplified geometries, which can introduce disscrepancies between calculated and measured neutron fluxes, especially in regions sensitive to local heterogeneities or shielding effects.

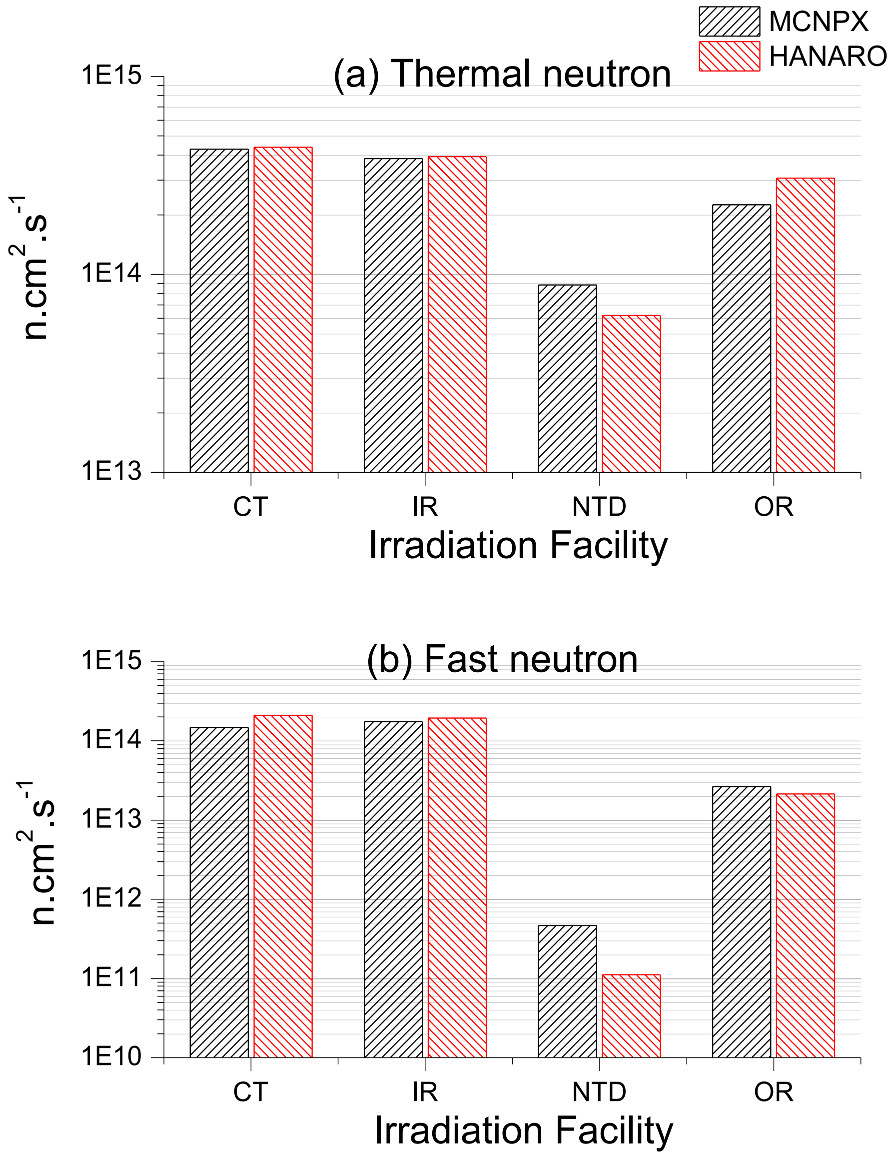

Figure 4. Thermal (a) and fast (b) neutron flux comparison between MCNPX and HANARO data at CT, IR, NTD, and OR.

