Radiation shielding design of neutron source from Kartini reactor’s beam-port for SAMOP test facility

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Abstract. SAMOP is a subcritical assembly for 99Mo production fueled with uranyl nitrate solution which is designed to be operated by using external neutron source from a neutron generator. The most common radioisotope used in diagnosis for medical imaging in nuclear medicine is 99mTc which is daughter isotope of 99Mo. The SAMOP experimental or test facility is a facility to be used for performance testing of the SAMOP system by using neutron source from radial beam port of Kartini reactor instead of using neutron generator. This paper presents the results of the radiation shielding design and calculation for external neutron source of SAMOP experimental facility. The method used is a modeling and calculation by using MCNPX computer code. Based on the assumed source terms, the materials being used, and the geometrical arrangements, it is concluded that by using paraffin of 60 cm thickness for the beam catcher and 50 cm for the concrete of the outer shield would be sufficient to reduce the radiation dose below the maximum recommended limit. The presence of beam catcher can significantly reduce the contribution of neutrons and secondary particles to the radiation dose.

Keywords: Radioisotope, 99Mo, nuclear medicine, radiation shielding, SAMOP.

1. Introduction
A non-critical reactor system for 99Mo production has been designed and developed at the Centre for Accelerator Science and Technology (CAST), National Nuclear Energy Agency (BATAN). The system is called subcritical assembly for 99Mo production (SAMOP) which is fueled with uranyl nitrate and designed to be operated by using external neutron source from a neutron generator [1]. Uranyl nitrate solutions has been choose as fuel and target because of easier to prepare than uranyl sulfate solutions [2,3,4]. Neutron source from radial beam port of Kartini reactor will be used as a preliminary step to investigate and to prove the design results.

Kartini reactor is a pool type TRIGA mark II reactor, operated with nominal power of 100 kW, and has been operated since 1979 with the main function for education and research services. It has several irradiation facilities, including 4 beam ports that can be used for many different applications. A proposed SAMOP experimental facility will utilizing one of the beam ports which are still not utilized optimally, i.e. radial beam port. The characterization of the existing beam port to select one which is considered to be the most suitable have been done [5,6]. Based on the result obtained, it was then decided to use the radial beam port of Kartini reactor for this purpose. The next step of work was to model the collimator at the radial beam port intended to get mostly thermal neutron beam that will be used as neutron source of SAMOP.

This paper present the work concerning with the design of shielding system at the outer area in front of the beam port that will be used for SAMOP experiment. This includes the design neutron beam catcher and the outer radiation shield. The beam catcher is intended to capture the neutron beam steaming out from the collimator exit and passing through the SAMOP experimental tank, whilst the
outer shield is intended to shield mostly the gamma rays originating both from the beam port exit and those produced and escape from the SAMOP coolant tank.

The effective thickness of the material needed for both the beam catcher and the outer shield would depend on several factors such as the source strength of the beam source, the kinds of material to be used, and the distance to the source. In evaluating the fluxes and the corresponding dose rates as a function of material’s thickness used in the model of beam catcher and outer shield, the MCNPX code was used [7,8].

2. Layout of SAMOP test facility
Figure 1 depicts the layout of Kartini reactor and location of SAMOP test facility, using radial beam-port as an external neutron source to drive SAMOP. Figure 2 describes the shielding system for the SAMOP test facility.

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**Figure 1.** Layout of Kartini reactor and SAMOP experimental facility

**Figure 2.** Collimator, sample position and shielding system
The main components of the shielding system i.e. the neutron beam catcher and the outer shield. The main concern in this calculation is to determine the effective thickness of materials to be used both in the beam catcher \((b)\) and the outer shield \((d)\), for a defined distance \((a)\) and \((c)\) which in this study was taken for 50 cm and 70 cm respectively. The distance from the source to the beam catcher’s gate is 30 cm. The diameter of the beam catcher’s gate \((e)\) in this simulation was taken of 30 cm.

