DESIGN OF AN IRRADIATION RIG USING SCREEN METHOD FOR SILICON TRANSMUTATION DOPING AT THE DALAT RESEARCH REACTOR

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Abstract: The neutron transmutation doping of silicon (NTD-Si) at research reactors has been successfully implemented in many countries to produce high-quality semiconductors. In the late 1980s, NTD-Si has been tested at the Dalat Nuclear Research Reactor (DNRR) but the results have been limited. Therefore, the design and testing of an irradiation rig for NTD-Si at the DNRR are necessary to have a better understanding in order to apply the NTD-Si in a new research reactor of the Research Centre for Nuclear Science and Technology (RCNEST), which has planned to be built in Viet Nam. This paper presents the design and testing of a new irradiation rig using screen method for testing NTD-Si at the DNRR. The important parameters in the rig such as neutron spectrum and thermal neutron flux distribution were determined by both calculation using MCNP5 computer code and experiment. The aluminum ingots, which have similar neutronic characteristics with silicon ingots, were irradiated in the rig to verify the appropriate design. The uniformity of thermal neutron flux in the rig is less than 5% in height and 2% in radius, respectively. These values are satisfied the requirements of NTD-Si experiment on the DNRR.

Keywords: *NTD-Si, MCNP5, Dalat Nuclear Research Reactor (DNRR), screen method, foil activation, irradiation rig.*

1. INTRODUCTION

One of the most important requirements of NTD-Si is to maintain the uniformity of radial and axial thermal neutron flux in irradiation channels. At present, the uniformity of irradiation from ± 5 ~ 6% would meet commercial request depend on ingot dimension [1]. In a research reactor, normally, the axial and radial distributions of thermal neutron in irradiation channels are not uniformed. In order to meet customer's requirements, a change in configuration of irradiation channels is needed to uniform thermal neutron flux distribution and maximize usage of the channel. Three major methods could be applied to axially uniform thermal neutron flux distribution are inversion, reciprocating motion and flux screen [1]. Depending on the design characteristics and core configuration of the reactors, the method applied for neutron flux uniformity should be chosen. In the late 1980s, the inversion method has been tested at the DNRR but the results have been limited. Therefore, the design and testing of an irradiation rig using flux screen method for NTD-Si at the DNRR are necessary to have a better understanding and experience in NTD-Si application. The new irradiation rig using various screen materials (stainless steel, aluminum and light water) was designed, installed and tested in the core of DNRR. The results would provide good experiences in the application of NTD-Si on the new research reactor, which has planned to be built in Vietnam.

2. CALCULATION MODEL AND EXPERIMENTAL METHOD

2.1. Dalat reactor and its parameters

The DNRR, 500-kW pool-typed, light water cooled and moderated, was reconstructed and upgraded from the USA TRIGA MARK II reactor. The reactor has been officially put into operation for the purposes of radioisotope production, neutron activation analysis, fundamental and applied research, and manpower training. The summary description of the DNRR is shown in Table 1 and Figure 1 [2].

Table 1. Summary description of the DNRR [2]

Parameter	Description
Nominal power	500 kW
Neutron flux (thermal, max)	2×10^{13} n/cm^2.s
Fuel	VVR-M2, mixed UO_2-Al, 19,75% enrichment
Moderator and coolant	Light water
Reflector	Graphite, beryllium and light water
Core cooling	Natural convection
Heat rejection	Two-loop cooling system
Control rods	2 safety, 4 shim (B_4C) and 1 regulating (stainless steel)

