

LARGE OBJECT IRRADIATION FACILITY IN THE TANGENTIAL CHANNEL OF THE JSI TRIGA REACTOR

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ABSTRACT

This paper presents the design and installation of a new irradiation device in the Tangential Channel of the JSI TRIGA reactor in Ljubljana, Slovenia. The purpose of the device is to enable on-line irradiation testing of electronic components considerably larger in size (of lateral dimensions of at least 12 cm) than currently possible in the irradiation channels located in the reactor core, in a relatively high neutron flux (exceeding 10^{12} n cm⁻² s⁻¹) and to provide adequate neutron and gamma radiation shielding.

1. Introduction

The JSI TRIGA reactor in Ljubljana, Slovenia is a 250 kW, light water, pool type research reactor, extensively used for education and training [1], irradiation of samples for neutron activation analysis (biological and geological samples) [2-4], irradiation of materials for fusion reactors, experimental testing of nuclear instrumentation sensors (fission and ionization chambers and self-powered neutron detectors - SPNDs) in collaboration with the Instrumentation, Sensor and Dosimetry Laboratory of CEA Cadarache [5-8], testing and validating nuclear data, computer code benchmarking [9-13], research and development of radiation resistant electronic components, etc. The reactor is a reference centre for neutron irradiation testing of semiconductor radiation detectors used in the large experimental facilities at CERN [14, 15].

In the framework of the AIDA 2020 collaboration project, a large object irradiation facility has been designed, manufactured and installed in the Tangential Channel of the JSI TRIGA reactor. The irradiation locations used previously are located in the reactor core. Several vertical irradiation channels occupying one standard TRIGA fuel element position are available, with an internal diameter of 31.6 mm, one particular irradiation location - the Triangular Irradiation Channel - occupies three fuel element positions and allows irradiations of objects of lateral dimensions of around 60 mm. The objective of the new facility installed in the Tangential Channel – a horizontal channel piercing the concrete biological shield, the reactor tank and the graphite reflector around the core - is to allow irradiations of objects of lateral dimensions of at least up to 12 cm in a relatively intense neutron flux, of the order of 10^{12} n cm⁻² s⁻¹.

The requirements for the irradiation device were to allow for easy insertion and withdrawal of samples, the capability of on-line irradiation testing of electronic components, i.e. the provision of cable and coolant line feedthrough capability, and to ensure adequate shielding from neutron and gamma radiation originating from the reactor core. Additionally, no modifications to the Tangential Channel were admissible. In fact, a key role of the irradiation device was to ensure protection of the interior channel components, especially the water-tight stainless steel manifolds which join the sections of the channel located inside the reactor tank.

The first part of the paper presents the irradiation device design stage, on the basis of Monte Carlo calculations with the MCNP particle transport code [16], deterministic shielding calculations with the MicroShield code [17] and experimental dose-rate measurements performed with a mock-up of the shield configuration. The second part presents the first measurements and calculations,

performed to characterize the neutron flux and the gamma field intensity inside the irradiation device.

2. Irradiation device design

The irradiation device was designed on the basis of the pre-established performance requirements, i.e. to allow:

- irradiations of objects of lateral dimensions of at least 12 cm,
- on-line irradiation testing (i.e. provision of cable feedthrough capability),
- irradiations in a high neutron flux, at least $10^{12} \text{ n cm}^{-2}\text{s}^{-1}$.

Since the dimensions of the vertical irradiation channels present in the reactor core are not sufficiently large and there are no immediate possibilities of a core modification, the larger horizontal channels the reactor is equipped with were taken into consideration. Figure 1 schematically displays the horizontal irradiation channels in relation to the reactor core and biological shield.