3. Material and methods

3.1 Materials

Paraffin wax, composed of carbon and hydrogen \((C_n H_{2n+2})\), with the typical formula of \((C_{31} H_{64})\) is a good neutron moderator that can be effectively used to slowdown the fast neutrons \([9]\). It is easy to manufacture and inexpensive and therefore it was used as the main component of beam catcher in this design study. The outer shield was to use ordinary concrete which is commonly used for radiation shielding especially for gamma rays \([8,10,11]\). Table 1 presents the data of chemical compositions and the corresponding atom density of ordinary concrete and air that was used in this calculation \([12]\).

| Chemical composition and atom density for concrete and air \([12]\) |
|-------------------------|-----------------------------|
| Ordinary Concrete – KENO standard mix (density = 2.3 g/cm³) |                       |
| Chemical element         | Atom density (atoms/barn.cm) |
|-------------------------|-----------------------------|
| H                       | 1.374×10⁻²                  |
| O                       | 4.606×10⁻²                  |
| Si                      | 1.662×10⁻²                  |
| Al                      | 1.75×10⁻³                   |
| Na                      | 1.75×10⁻³                   |
| Ca                      | 1.52×10⁻³                   |
| Fe                      | 3.5×10⁻⁴                    |
| Air                     |                             |
| O                       | 1.0868×10⁻⁵                 |
| N                       | 4.3479×10⁻⁵                 |

3.2 Methods (defining source terms)

If the beam is well collimated, it is expected to contain mostly neutrons with a specific characteristic, which will be more easily to handle. However it could be different in reality, due to both the engineering and the computational factors. Therefore for the purpose of improving the safety margin, the source terms for this design was to use the beam characteristics of radial beam port’s exit for the condition without the presence of collimator when the reactor was operated at 100 kW. Thus, it contains both neutrons and photons for the whole energy ranges, as obtained in the previous calculation \([5,6]\). The correction factor of 10% of was then added to the calculated values to include the reactor power calibration factor. Table 2 presents the characteristics of the beam source which was used as the source terms in this calculation. Further, the beam was assumed to have mono direction.

| Range of Energy | Neutron flux \((n \text{ cm}^{-2} \text{ s}^{-1})\) | Photon flux \((n \text{ cm}^{-2} \text{ s}^{-1})\) |
|-----------------|--------------------------------|------------------|
| \(E \leq 0.5 \text{ eV}\) | 1.6801×10⁸                     | -                |
| \(0.5 \text{ eV} < E < 0.1 \text{ MeV}\) | 1.4875×10⁸                     | 2.41×10⁷         |
| \(E > 0.1 \text{ MeV}\) | 5.4437×10⁷                     | 1.89×10⁸         |
MCNPX was used to evaluate the fluxes and the corresponding doses as a function of thickness of the material being investigated. MCNPX is a general purpose particle transport Monte Carlo code developed by the Los Alamos National Laboratory designed to track many particle types over broad ranges of energies [7]. In this case, point detector tally (F5) has been used at several points of interest along the beam axis, and some variance reduction techniques were applied in this simulation [13]. The standard dose function as prescribed by NCRP-38 1971; ANSI/ANS-6.1.1-1977 [14] was used for converting the flux to radiation dose rate.

4. Results and Discussion

Figure 3 is the MCNP geometrical representation of the beam catcher model, showing vertical and horizontal cut view. In this case, the distance from the source to the beam catcher’s gate and to the paraffin block was 30 cm and 70 cm respectively. The paraffin block was divided into several identical segments, each of 5 cm thickness and aluminum (Al) of 0.5 cm thickness was used as casing for the paraffin.

As the beam entering the beam catcher’s gate it will undergo collision process within the paraffin block. Figure 4 shows the collision density distributions of the neutrons in the beam catcher indicating that most collision reactions occurred in the center area of the paraffin block. At the depth of around 60 cm the neutron intensity has been significantly reduced.