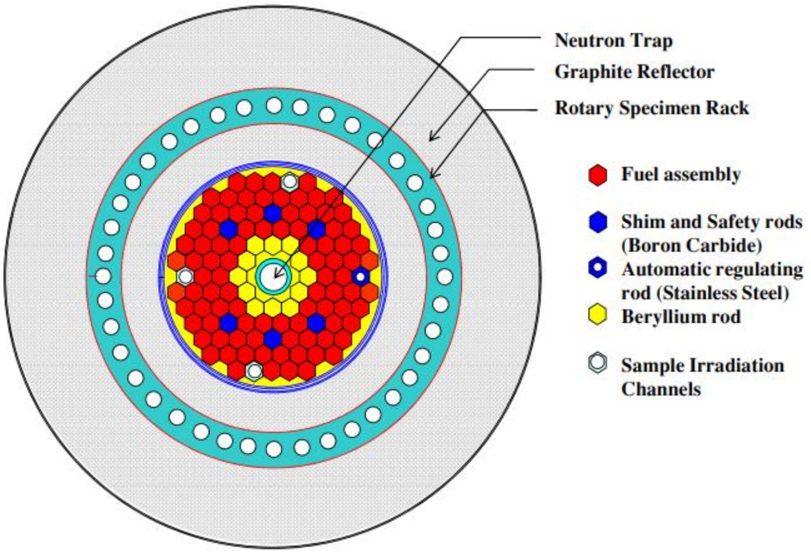

Figure 1. Present working configuration of DNRR.

In the working configuration of DNRR, the neutron trap, which has maximum thermal neutron flux in reactor core is dedicated for NTD-Si testing. Neutron trap is a water cylinder surrounded by Beryllium blocks located in center of the core. The neutron trap has 6.5 cm in diameter and about 2050 cm^3 in volume [2]. An aluminum tube installed in the neutron trap to load irradiation samples which have maximum of 4.2 cm in diameter as shown in Figure 1.

2.2. Determination of neutron spectrum and neutron distribution at the neutron trap

Determination of neutron spectrum and flux distribution at the neutron trap was obtained by using foil activation method. Bare gold foils and Cadmium-wrapped gold foils were irradiated to obtain absolute neutron flux [3]. The following equation can be used to obtain thermal neutron flux:

$$\Phi_{th} = \frac{2.Ae^{-\lambda\tau}}{\sqrt{\pi}N_A\alpha\sigma_{0,act}G_{th}(1-e^{-\lambda T})}\sqrt{\frac{T_n}{T_0}}\left[\frac{A_b(T,\tau)}{m_b}-\frac{A_{Cd}(T,\tau)}{m_{Cd}}\right] \quad (1)$$

Where, m_b - mass of bare gold foil (g); m_{Cd} - mass of Cadmium-wrapped gold foils (g); T - Irradiation duration (s); τ - cooling duration after irradiation (s); t_m - measurement in real time (s); $t_{m,eff}$ - effective time measurement (s); λ - decay constant of nuclide compound (s^{-1}); η - counting efficiency of detector; γ - gamma abundance factor; m - mass of foil (g); α - isotope enrichment; G_{th} - self-shielding factor; N_A - Avogadro constant; A - Atomic number of isotope; G – ratio of isotope in foil.

To determine the thermal neutron flux distribution at the neutron trap, Lu-176 foils which have large thermal neutron absorption cross-section were used. Thermal neutron flux distribution was scaled relatively. The relative thermal neutron distribution was obtained by comparing of ratios of corresponding Lu-176 activities at various positions in the neutron trap.

The method used for neutron spectrum measurement is based on activation of a set of foils and determination of reaction rates. SANDBP program is used to obtain the neutron energy spectra after irradiation of the foils. The SANDBP program is designed to obtain a 'best fit' neutron flux spectrum for a given input set of infinitely dilute foil activities [3], [4].

MCNP5 computer code was also used for calculation of neutron flux distribution and neutron spectrum. This computer code is developed at the Los Alamos National Laboratory, USA [5]. The MCNP5 has been being officially used for core management of DNRR with ENDF/B7.0 library [6]. The calculation model for DNRR using MCNP5 computer code is shown in Figure 2.

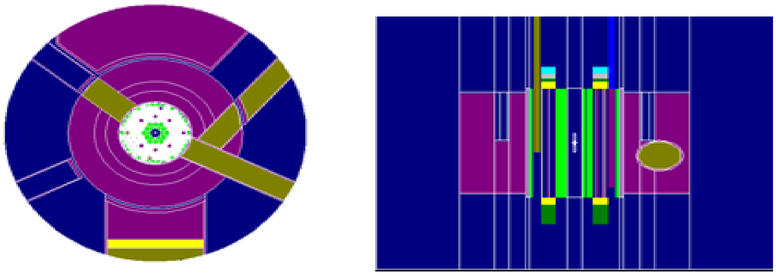

Figure 2. Calculation model of DNRR using MCNP5 computer code.

Neutron spectra divided into 3 energy groups calculated using MCNP5 computer code and neutron spectra obtained from experiments at the neutron trap are shown in **Table 2**. The reliability of simulation by MCNP5 computer code was confirmed through the good agreement of experimental results and calculation. The discrepancies are about 2% in thermal neutron flux and more than 4% in epithermal neutron flux between experiment and calculation results.