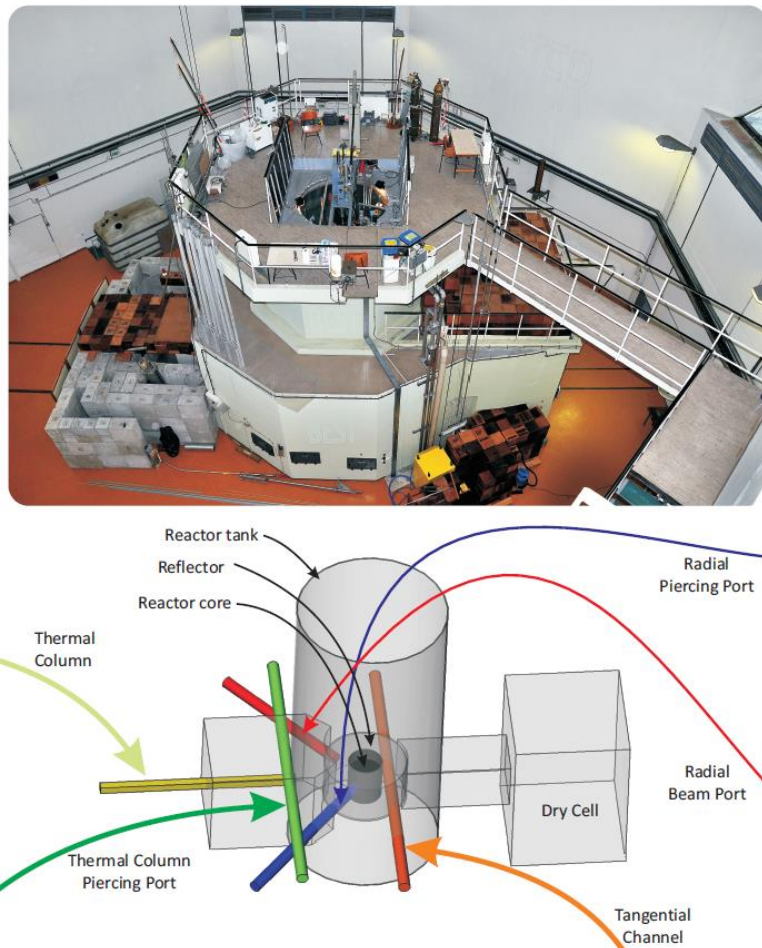


Figure 1: Top: photograph of the JSI TRIGA reactor, bottom: schematic drawing of the main reactor components and horizontal irradiation channels.

Typically, the horizontal channels are used as neutron beam ports, to convey neutrons from the reactor core through the reactor biological shield for the needs of particular experiments, e.g. neutron imaging, neutron diffractometry, neutron activation analysis, etc. The inner sections of the

horizontal channels closest to the reactor core (with the highest neutron flux), i.e. the Radial Beam Port (RBP), the Radial Piercing Port (RPP) and the Tangential Channel (TangCh) are 15.4 cm in diameter.

The Tangential Channel was chosen on account of the neutron flux profile having a characteristic bell shape, enabling more uniform irradiations of large samples than the RPP channel, where the neutron flux profile is sloped. Figure 2 displays the thermal, epithermal and fast neutron flux profiles and 640-group neutron spectra in the Tangential Channel, obtained from Monte Carlo calculations with the MCNP code, normalized to full reactor power (250 kW). The calculations were part of a characterization of the neutron flux and spectra in the ex-core irradiation facilities in the JSI TRIGA reactor, supported by experimental activation rate measurements [18, 19].