Figure 5 presents the neutron flux spectra at three representative irradiation sites of the reactor core. All spectra exhibit the characteristic thermal peak around 10^{-8}–10^{-7} MeV, consistent with a Maxwellian distribution typical of light water moderated systems. The CT achieves the highest thermal neutron flux, exceeding 10^{13} n/cm²·s·MeV, followed closely by the OR, while the NTD remains distinctly lower, reflecting the intentional suppression of epithermal and fast components to meet the specific requirements of silicon doping applications. In the epithermal region (10^{-6}–10^{-3} MeV), the CT maintains a slightly elevated flux compared to the OR, owing to its central position within the core, whereas the NTD shows a marked reduction. A similar trend is observed in the fast energy region (>0.1 MeV), where the CT retains the strongest flux, followed by the OR, with the NTD again strongly attenuated. These spectral differences highlight the functional specialization of each facility: the CT provides the highest flux levels suitable for demanding fuel and materials testing, the OR serves as a flexible site for general irradiation purposes, and the NTD offers a moderated environment optimized for semiconductor applications. The overall spectral shapes and relative magnitudes are consistent with reported measurements and simulations of HANARO.

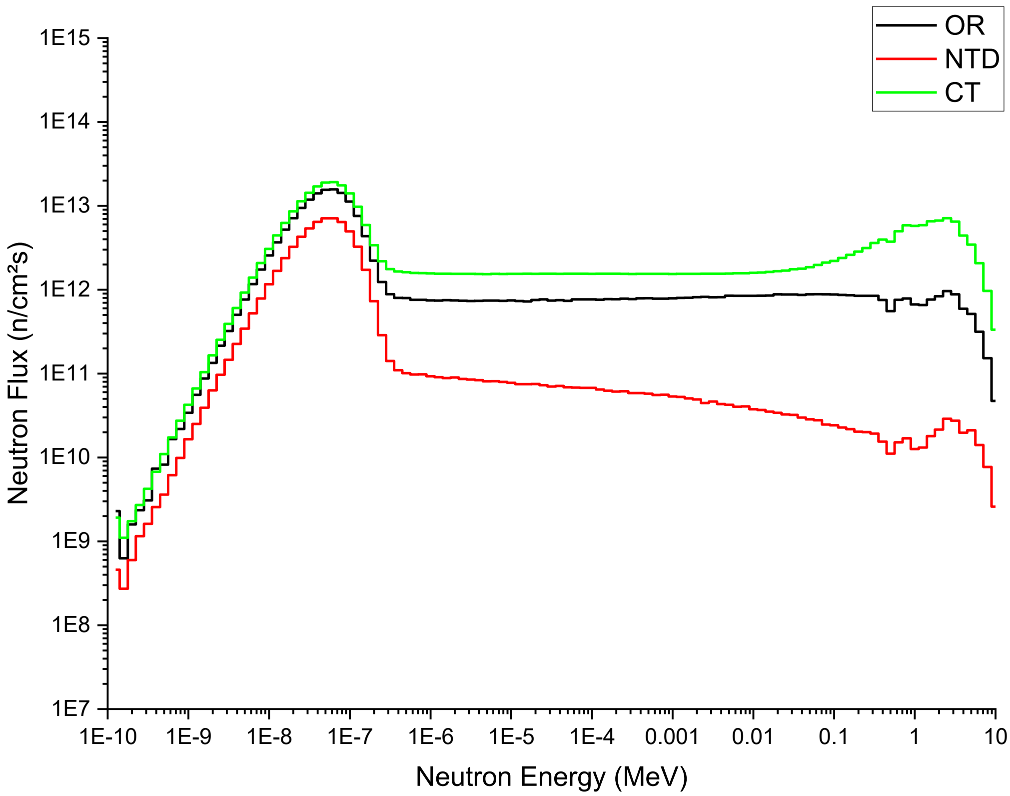

Figure 5. Neutron flux spectra at the CT, OR, and NTD facilities.