Table 2. Comparison of neutron flux of the neutron trap between calculation and experiment.

Flux (n/cm^2.s)	**Calculation**	**Experiment (error ±5-6%)**
Thermal	$2{,}24.10^{13}$	$2{,}29.10^{13}$
eipthermal	$6{,}52.10^{12}$	$6{,}22.10^{12}$
Fast	$2{,}56.10^{12}$	$2{,}64.10^{12}$

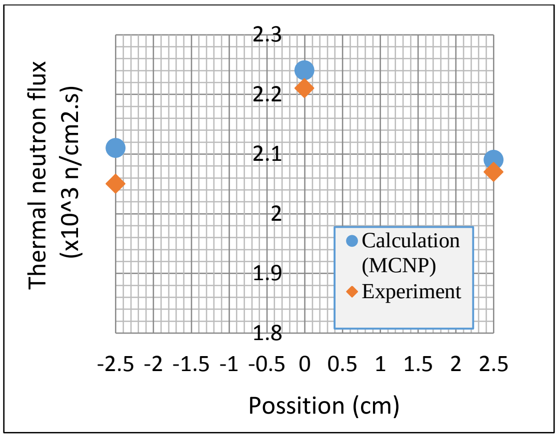

Figure 3. Thermal neutron distributions in radial direction at the neutron trap (zero position is the neutron trap center)

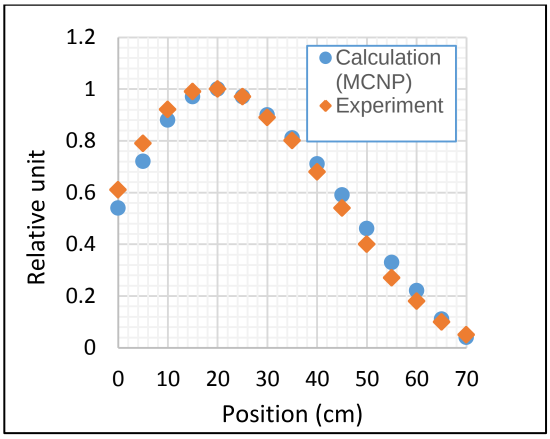

Figure 4. Thermal neutron distributions in axial direction at the neutron trap

The experiment and calculation results in **Figures 3** show that the deviation of thermal neutron flux distribution in the radial direction at the neutron trap ranges from 5-7%. The experiment result is consistent with the calculation result for thermal neutron flux distribution in the axial direction as shown in **Figures 4**. The maximum thermal neutron flux value is located at 20 cm away from the core bottom. The shift of maximum flux to the bottom of the neutron trap is mainly affected by control rod positions. With a large discrepancy of the thermal neutron distribution in both radial and axial directions, the current neutron trap needs to be redesigned to meet the requirements of NTD-Si test.

2.3. Design and testing of irradiation rig using flux screen.

The flux screen method has been selected for the purpose of NDT-Si testing in DNRR because of its consistent with characteristics design of DNRR core configuration. The principle of flux screen method to flatten the neutron flux distribution is shown in **Figure 5**. A uniformity of the irradiation flux is achieved by using screens from different materials to absorb neutron or change the appropriate thickness of neutron absorbers. The screens are made of strong neutron absorbers in high neutron flux region (stainless steel) and weak absorbers in low flux region (aluminum).

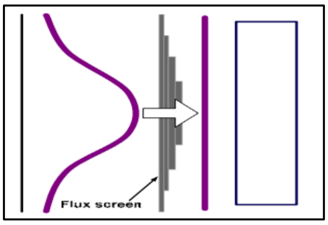

Figure 5. Flux screen method [1].

Dimensions of the irradiation rig and screen materials used to flatten flux distribution has been calculated and modified repeatedly by comparing calculation results using MCNP computer code and experiment results. **Figure 6** shows details of the design and materials of irradiation rig.