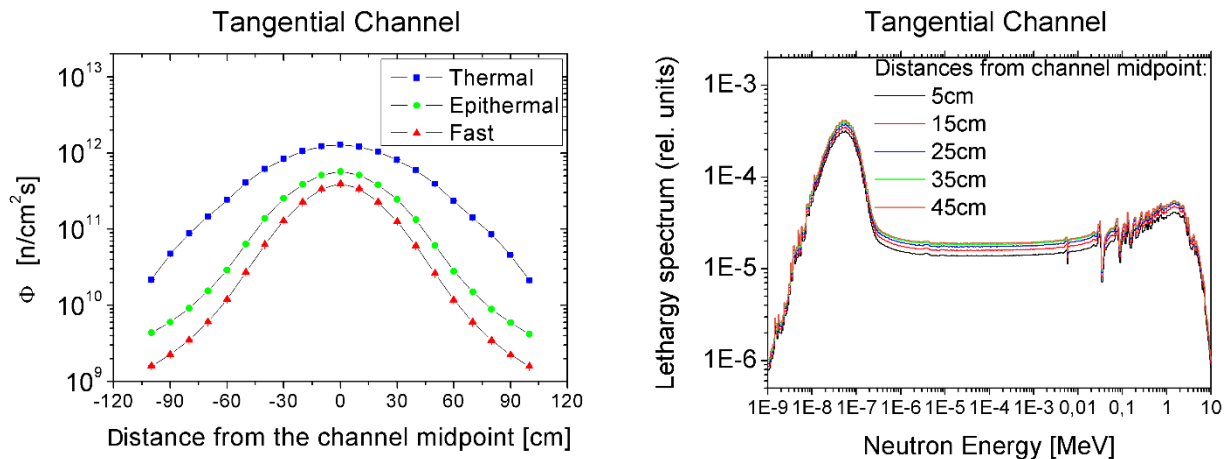


Figure 2: Left: calculated neutron flux profiles (thermal, epithermal and fast), right: neutron spectra in 10 cm sections of the Tangential Channel, both normalized to full reactor power (250 kW).

The first task we performed was to measure the dimensions of the interior components of the channel. The internal dimensions (lengths and diameters) were determined using specially made polyethylene calipers to transfer the internal diameters of the channel and a laser distance meter. Following the measurements, a conceptual design of the irradiation device was formed, comprising of an aluminium liner to protect the channel interior and enable easy insertion and withdrawal of samples and a combined neutron and gamma shield mounted onto a trolley on rails, located next to the channel port in the biological shield. The envisaged configuration of the combined neutron and gamma shield was comprised of an interior cylindrical section made from borated polyethylene and an outer lead block. A curved passage in the borated polyethylene and the lead shielding would allow for the connexion of power and data cables or coolant lines to the sample under irradiation. Figure 4 displays the conceptual design of the irradiation device in the »open« position. During reactor operation the shield has to be in the »closed« position. For sample irradiations, a polyethylene sample holder would be attached to the end of the interior cylindrical section; by moving the shield into the »closed« position, the sample would be moved to the reactor core centreline. Figure 4 displays the conceptual design of the irradiation device.

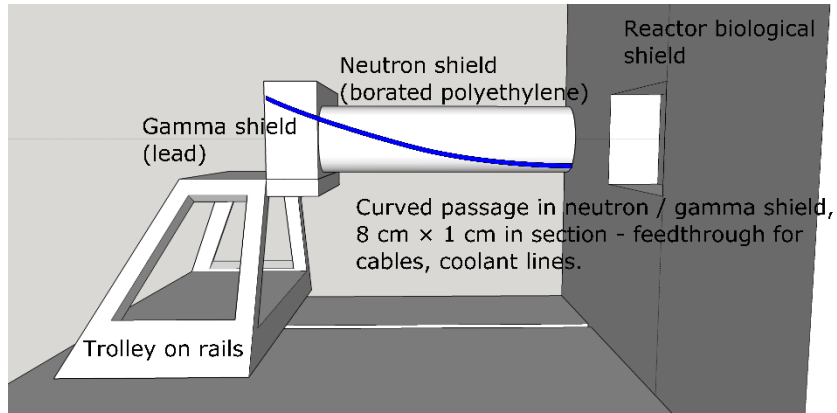


Figure 4: Conceptual design of the irradiation device.

3. Dose rate measurements and device dimensioning

The feasibility and adequacy of the conceptual design was determined on the basis of experimental dose rate measurements with a mock-up of the proposed shield configuration. In principle it would be possible to perform the determination and optimization of the shield dimensions by Monte Carlo calculations, however the experimental approach was favoured since firstly, the distance from the shield and the reactor core is considerable, which implies that a prohibitively large number of particle histories would have to be specified to obtain results of satisfactory statistical uncertainty, or that a complex variance reduction technique would have to be employed and secondly, it is difficult to realistically model clearances / gaps between the shield components which are significant contributors to both the neutron and gamma dose rates in the vicinity of the proposed shield.

