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Validation of the Batan-3DIFF Code against Fission Chamber Measurements for In-Core Thermal Neutron Flux in the RSG-GAS Reactor

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A B S T R A C T

The accurate determination of neutron flux distribution is essential for reactor physics analysis and supports various applications, including material irradiation and radioisotope production. This study presents a comparative analysis of the axial thermal neutron flux distribution, evaluating results from the deterministic diffusion code Batan-3DIFF against experimental measurements obtained using a fission chamber detector. Measurements were performed at three irradiation positions—D-7, E-7, and G-7—within the RSG-GAS reactor core. At position D-7, the Batan-3DIFF calculation yielded a maximum thermal neutron flux of approximately 1.34×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, while the fission chamber measurement recorded a slightly lower value of 1.26×10^{14} $n\cdot cm^{-2}\cdot s^{-1}$, corresponding to a relative deviation of 6.0%. Similar levels of discrepancy were observed at positions E-7 (6.7%) and G-7 (6.8%), with the computational results consistently overestimating the measured flux. These systematic deviations are primarily attributed to the geometric and material homogenization approximations inherent in the diffusion model, as well as differences in the neutron energy response of the fission chamber compared to the modeled spectrum. Despite these minor discrepancies, the overall agreement between the calculated and experimental flux profiles confirms that Batan-3DIFF is capable of reliably representing axial neutron flux distributions in the RSG-GAS reactor.

1. INTRODUCTION

The G.A. Siwabessy Multipurpose Reactor (RSG-GAS) is a 30 MWth open-pool research reactor that has been operational since 1987 at the Puspiptek Science Center in Serpong, Indonesia. It utilizes low-enriched uranium silicide (U_3Si_2-Al) fuel with a ^{235}U enrichment of 19.75% and a uranium density of 2.96 gU/cm³. The equilibrium core configuration comprises 40 standard fuel elements and 8 control fuel elements, all of the plate-type

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design. Schematic illustrations of the fuel and control element geometries, along with the full equilibrium core layout, are provided in Figures 1 and 2 [1–3]. The RSG-GAS reactor employs light water as its primary coolant and neutron moderator, complemented by a beryllium metal reflector. Designed as a multipurpose facility, its core functions encompass radioisotope production (e.g., ^{99}Mo), neutron activation analysis, materials testing, and dedicated irradiation services through various in-core and ex-core facilities. The reactor's flexible core configuration enables the generation of a high and relatively uniform thermal neutron flux across multiple irradiation positions, making it a versatile neutron source for fundamental research, as well as industrial and medical applications.

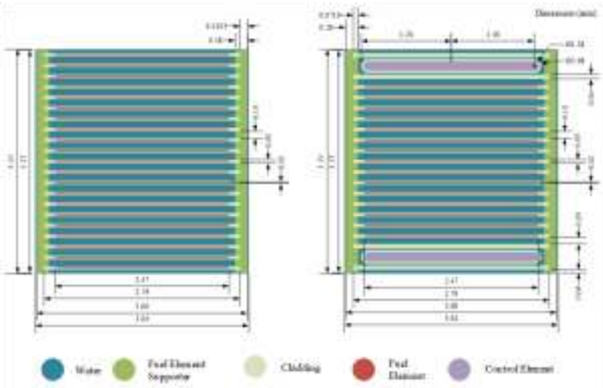

Figure 1. Fuel element (left) and control element (right) of RSG-GAS (units in mm) [4].

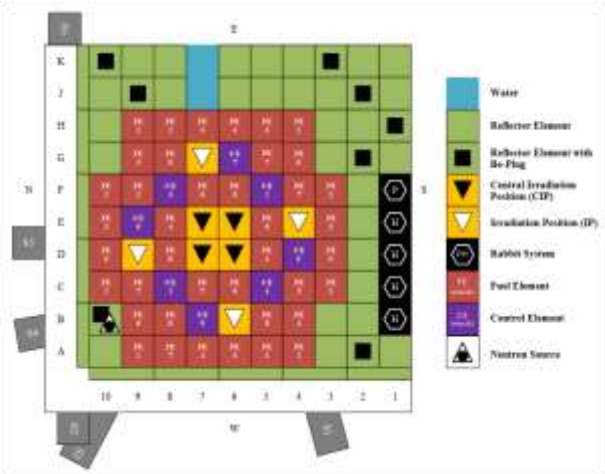

Figure 2. Equilibrium core configuration of the RSG-GAS reactor [1].