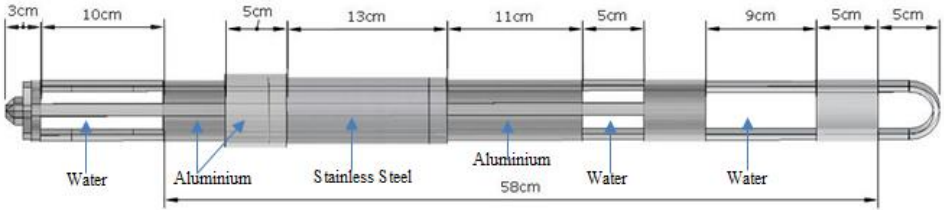

Figure 6. The design of the irradiation rig for NTD-Si testing using flux screen method

Figure 7. Silicon ingot (left hand side) and aluminum ingot (right hand side)

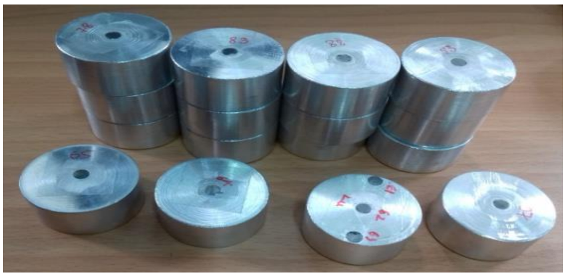

Figure 8. Aluminum ingots with Lu foils

The important parameters related to nuclear safety and radiation safety issues such as reactivity and radiation dose rate have been calculated before installing the irradiation rig in the neutron trap [7], [8]. Then, the experiment using aluminum ingots were performed. The aluminum ingots have similar neutronic characteristics and dimension of testing silicon ingots were inserted in the rig. Each ingot has a diameter of 4 cm and 2 cm in length. Lu-176 foils were attached on aluminum ingots and irradiated for the determination of thermal neutron flux distribution in the rig as shown in **Figure 7** and **Figure 8**.

3. CALCULATION AND EXPERIMENT RESULTS ON THE IRRADIATION RIG

Figure 9 shows calculation results of neutron spectra when using water, aluminum or silicon in the irradiation rig. The fast neutron flux is higher than that in case of replacing of water volume in the rig by aluminum or silicon ingots.

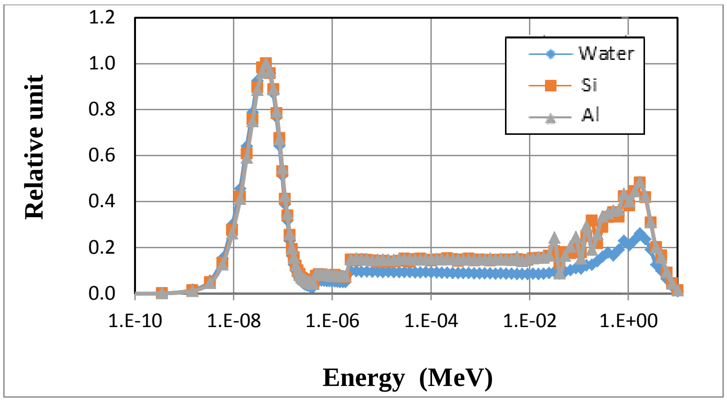

Figure 9. Calculation of neutron spectra with water, aluminum or silicon in the irradiation rig

Experimental results and calculations presented in **Table 3** show that the deviation of thermal flux in axial direction at the irradiation region is within 5%. The distribution of thermal neutron flux in experiment is consistent with the calculation results. The thermal neutron flux peaks at 10 cm away from the bottom of the irradiation rig in the experiment and about 12 cm in the calculation. The flux distribution in the top half from the 30-34 cm of the irradiation rig tends to increase but still below the desired value of 5% in both calculation and experiment. These results show that the irradiation region of 10 cm to 36 cm in axial direction of the irradiation rig is suitable for NTD-Si test.

Table 3. The relative distribution of thermal neutron flux in axial direction at the irradiation rig using flux screen method

Position (cm)	**Experiment** (error ±5-6%)	**Calculation**
10	1.000	0.991
12	0.992	1.000
14	0.999	0.995
16	0.991	0.987
18	0.960	0.972
20	0.980	0.974
22	0.967	0.969
24	0.968	0.955
26	0.979	0.962
28	0.961	0.951
30	0.958	0.953
32	0.955	0.968
34	0.960	0.971
36	0.948	0.954

Table 4 and **Table 5** present the deviation of thermal neutron flux in radial direction at the irradiation rig in calculation and experiment. The deviation is about 2% in experiment and

1% in the calculation for each position in the rig. The distribution of thermal neutron flux in radial direction in experiment is consistent with calculation result. The maximum thermal neutron flux value drops to 1.31×10^{13} n/cm^2.s in the irradiation rig due to combination of the absorption effects from the screen layers and the replacement of moderator by aluminum ingots in the irradiation rig. These results confirm that the irradiation rig using flux screen is capable of NTD-Si testing with the neutron flux uniformity in both radial and axial direction are less than 5%.