The shield mock-up was constructed using a cylindrical borated paraffin insert and lead blocks, available on site. The (natural) boron content in the paraffin insert is approximately 5%; the insert is encased in aluminium, overall 20 cm in diameter and 75 cm in length. This insert was used as a neutron shield in past experimental activities at the reactor. The lead blocks measure around 10 cm × 4 cm × 21 cm and are used to construct temporary gamma shielding according to specific needs. They were arranged in the recess in the concrete biological shield at the outer end of the Tangential Channel. Additionally, the channel hatch provided an extra 2.5 cm of lead shielding, therefore the total lead shielding thickness was 12.5 cm.

Ambient dose equivalent measurements were performed using a neutron dose rate meter (Berthold, model LB6411 ³He proportional counter) and two gamma dose rate meters (Automess, model 6150 AD6/H GM counter and Victoreen, model 451-DE-SI ionization chamber) by the JSI Radiation Protection Unit at low reactor power - 250 W and 2.5 kW. These instruments are routinely calibrated. The combined relative standard uncertainty of dose rate measurements with portable instruments has been estimated to 20% (k=1). Table 1 reports the measured dose rate values at low reactor power and the dose rate values scaled to full reactor power (250 kW). Based on the experimental dose rate values, the expected values of the neutron and gamma dose rates at full reactor power were several 10 μSv/h and over 1 mSv/h, respectively. The neutron shielding capability of the borated paraffin insert was deemed acceptable, the gamma dose rate, however, was deemed excessive. Therefore, in order to reduce the gamma dose rate to a level comparable to the neutron dose rate (10 μSv/h), the required thickness of the lead shielding was estimated determined using the literature data [20] and by calculations using the MicroShield code. The determined total lead thickness was 30 cm.

Table 1: Measured neutron and gamma dose rates at low reactor power using a neutron and gamma shield mock-up (borated paraffin cylinder, L=75 cm, 2R=20 cm and lead blocks, effective shielding thickness 12.5 cm).

Power [kW]	Measured			Scaled to full reactor power		
	n Berthold LB110 [$\mu\text{Sv/h}$]	γ Automess [$\mu\text{Sv/h}$]	γ Victoreen [$\mu\text{Sv/h}$]	n Berthold LB110 [$\mu\text{Sv/h}$]	γ Automess [mSv/h]	γ Victoreen [mSv/h]
0 (Background)	0	< 0.1	< 0.1	/	/	/
250 W	0.01	1.4	1	10	1.4	1
2.5 kW	0.4	13	8	40	1.3	0.8

The material chosen for the neutron shield was borated polyethylene, on account of its good machinability and the fact that no additional encapsulation would be required. In order to choose the most appropriate material between the available options on the market (normal and high-density borated polyethylene, of densities 0.95 and 1.07 g cm⁻³, respectively), and to optimize the dimensions of the curved passage through the neutron shield, several Monte Carlo simulations with the MCNP6 code in conjunction with the ENDF/B-VII.1 library were performed in which the borated polyethylene insert, including the curved passage for cable feedthrough was modelled. The source neutron spectrum for the Tangential Channel was taken from previous Monte Carlo simulations in which we determined the neutron spectra in several irradiation locations in the JSI TRIGA reactor [18]. A track length estimator tally of the neutron flux (F4 tally in MCNP terminology) was employed, multiplied by the fluence-to-dose conversion function (in pSv cm²) taken from the ICRP-21 publication [21] to calculate the total neutron dose rate outside the polyethylene insert. Figure 5 displays the computational models of the neutron shield with no passage and with varying dimensions of the passage (10 cm in width, 1 cm and 2 cm in height).