Having successfully validated the MCNPX model against experimental benchmarks for k_{eff}, burnup, and flux values, the model was then applied to generate a detailed neutron flux map of the core model. Figure 6 illustrates the MCNPX-calculated neutron flux distributions in the MCNPX core model, showing the core configuration (a) and the fast (b) and thermal (c) flux profiles. The top (a) panel provides a schematic of the reactor core, depicting the central active fuel region (hexagonal configuration) encased by the heavy water reflector and the outer pool, while marking key irradiation sites such as NTD positions, OR channels, and central channels (CT, IR1, IR2). The middle panel (b) illustrates the fast neutron flux distribution, showing maximum intensity within the central fuel assemblies. This is consistent with theoretical expectations, as fast neutrons are produced directly by fission events, and their flux naturally diminishes with increasing distance from the fission sites. The spatial localization of high fast-neutron flux directly informs irradiation capabilities, identifying regions suitable for experiments or processes that require high-energy neutrons, such as displacement damage studies or fast-neutron activation (Marsden & Hall, 2012). In contrast, the right panel (c) depicts the thermal neutron flux, which peaks not within the fuel itself but in the surrounding heavy water reflector. This behavior arises from the moderation process: fast neutrons escaping the core are effectively slowed, or "thermalized," by collisions with hydrogen nuclei in the heavy water, creating a spatial region of enhanced thermal flux (Tsuruta, 2002)

Such thermal flux peaks are crucial for irradiation applications requiring high thermal neutron exposure, including isotope production, neutron activation analysis (NAA), and silicon doping. Collectively, the figure confirms the expected inverse spatial distributions of fast and thermal neutrons—fast neutron flux concentrated in the core and thermal flux enhanced in the reflector—while directly linking these distributions to the reactor's practical irradiation capabilities across multiple experimental and industrial applications. Thermal and fast neutron flux maps further define irradiation and beamline conditions, informing applications such as neutron radiography, NAA, and boron neutron capture therapy (BNCT). Flux data also guide beam port design to maximize thermal neutron flux while minimizing fast-neutron and gamma contamination.