Table 4. Calculation result of thermal neutron flux distribution in radial direction at the irradiation rig using flux screen

Flux $\times10^{13}$ n/cm^2.s			
Position (cm)	Left side (-1.9 cm)	Center (0 cm)	Right side (+1.9 cm)
10	1.30	1.30	1.31
20	1.26	1.28	1.27
30	1.25	1.26	1.25

Table 5. Experiment result of thermal neutron flux distribution in radial direction at the irradiation rig using flux screen

Relative unit			
Position (cm)	Left side (-1.9 cm)	Center (0 cm)	Right side (+1.9 cm)
10	0.98	1.00	0.99
20	0.96	0.98	0.97
30	0.95	0.96	0.96

4. CONCLUSION

The irradiation rig using various screen materials has been designed and installed for NTD-Si testing at DNRR. The test with aluminum ingots, which have similar characteristics with silicon ingots, was carried out to confirm the appropriate design of the irradiation rig. The calculated and experimental results show that the uniformity of thermal neutron flux in the irradiation rig is less than 5% in height and 2% in radius, respectively. These values are satisfied the requirements for testing NTD-Si on DNRR. The processes of designing, installing and testing of the irradiation rig would provide good experiences in the application of NTD-Si on the new research reactor which has planned to be built in Vietnam.

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THIẾT KẾ CỐC CHIẾU XẠ SỬ DỤNG PHƯƠNG PHÁP MÀN CHẮN THỬ NGHIỆM PHA TẠP ĐƠN TINH THỂ SILIC TRÊN LÒ PHẢN ỨNG HẠT NHÂN ĐÀ LẠT

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Tóm tắt: Kỹ thuật chiếu xạ pha tạp đơn tinh thể Silic bằng neutron nhiệt (NTD-Si) trên các lò phản ứng nghiên cứu đã được triển khai có hiệu quả tại nhiều nước nhằm tạo ra các thiết bị bán dẫn có chất lượng cao. Tại Việt Nam, kỹ thuật NTD-Si đã được thử nghiệm trên lò phản ứng hạt nhân Đà Lạt (LPƯ Đà Lạt) vào cuối những năm 1980 nhưng các kết quả thu được còn rất hạn chế. Do vậy, việc thiết kế và thử nghiệm cốc chiếu xạ phục vụ NTD-Si trên LPƯ Đà Lạt là cần thiết nhằm có được những hiểu biết đầy đủ hơn để có thể áp dụng kỹ thuật NTD-Si trên lò phản ứng nghiên cứu mới thuộc Trung tâm Nghiên cứu khoa học công nghệ hạt nhân (RCNEST) đã được lên kế hoạch xây dựng tại Việt Nam. Báo cáo này trình bày kết quả thiết kế và thử nghiệm cốc chiếu sử dụng phương pháp màn chắn để thử nghiệm NTD-Si trên LPƯ Đà Lạt. Các thông số quan trọng như phổ và phân bố thông lượng neutron nhiệt tại cốc chiếu xạ đã được xác định bằng thực nghiệm và tính toán sử dụng chương trình MCNP5. Việc chiếu xạ thử nghiệm trên các thỏi nhôm là vật liệu thay thế có tính chất về neutron tương đương Silic cũng được tiến hành nhằm khẳng định thiết kế phù hợp của cốc chiếu. Độ bất đồng đều về thông lượng neutron nhiệt nhỏ hơn 5% theo chiều cao và 2% theo bán kính trong cốc chiếu. Các kết quả thu được cho thấy cốc chiếu mới được thiết kế đảm bảo các yêu cầu thử nghiệm NTD-Si trên LPƯ Đà Lạt

Từ khóa: *Chiếu xạ pha tạp đơn tinh thể Silic bằng neutron nhiệt (NTD-Si), MCNP5, LPƯ Đà Lạt (DNRR), phương pháp màn chắn, phương pháp kích hoạt lá dò, cốc chiếu xạ.*