The neutron flux distribution is a fundamental parameter governing both the performance and safety of a research reactor, spanning the core and its associated irradiation facilities. Precise knowledge of the neutron flux is critical not only for predicting experimental results such as isotope yield and material damage but also for ensuring the fulfillment of reactor operational safety requirements [5]. Neutron flux measurements in research reactors are conducted using the foil activation method [6, 7]. Although well-established and reliable, the foil activation method is inherently time-consuming and introduces several sources of uncertainty, including self-shielding effects within the foil and dependencies on specific nuclear data libraries. In contrast, the fission chamber detector provides a direct, real-time measurement capability with high neutron sensitivity, offering a significant operational advantage for in-core flux monitoring. The detector's operation is based on fission reactions within a fissile coating, typically ^{235}U deposited on the electrode of an ionization chamber. The resulting fission fragments ionize the chamber's fill gas, generating an electrical current directly proportional to the incident neutron flux. This study employed a Sub-miniature Fission Chamber (SMFC) CFUZ53, characterized by an ultra-compact outer diameter of 1.5 mm. The detector incorporates 10 µg of highly-enriched uranium (90% ^{235}U) and is filled with argon gas at 110 kPa (1.1 atm) [8]. With a calibrated thermal neutron sensitivity of $5x10^{-18}$ A/n.cm^{-2}.s^{-1}, the SMFC was operated in current mode, enabling neutron flux measurements up to $1x10^{14}$ n·cm^{-2}·s^{-1}. These characteristics render the fission chamber an ideal instrument for benchmarking calculated neutron flux distributions against experimental data.

This study aims to benchmark the three-dimensional neutron flux distribution calculated by the Batan-3DIFF code, a multigroup diffusion code developed by BATAN, against direct experimental measurements from a fission chamber in the RSG-GAS reactor [9]. The Batan-3DIFF code solves the three-dimensional neutron diffusion equation to predict the flux distribution throughout the reactor core, explicitly modeling the geometry of fuel assemblies, control elements, and reflector. While it is extensively used for the design and safety analysis of the RSG-GAS reactor, its predictive accuracy requires experimental validation. This study provides a direct evaluation of the code's performance in modeling neutron flux by benchmarking its results against in-core fission chamber measurements. The findings are expected to enhance reactor analysis methodologies, reduce flux prediction uncertainties, and support the safe and optimized utilization of the RSG-GAS reactor.

2. METHODOLOGY

Batan-3DIFF Code Calculation

The thermal neutron flux distribution was computed using the three-dimensional, multi-group diffusion code Batan-3DIFF. The code iteratively solves the multi-group neutron diffusion equations via the finite difference method. Reflective boundary conditions were applied to the symmetry planes, while the outer reflector boundary was modeled with a vacuum condition. The simulation was configured to replicate the experimental operating conditions, including reactor power level, control rod bank positions, and the presence of neutron-absorbing fission products (e.g., ^{135}Xe and ^{149}Sm).

Few-group homogenized cross-section data for the fuel elements, moderator, reflector, and control rods were generated using the continuous-energy Monte Carlo code Serpent 2 [10]**.** The resulting cross-section data were subsequently condensed into a 16-energy-group structure compatible with the Batan-3DIFF scheme [11]. The neutron cross sections for the fuel and control elements were calculated at the lattice level, achieving an accuracy of up to 20 pcm in the infinite multiplication factor. The calculation covers conditions from fresh fuel to 170 MWd/kgU burnup, under both cold and hot operating temperatures.

The full-core model, constructed in three-dimensional (XYZ) geometry, explicitly represented the actual RSG-GAS equilibrium core configuration: 40 standard fuel elements, 8 control elements, a beryllium reflector, and all relevant structural components (e.g., core shroud, reflector shroud), coolant (water), and irradiation facilities. For non-fuel regions, homogenized cross-sections were extracted directly from this full-core model. All cross-section data were generated using the ENDF/B-VIII.0 nuclear data library, collapsed to a 16-group structure with enhanced resolution in the thermal energy region. Simulations employed a substantial number of neutron histories to ensure statistical uncertainties were negligible for the analysis. Simulations employed a substantial number of neutron histories so that statistical fluctuations remained insignificant. The computational setup was subsequently refined by increasing the number of neutron histories to 200,000 and executing 1000 cycles, including 100 inactive cycles. The resulting multi-group neutron flux distributions from the calculation were subsequently compared with experimental measurements.