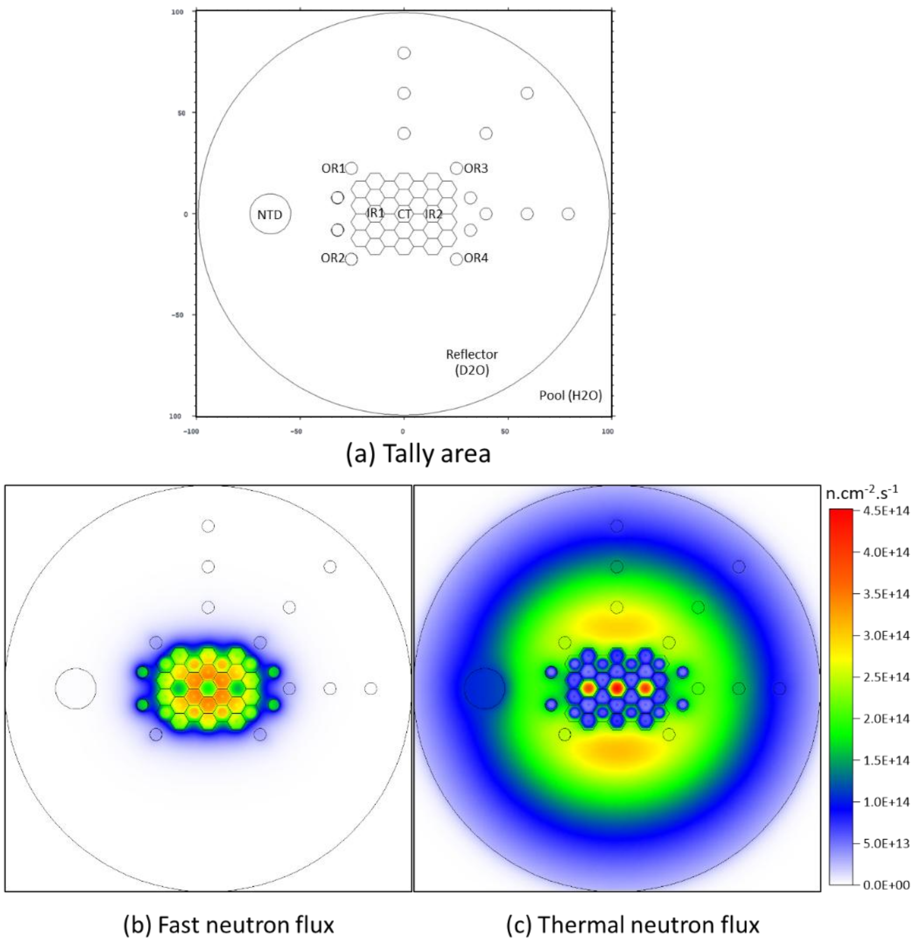

Figure 6. MCNPX-calculated neutron flux distributions.

Power distribution

The neutron flux distribution directly dictates the power distribution since local power density depends on the fission rate, which is governed by the flux. Figure 7 presents the simulated power density distribution across the core, generated using MCNPX. The distribution shows the expected profile, with maximum power concentrated at the central region and decreasing towards the periphery, consistent with the neutron flux profile and diffusion theory predictions (Lamarsh & Baratta, 2001). Within individual hexagonal assemblies, a clear "power peaking" effect is observed, where fuel rods along the outer edges exhibit higher power densities than those at the center. This intra-assembly peaking arises because thermalized neutrons from the water moderator between assemblies preferentially enhance fission at the fuel–moderator boundary, a well-established effect in lattice physics. Both the core-wide and intra-assembly patterns are consistent with reactor physics theory and previous validated neutronic analyses of HANARO and similar research reactors.

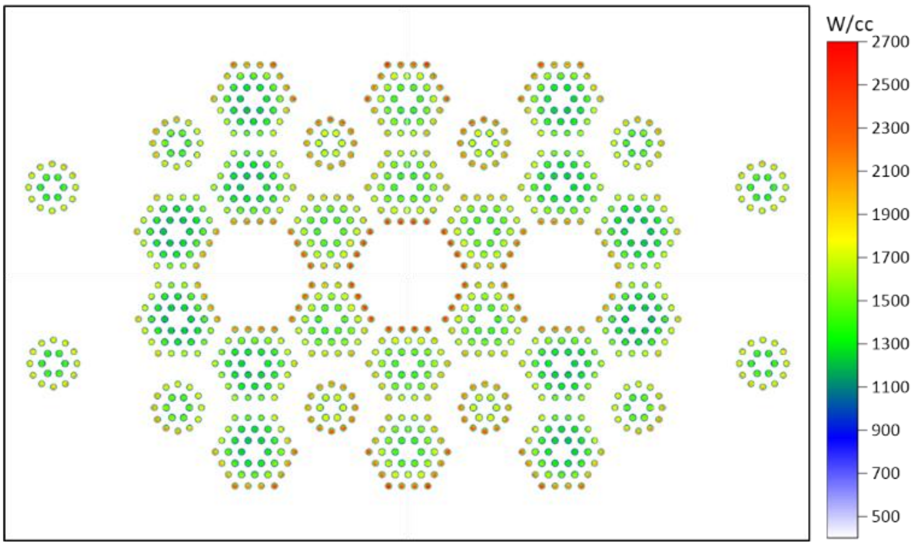

Figure 7. MCNPX-calculated power density distributions.