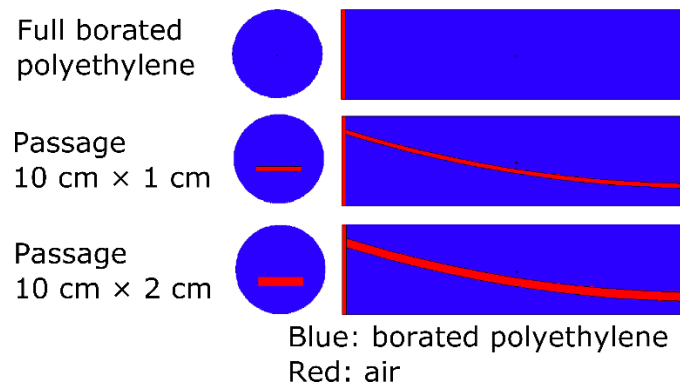


Figure 5: Computational model of the borated polyethylene neutron shield with a varying thickness of the curved passage for cable feedthrough.

An additional simulation was carried out with the same geometry as in the first case (no passage), with air instead of borated polyethylene – this was taken as the reference case. Table 2 presents the simulation results for different apertures in the shield for both normal and high-density borated polyethylene.

Table 2: Calculated neutron dose rates for neutron shields for different borated polyethylene densities (normal-density polyethylene – NDPE: 0.95 g cm⁻³ and high-density borated polyethylene – HDPE: 1.07 g cm⁻³) and for different dimensions of the aperture.

Case	Dose Air [pSv / source n]	Rel. unc. [%]	Dose NDPE [pSv / source n]	Rel. unc. [%]	Dose HDPE [pSv / source n]	Rel. unc. [%]	Dose NDPE vs. Dose Air	Dose HDPE vs. Dose Air
Air	7.8×10 ⁻²	0						
2 cm			7.8×10 ⁻⁷	1.9	4.4×10 ⁻⁷	2.4	1.01×10 ⁻⁵	5.68×10 ⁻⁶
1 cm			2.3×10 ⁻⁷	3.8	7.8×10⁻⁸	6.0	2.97×10 ⁻⁶	1.00×10⁻⁶
Full			1.1×10 ⁻⁷	5.6	3.1×10 ⁻⁸	10.4	1.45×10 ⁻⁶	3.94×10 ⁻⁷

High-density polyethylene material was chosen on account of its superior neutron attenuation performance. The chosen passage dimensions was 1 cm. As indicated in Table 2 in green, the attenuation factor in the dose rate for a high-density polyethylene neutron shield with a 1 cm passage is only around 30% smaller than the attenuation factor of a normal density polyethylene shield with no passage.

4. Device manufacture and installation

After initiating a formal procedure and obtaining the required authorization from the regulatory body, the Slovenian Nuclear Safety Administration, the manufacture of the device was started. Measurements were performed to assess if the Tangential Channel and the floor of the reactor hall onto which the guide rails would be mounted to were both level and parallel. The aluminium liner was manufactured from a length of aluminium tubing, outer diameter 15 cm, inner diameter 14.6 cm. A mounting flange with an interior conical section for easier sample insertion and withdrawal and three friction fasteners was machined out of solid aluminium and welded to the tube. The trolley was manufactured out of stainless steel square tubing, with one chain driven axle for easy manipulation. A steel cage was mounted to the trolley which contains V-grooved lead blocks used for the gamma shielding. The cage also serves as a mounting structure for the borated polyethylene neutron shield. The high-density borated polyethylene was acquired in the form of a 1.22 m × 1.22 m × 10.2 cm plate. The plate was divided into rectangular sections, which were bolted together and turned as an assembly down to the required diameter. The curved passage was machined out of the inner two polyethylene sections, thereby creating a removable curved insert in the cylindrical neutron shield. This greatly facilitates the setup phase in the experimental activities. A passage smoothly following the one in the borated polyethylene was machined in the lead blocks. Figure 6 displays the trolley on rails during the installation phase.