Figure 5 presents the results of the neutron flux calculation at several points along the beam axis in the beam catcher with relative error of 4 %. The result shows the changes of the neutron characteristics with the increase of the penetration in the paraffin. At 70 cm of depth the intensity has been significantly reduced.
Figure 5. Neutron characteristics as a function of paraffin thickness

Figure 6 presents the corresponding dose rates as a function of paraffin thickness. The result shows that at 60 cm of thickness the dose rate has significantly reduced to be 12.4 $\mu$Sv/h, below the recommended maximum limit of 25 $\mu$Sv/h [15], and at 80 cm of thickness it becomes around 0.5 $\mu$ Sv/h. While, the dose limit according to National Regulatory Body regulation is 2.3 $\mu$Sv/h [16]. This result is in accordance with the similar shielding design works by using concrete material instead of paraffin, where the shielding capability and requirements for neutron attenuation at the Neutron Radiography Facility in South Africa (SANRAD) concrete shielding thickness is varied from 60, 80 to 100 cm [8]. In the SANDRAD shielding a 60 cm concrete thickness, a 99.88% and 98.5% drop of neutrons and gamma-rays counts respectively, is possible.

Figure 6. Neutron dose rates as a function of the paraffin thickness (cm)
Figure 7 is the MCNP geometrical model of the shielding system, showing the beam source, beam catcher and the outer shield.

![MCNP Geometrical Model](image)

**Figure 7.** MCNP geometrical model of the shielding system.

The outer shield was divided into several layers, each of 10 cm thickness. In this simulation the distance between the outer shield and the beam source was set for 200 cm and the gap between beam catcher and the outer shield was 50 cm. The simulation had been conducted for two cases i.e. the radiation dose for the case with and without the presence of beam catcher.

Figure 8 presents the results of simulation, for the conditions without and with the presence of beam catcher. In case of beam catcher was removed, both neutrons and gamma rays from the beam port will be slightly reduced during the passage to the outer shield. More gamma rays will be produced in the concrete as the result of neutron interaction with the concrete material. The result indicates that the neutron and the gamma radiation at the outer surface (60 cm of thickness) is still very significant with dose rate of around 0.42 and 0.11 mSv/h respectively or with the total 0.53 mSv/h. This value is still far high beyond the maximum recommended limit of 25 µSv/h [15,16].

If the beam catcher was retained in the place, most of neutrons will be well absorbed in the beam catcher, and the gamma rays originating from the beam port and those produced as secondary particle will also undergo attenuation in it. As the result, the radiation dose at the outer shield of concrete with 60 cm thickness had been significantly reduced, to be around 7.9 µSv/h, where the neutron contribution becomes extremely low, and was well negligible. Whiles, the outer shield of concrete with 50 cm of thickness, the radiation dose is around 11.9 µSv/h, still below the maximum recommended limit. Therefore, the outer shield is fixed using concrete with 50 cm of thickness.

The shielding system has been installed and the preliminary measurement at the above condition shows a total dose rate of 4.3 µSv/h of gamma and neutron radiations. Further activities are conducting neutron flux measurement and preliminary experiments to operate SAMOP experimental test facility using this neutron source.
5. Conclusion.
The design of shielding system for the proposed SAMOP experimental facility at Kartini reactor had been conducted using MCNPX computer code. The calculation included the design of neutron beam catcher and the outer shield. Based on the assumed source terms, and the geometrical arrangements, a minimum paraffin thickness of 60 cm of beam catcher will be sufficient to reduce the neutron’s radiation dose below the maximum recommended limit. For the outer shield, using ordinary concrete, with the beam catcher in place, the radiation dose rate at the outer surface of 50 cm thickness is around 11.9 $\mu$Sv/h still below the recommended limit of 25 $\mu$Sv/h. The presence of beam catcher can significantly reduce the contribution of neutrons and secondary particles to the radiation dose, and as the result the dose rates meets the recommended limit.

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