Figure 8 presents the analysis of power peaking factors for selected fuel assemblies (FAs) in the MCNPX core model. The top panel (a) illustrates a quarter-core symmetry model, a standard approach in neutronic simulations that reduces computational demand while maintaining fidelity. Within this model, nine fuel assemblies (FA1–FA9) are represented, with fuel rods and the surrounding heavy water reflector in blue. Additional core features, such as an empty channel and a light water region, are also included. The arrangement of fuel assemblies, whether centrally located or positioned near the periphery, strongly influences neutron flux distribution and local power generation (Lamarsh & Baratta, 2001). The right bottom plots the power peaking factor of each assembly as a function of operational time. The power peaking factor, defined as the ratio of the maximum power density in a given assembly to the average power density across the core, provides a measure of relative power contribution. Assemblies positioned near the reflector (FA2, FA3, FA5, and FA8) exhibit factors greater than unity, as neutron reflection enhances their local flux and power output. In contrast, centrally located assemblies (FA1 and FA7) remain close to or slightly below unity, consistent with the more uniform neutron flux in the core interior. FA6 and FA9 exhibit power peaking factors below 1.0 despite being adjacent to the heavy water reflector. This behavior can be attributed to their comparatively greater distance from the core center, which reduces the

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overall neutron flux available to these positions relative to other peripheral fuel assemblies. FA4 shows the lowest factor, remaining below 0.9, due to its placement near a region with reduced thermal neutron flux.

In summary, the maximum assembly-wise power peaking factors are observed in peripheral assemblies FA2, FA3, FA5, and FA8, remaining below 1.1, while the highest pin-wise peaking occurs along the fuel–moderator boundaries within these assemblies. Central assemblies (FA1 and FA7) and other peripheral assemblies (FA6, FA9) show factors at or below unity, reflecting a uniform and well-controlled power distribution. These results indicate that all fuel rods operate within safe thermal margins, with no risk of localized overheating, supporting the overall safety and reliable performance of the core.