5. Test measurements

5.1 Dose rate measurements

After installation of the trolley, test neutron and gamma dose rate measurements at full reactor power were performed again, to verify the adequacy of the neutron and gamma shield. The measured neutron and gamma dose rates are reported in Table 3

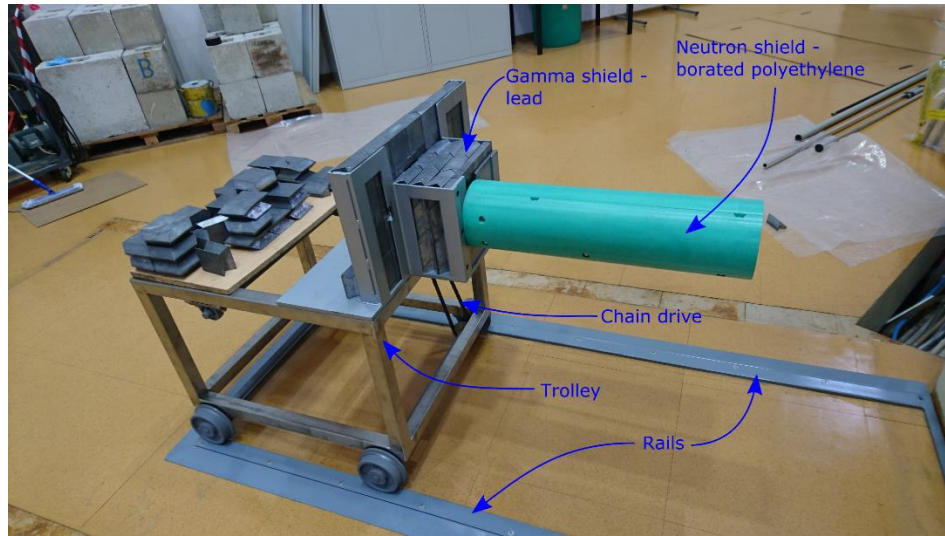


Figure 6 Trolley with neutron and gamma shield during the installation phase.

Table 3: Measured neutron and gamma dose rates in different locations around the shield in the closed position at full reactor power (250 kW).

Location	Measured at full reactor power (250 kW)		
	n Berthold LB110 [$\mu\text{Sv/h}$]	γ Automess [$\mu\text{Sv/h}$]	γ Victoreen [$\mu\text{Sv/h}$]
Channel centre, behind Pb shielding	70	2	2.3
Gap above lead shielding		~ 1000	~ 300
Close to cable passage		~ 4000	~ 700

From the measured dose rates immediately behind the lead shielding it can be concluded that the neutron and gamma shield was dimensioned adequately. The maximum values dose rates were measured in the immediate vicinity of the gaps between the lead shielding and the reactor biological shield and in the cable passage through the lead shielding. On account of the local high dose rates, it was decided to erect a concrete labyrinth made from concrete blocks available on site (30 cm \times 30 cm \times 60 cm) around the channel port. The labyrinth has two functions: firstly, to physically limit access of personnel to the vicinity of the device during reactor operation and secondly, to provide additional gamma shielding against gamma rays emitted by samples irradiated samples in temporary storage close to the irradiation device.

5.2 Gamma field profile and neutron flux measurements

After the construction of the concrete labyrinth, the first experimental measurements in the device were performed. A prototype of a polyethylene sample holder was manufactured, which mounts onto the cylindrical neutron shield. The prototype is currently being modified and refined according to experimental requirements. Firstly the gamma flux profile in the channel was measured using a 3 mm miniature ionization chamber manufactured by the CEA (French Atomic and Alternative Energy Commission) in current mode. The chamber was inserted into an aluminium guide tube, mounted to the polyethylene sample holder. The chamber was moved in steps using a specially developed pneumatic positioning system used for axial fission rate measurements with miniature

fission chambers in the reactor core. The measured gamma flux profile is displayed in Figure 7, along with a schematic drawing of the polyethylene sample holder in relation to the cylindrical neutron shield.