Fission Chamber Measurement

Neutron flux measurements were performed using the SMFC CFUZ53 detector, which utilizes a thin coating of ^{235}U as the fissile target material [12]. The output current, generated by ionization from fission events within the chamber, was continuously

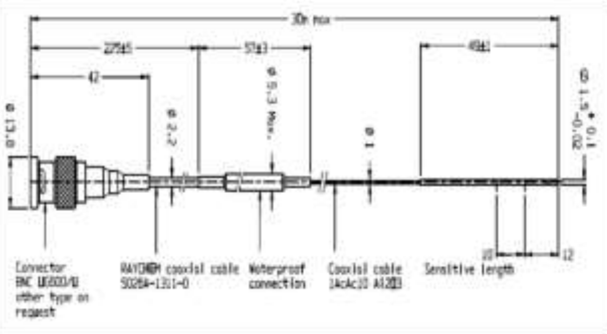

Figure 3. Schematic of SMFC CFUZ53.

measured and recorded during the steady-state reactor operation at full power. The schematic configuration and dimensional specifications of the SMFC CFUZ53 detector are presented in Figure 3.

The raw current data were corrected for background signal and acquisition dead time, then normalized to the reactor's thermal power (MWth) to obtain neutron flux per unit power. To address the detector's energy-dependent response, a spectral correction factor was applied, derived from either the manufacturer's calibration certificate or a direct comparison with foil activation reference measurements. This procedure enabled a direct, like-for-like comparison between the corrected experimental flux values and the computational results. The agreement was quantitatively assessed by calculating the average relative deviation, standard deviation, and root-mean-square error (RMSE). Finally, the axial and radial neutron flux profiles were visualized by superimposing the calculated curves onto the experimental data points.

In-core neutron flux measurements within the RSG-GAS reactor were conducted using the sub-miniature fission chamber (SMFC) CFUZ53 detector. The detector was positioned at axial locations D-7, E-7, and G-7 within the irradiation facilities. Upon exposure to neutron flux, neutron-induced fission reactions in the detector's ^{235}U coating generate fission fragments, which ionize the fill gas and produce a measurable electrical current. This output current was converted into a thermal neutron flux value using Equation (1) [13].

$$\phi = \frac{I}{S} \tag{1}$$

where,

ϕ = Thermal neutron flux (n.cm^{-2}.s^{-1})
I = Detector output current (A)
S = Detector sensitivity (A/n.cm^{-2}.s^{-1})

3. RESULTS AND DISCUSSION

A comparison of the axial thermal neutron flux at position D-7, as determined by Batan-3DIFF calculations and fission chamber measurements, is presented in Figure 4. The computational result exhibits a peak flux of 1.34x10^{14} n.cm^{-2}.s^{-1} at an axial depth of ~19 cm, whereas the experimental measurement yields a corresponding peak of 1.26x10^{14} n.cm^{-2}.s^{-1} at ~20 cm. The maximum observed discrepancy is 8.00x10^{12} n.cm^{-2}.s^{-1}, corresponding to a relative difference of 6.0%. This level of agreement falls within the accepted margin of error for benchmarking deterministic diffusion codes against in-core experimental data.

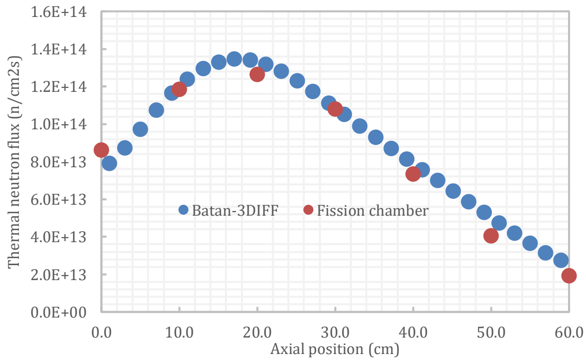

Figure 4. Thermal neutron flux at position D-7.

Figure 5 compares the axial neutron flux profiles from Batan-3DIFF calculations and fission chamber measurements at position E-7. The calculated result shows a peak flux of 1.49x10^{14} n.cm^{-2}.s^{-1} at ~17 cm depth, whereas the experimental measurement yields a peak of 1.39x10^{14} n.cm^{-2}.s^{-1} at ~18 cm. The absolute discrepancy of 1.00 x 10^{13} n.cm^{-2}.s^{-1} corresponds to a relative difference of 6.7%, with the measurement consistently lower than the calculation. This deviation is considered acceptable and can be attributed to several factors: the fission chamber's predominant sensitivity to the thermal neutron spectrum, contrasting with the multi-group total flux representation in Batan-3DIFF, as well as potential minor uncertainties in reactor power normalization for the simulation and the detector's absolute calibration.