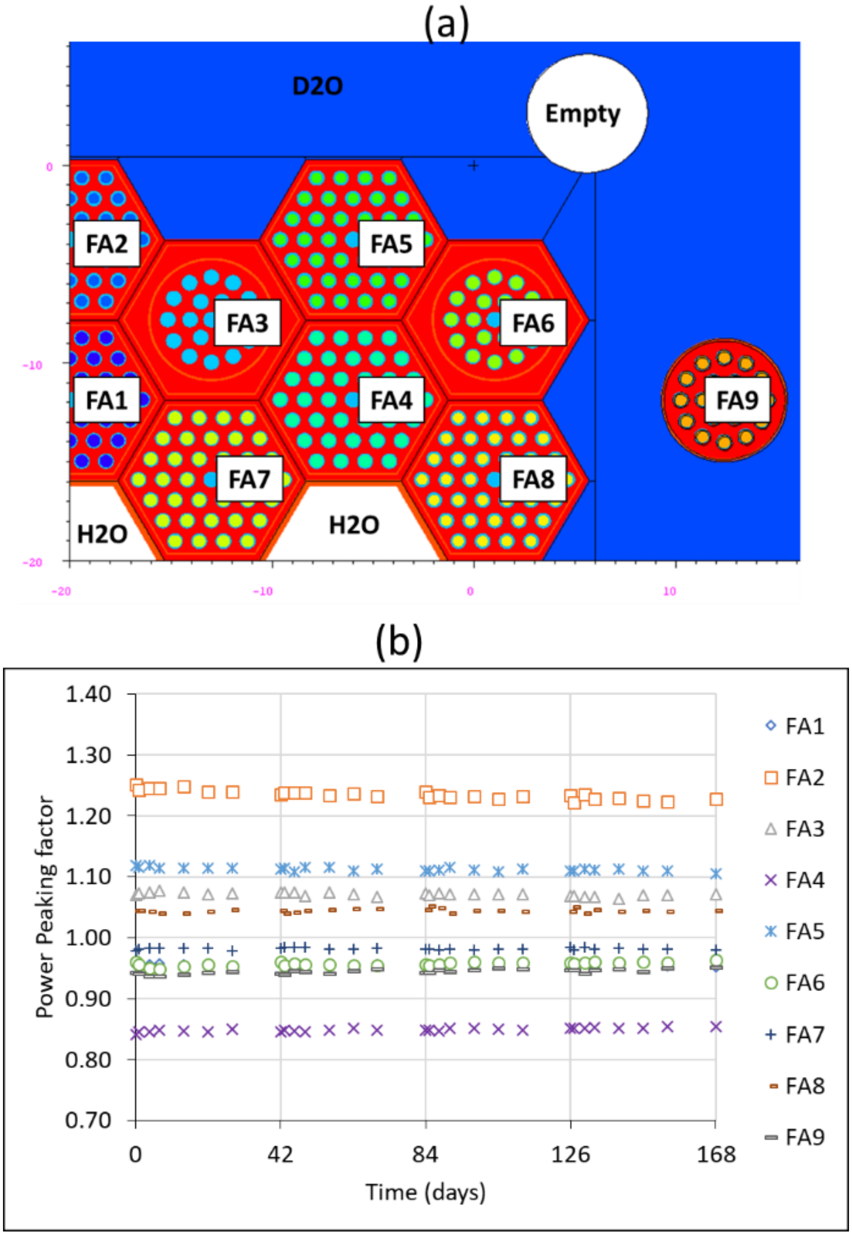

Figure 8. MCNPX-calculated power peaking factor.

PRELIMINARY CORE DESIGN CONSIDERATIONS FOR THE MALAYSIAN MPR

Based on the current, limited analysis and following the successful validation of the MCNPX model against the HANARO reference reactor, this section tentatively outlines a preliminary core design and operational concept for the proposed Malaysian MPR. This initial concept attempts to leverage the validated model to ensure the reactor could potentially fulfill national objectives in research, isotope production, and neutron beam aplication.

Core Design and Operational Concept

For this initial conceptualization, the Malaysian MPR might adopt a hybrid configuration, drawing a parallel to HANARO, by combining 36-rod hexagonal and 18-rod cylindrical fuel assemblies. The core is envisioned to consist of 32 assemblies utilizing low-enriched uranium silicide dispersion fuel, clad in aluminum, with an enrichment of 19.75%. A heavy water reflector is proposed to potentially enhance neutron economy and increase thermal flux in the irradiation channels. Preliminary simulations suggest a central thermal neutron flux of approximately 4.3×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, a level which should be sufficient for a broad range of high-flux neutron applications.

Fuel management is currently based on a batch refueling scheme, with 28-day operating cycles followed by 14-day shutdowns for maintenance and shuffling. The model estimates a fuel residence time of about 175 full-power days, corresponding to an average burnup potentially exceeding 50% of the initial U-235. This design is intended to ensure efficient fuel utilization, compliance with international LEU standards, and operational flexibility. The calculated k_{eff} values appear to confirm that the fresh core could offer adequate excess reactivity to remain supercritical over each cycle, even when accounting for xenon transients.

Based on HANARO's operational practices, a proposed refueling scheme would adopt a frequent, partial core replacement strategy (Rho, 1997; Lee et al, 2009). This method is designed to maintain a consistent, high neutron flux necessary for continuous research and isotope production. The reactor would tentatively operate on a short, fixed 28-day cycle of full-power operation, followed by a scheduled 14-day outage for maintenance and fuel handling, resulting in a refueling opportunity approximately every six weeks. During each of these planned outages, only a small batch of the most depleted fuel assemblies would be discharged. The remaining partially-spent assemblies would be strategically shuffled to new, optimized core locations to maximize their energy use and flatten the power distribution. Finally, a new batch of fresh fuel would be loaded into the vacant positions.