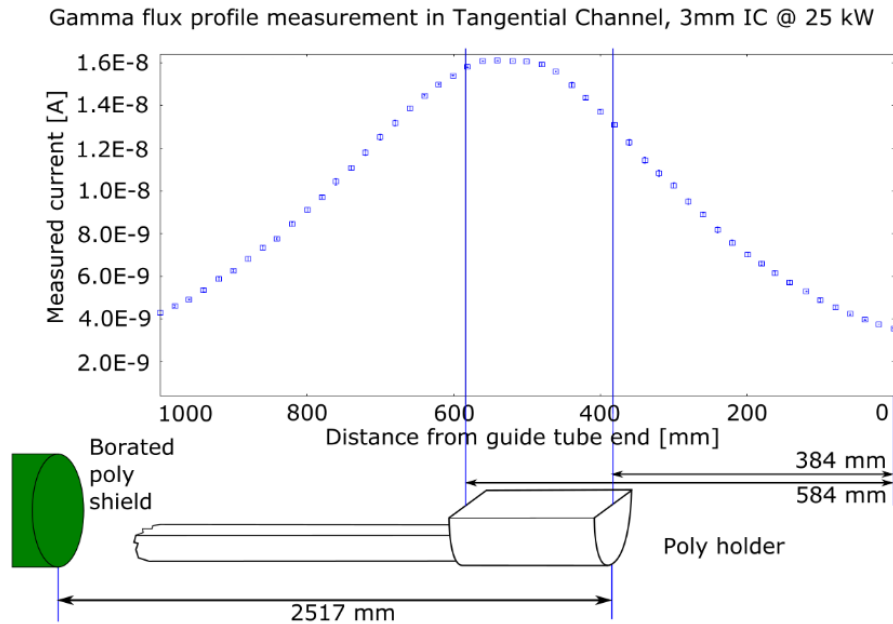


Figure 7: Measured gamma flux profile in the Tangential Channel at 25 kW.

A 130 min test irradiation at 2.5 kW was performed of four foils of Al-0.1%Au (certified reference material IRMM 530R) and two PIN diodes to measure the neutron flux and the hardness factor. The foils and diodes were attached to a poly-methyl methacrylate (Plexiglas) plate, which was positioned vertically within the polyethylene sample holder. One foil and one diode were positioned in the location where the measured gamma flux was the highest, another foil and diode 10 cm deeper in the channel, away from the reactor core, the two remaining foils +/- 5cm in the vertical direction with respect to the first foil and diode. The induced ^{198}Au activity in the Al-0.1%Au foils was measured using a HPGe detector. The specific saturation activities, equal to the reaction rates, were determined from the measured activities and timing information. The total neutron flux was obtained by scaling the neutron spectrum obtained from Monte Carlo calculations to the measured $^{197}\text{Au}(n,\gamma)$ reaction rates. A photograph of the foils and diodes as irradiated, along with the reported experimental results for the total neutron flux is displayed in Figure 7.

From the results of the activation measurements we can see that the total neutron flux in the peak position, scaled to full reactor power is $2.67 \times 10^{12} (1 \pm 0.027) \text{ n cm}^{-2} \text{ s}^{-1}$, the variation in the neutron flux over +10 cm in the longitudinal direction is around -11% and the variation over +/- 5 cm in the vertical direction is -4% at most.

The 1 MeV NIEL (Non Ionizing Energy Loss) equivalent fluxes were determined with measurements of the leakage current in fully depleted and irradiated PIN diodes [22]. The current was measured after 48 h of annealing at 60°C. The NIEL equivalent flux scaled to full reactor power was found to be $(3.9 \pm 0.4) \times 10^{11} \text{ n cm}^{-2} \text{ s}^{-1}$. The measured hardness factor [23] for the total neutron flux is 0.146, the hardness factor for the fast neutron flux is 0.83. For comparison, the results for the hardness factors in the reactor core are 0.225 for the total and 0.9 for the fast neutron flux [22].