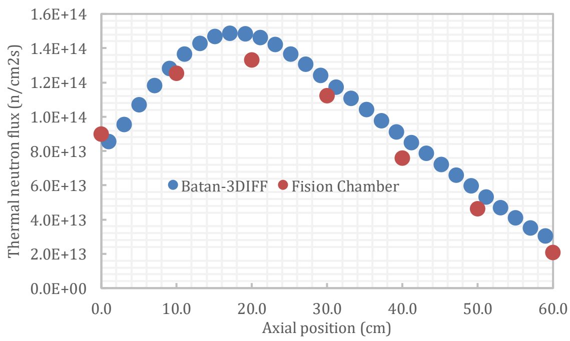

Figure 5. Thermal neutron flux at position E-7.

A comparison of the neutron flux distributions from Batan-3DIFF calculations and fission chamber (FC) measurements at position G-7 reveals discrepancies in both the magnitude of the peak flux and the shape of the axial profile, as depicted in Figure 6. The computational model predicts a maximum flux of 1.17x10^{14} n.cm^{-2}.s^{-1} at a depth of ~20 cm, whereas the experimental data yield a maximum of 1.09x10^{14} n.cm^{-2}.s^{-1} at the same axial position. This corresponds to a relative deviation of 6.8%, with the calculation consistently overestimating the measured flux. The observed discrepancies are attributed to inherent limitations of the diffusion-based computational model, which relies on material homogenization and simplified boundary conditions, in contrast to the FC's direct physical response to the local neutron spectrum. A further contributing factor is the spectral mismatch between the detector's calibrated energy response and the multi-group flux representation used in the Batan-3DIFF simulation.

Overall, the axial neutron flux distributions from both calculation and measurement demonstrate strong agreement, following a characteristic near-cosinusoidal profile typical of research reactor cores. This confirms that the Batan-3DIFF computational model reliably captures the fundamental physics of the reactor system. However, the observed local discrepancies, particularly in peak magnitude, highlight opportunities for model refinement—such as incorporating more detailed geometric definitions or advanced self-shielding treatments—to achieve even closer alignment with experimental data.

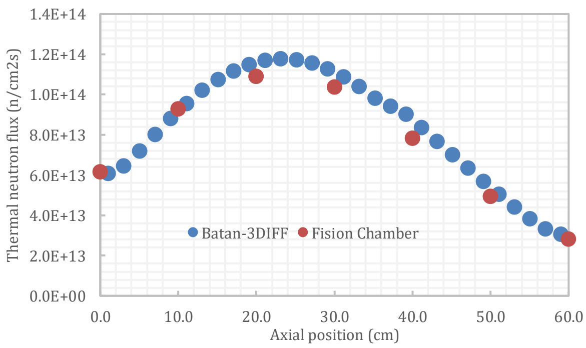

Figure 6. Thermal neutron flux at position G-7.

4. CONCLUSION

The axial neutron flux profiles derived from Batan-3DIFF calculations and fission chamber measurements exhibit strong qualitative agreement, both conforming to the characteristic cosine-shaped distribution expected in RSG-GAS reactor cores. Quantitatively, at positions D-7, E-7, and G-7, the code consistently provides a conservative overestimate. The observed discrepancies are attributed to well-understood factors, including the material homogenization and boundary condition approximations inherent in the diffusion model, coupled with a spectral response mismatch between the fission chamber's sensitivity and the modeled neutron energy groups. Despite these minor differences, the sub-10% deviation confirms that Batan-3DIFF provides an accurate and reliable representation of the in-core flux distribution, demonstrating good consistency between numerical simulation and experimental validation.

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AUTHOR CONTRIBUTION

Ranji Gusman, as the first and corresponding author who conceptualized the research and developed the methodology. Alexander Agung, as the author who guided the overall direction of the research. Mohammad Subekti, as a research supervisor who evaluated the technical methods and analysis process of data. Fitri Susanti contributed to calculating the flux neutron using the code Batan-3DIFF. Surian Pinem co-authored who contributed on the research introduction and discussion on nuclear physics and code Batan-3DIFF.

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