Spent Fuel Management Considerations

As a point of reference, HANARO operates on a partial core refueling scheme, typically replacing 7–8 fuel assemblies per year. Annual refueling removes only a fraction of the core at a time to maintain core reactivity and ensure stable power operation. This gradual replacement strategy is shown to support effective burnup management and preserves safety margins, with discharged fuel remaining in the wet storage pool pending further management or disposal Lee et al, 2009). Each standard fuel assembly is known to contain approximately 3.0–3.2 kg of uranium, with 19.75% enrichment in U-235. Based on the average annual replacement of 7–8 assemblies, approximately 23–25 kg of total uranium is discharged per year, comprising

~4.65 kg of U-235 and ~18.6 kg of U-238. Assuming continuous operation for a typical research reactor lifespan of 50 years, a total of approximately 375 fuel assemblies would be discharged, implying that the wet storage pool must have sufficient capacity to store at least this number of assemblies safely, accounting for decay heat removal and radiation shielding. These data provide a quantitative basis for estimating the annual fuel inventory, burnup management, and core fuel strategy, and serve as a valuable reference for designing spent fuel handling practices in similar research reactors.

Over a 50-year lifespan, HANARO's partial core refueling of 7–8 assemblies per year would discharge approximately 375 assemblies, totaling about 1,163 kg of uranium, including 230 kg of U-235. With an effective burnup of roughly 30% per assembly, around 69 kg of U-235 would be consumed, allowing gradual fuel utilization while maintaining core reactivity and safety margins. The wet storage pool must accommodate all discharged assemblies, ensuring adequate decay heat removal and radiation shielding.

Preliminary Utilization Planning

The proposed Malaysian MPR could potentially host a range of irradiation facilities to meet anticipated research and industrial demands. Approximately six in-core irradiation sites are envisioned to be placed within the active core and reflector to support high-flux applications such as neutron transmutation doping of silicon and radioisotope production. For conventional silicon, HANARO's NTD thermal flux of 8.83×10^{13} n·cm^{-2}·s^{-1} is sufficient to produce high-quality semiconductor-grade material. For emerging semiconductors like silicon carbide (SiC), which rely on fast neutron transmutation, the required fluence is typically 10^{18}–10^{19} n·cm^{-2}. At the NTD facility, the fast flux is only ~10^{11} n·cm^{-2}·s^{-1}, making it impractical for SiC doping, whereas in the central trap, where fast flux reaches ~10^{14} n·cm^{-2}·s^{-1}, the required fluence could be achieved in roughly 1–2 days of continuous irradiation, demonstrating that high-flux regions are essential for producing doped SiC efficiently while HANARO's NTD thermal flux remains suitable for conventional silicon. Similarly, the central trap and inner reflector regions, with fluxes potentially exceeding 10^{14} n·cm^{-2}·s^{-1}, would enable efficient production of medical isotopes such as molybdenum-99 (Mo-99), the precursor of technetium-99m (Tc-99m). In addition, 15 out-of-core irradiation sites are planned around the reflector and pool, operating at lower fluxes to support activities such as neutron activation analysis and materials testing under controlled neutron exposure. For comparison, HANARO's substantial in-core capacity, comprising seven irradiation holes (three hexagonal and four circular), complemented by over ten reflector positions and specialized facilities including beam tubes and NTD sites (Choo et al., 2014; Lee, Ahn & Lim, 2007).

CONCLUSIONS

The preliminary neutronic design and analysis for the Malaysian Multi-Purpose Research Reactor, benchmarked against the 30 MWth South Korean HANARO reactor, has successfully achieved all its stated objectives and established a robust technical foundation for the project. The study confirmed the suitability of the methodology and reference design. A detailed MCNPX model of the HANARO reactor was successfully developed and validated against experimental data for key parameters such as effective multiplication factor, burnup, and neutron flux values at various irradiation sites. The model demonstrated strong agreement, confirming its accuracy and reliability as a predictive tool. The validated model was then used to propose a preliminary core design and fuel management strategy for the Malaysian MPR. This included defining the hybrid fuel configuration, predicting a core residence time of 175 full-power days, and analyzing the core's operational behavior and power distribution. The insights gained from this analysis, particularly regarding the effects of the heavy water reflector and power peaking, are essential for optimizing fuel management and ensuring the reactor meets its safety and performance goals.

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DECLARATION OF COMPETING INTEREST

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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