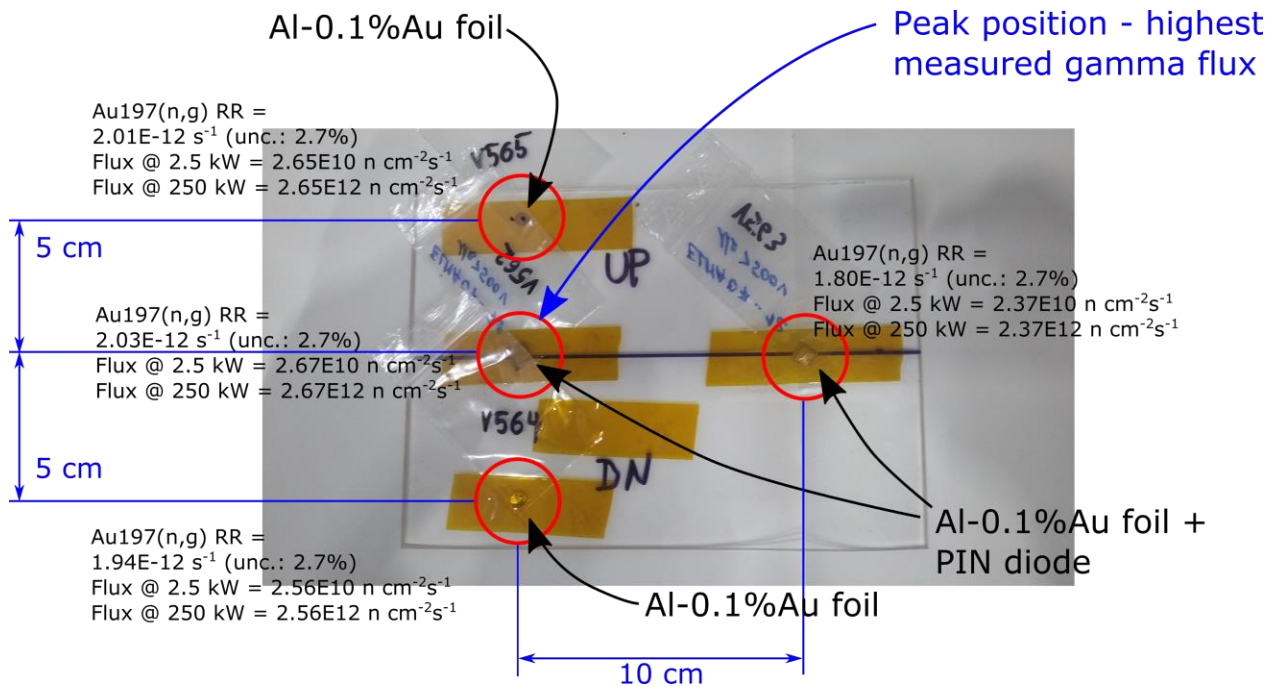


Figure 7: First neutron flux measurements: 4 × Al-0.1%Au foils and 2 × PIN diodes attached to a Plexiglas plate, irradiated at 2.5 kW and experimental results (“RR” denotes the measured specific saturation activities, equal to the reaction rates at 2.5 kW).

6. Conclusions and future work

This paper presents the design and installation of a large object irradiation device in the JSI TRIGA reactor and the first neutron flux measurements and gamma field profile within. The approach used to determine the dimensions of the neutron and gamma shielding components of the device through test dose rate measurements with a mock-up of the shielding configuration proved effective. After carrying out the planned modifications to the polyethylene sample holder an experimental campaign is planned in which a 3 mm U-235 fission chamber will be employed and activation measurements for different nuclear reactions will be performed. This information will allow us to completely characterize the irradiation device in terms of the neutron flux profile and the neutron spectrum in the channel. The new device is available for irradiation testing in the framework of the AIDA 2020 Transnational Access programme (<http://aida2020.web.cern.ch/content/how-apply-transnational-access>).

7. Acknowledgement

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8. References

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