Assessment of Gamma and Neutron Dose of Optimizing the Radiation Protection in TRIGA 2000 BATAN Bandung.

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Abstract. The TRIGA 2000 reactor is a nuclear installation which is operated at 1000 kW. Gamma and neutron radiation exposure will contribute to occupational doses that will be received by operators and other radiation workers. To maintain the safety of radiation workers, assessment of gamma and neutron radiation is needed in the TRIGA 2000 reactor. survey in the hall reactor using the RTD 30 and RadEye NL detectors. In addition to the direct measurement, calculations are also carried out with simulation data dose using Monte Carlo MCNPX for the calculation and followed by PHITS to get the graphical dose distribution of gamma and neutron. The study will focus on the assessment of the diffusion Gamma and neutron doses by doing the measurement and simulation data collect for the aim of the optimization of the Radiation protection policy in the reactor TRIGA 2000 Bandung. The result of the measurement is shown with graph in different power taken during the measurement and the simulation is the graphical distribution dose of gamma and neutron using PHITS code. A preventive dose reduction technique, worker and publics protection elaborate to optimise the Radiation protection program.

1. Introduction

The use of all technologies involving all sources of ionizing radiation requires a radiological protection program to protect the working people of the place, the public and the surrounding environment. There is an atomic radiation effect that should be controlled to reduce the occupational exposure caused by the Ionizing Radiation. According to a 2008 report by the United Nations Scientific Committee on the Effects of Atomic Radiation, some 23 million workers worldwide are exposed occupationally to ionizing radiation, some 75 per cent of them working in the medical field. [1]

The worker can be exposed by an artificial radiation or naturally occurring radioactive material. That exposure should be reduced and optimized by applying some program to protect them against the Ionizing Radiation. The program established to that issue is namely Radiation Protection. The Ionising Radiation comes from radioactive Particles and Photons which are Alpha, Beta, Neutron, X-Ray and Gamma. Those nature of radiation have the own protection regarding to the parameter that characterize the exposition.

For this study, the case chosen for the purpose is the nuclear facility of TRIGA 2000 Reactor at PSTNT Bandung, which is used to carry out research and radioisotope production by utilizing...
neutrons produced. Currently the reactor is operated with the power of 1000kW or less after being update by the Regulation of the BAPETEN.

To establish the radiation protection program, the fundamental case that should be processed is the calculation or the measurement of the dose rate diffused by the radiation sources from the reactor. A measurement using some equipment for gamma and neutron dose survey will be made and compared to the diffusion dose rate giving by one model calculations that can be used for assessment of the dose is Monte Carlo method with a computer program that is MCNPX (Monte Carlo N-Particle version X) to get the output of the calculation. The software is done by a team of monte Carlo X-5 (2003) of the Los Alamos National Laboratory, USA. Monte Carlo. Monte Carlo method is a numerical statistical method used to solve problems by simulating random numbers for problems that cannot be solved analytically. [2].

The aim of the comparison would be to assess the exposure dose that can affect the worker referred to the limitation dose attributed by the National and International organisation. After using the MCNP for the calculation simulation, another software called PHITS is using to get the graphical dose distribution of the gamma and neutron dose.

2. Measurement Method

2.1. TRIGA Reactor and mapping radiation design

Bandung TRIGA 2000 reactor is a tank-type reactor, entirely installed above ground with installed capacity of 2000 kW but operated at 1000 kW power. TRIGA reactor coolant system in TRIGA 2000 Bandung consisting of primary and secondary cooling systems. In the primary cooling system of heat transfer by convection naturally in the inlet pipe is placed at the bottom of the terrace, while the outlet pipe installed near the surface of the tank. In 2015 PSTNT modified the reactor by replacing two Fuelled Follower units. [3]. Figure 1 shows the horizontal and sections of Reactor TRIGA 2000.

![Figure 1: Horizontal Section of TRIGA 2000 Reactor](image)

Main sources of radiation at reactor TRIGA 2000 at power, includes the core and the coolant where fission products remained in the cladding and activation products in the water coolant. Used ion exchange resin in demineralizer system and used filters of ventilation system were also part of radiation sources in the reactor. At power, gamma rays and neutrons emerge from the radiation
sources mentioned above will contribute to the exposure received by operators and workers which known as occupational dose. The dose rate (µSv/h) were controlled and monitored by conducting radiation survey. The radiation survey of the area in TRIGA 2000 Bandung is managed by doing a routine mapping of the area. There are 32 critical points set in the reactor map which are the point mostly frequented by the worker around the reactor. Figure 2 shows the radiation mapping area and potential high radiation location in the reactor hall.

Figure 2: Radiation Mapping Area and Potential High Radiation Location in the reactor Hall

2.2. Radiation Operational Limit Condition

The Operational Limit Condition for radiation in TRIGA Reactor 2000 is given in the Table 1. The control room is the place which the most of operator in the reactor works when the reactor is operating. Due to nuclear regulation that worker cannot received the radiation dose of more than 20 mSv per year, the dose rate in the control room is set of 10 µSv/h total dose rate for neutrons and gamma rays. While in the top of water tank (1 meter above the water surface) the operational limit condition for radiation was set of 625 µSv/h gamma rays.

| Operation Limit Condition |   |
|---------------------------|---|
| Control room              | 10 µSv/h   |
| At the top of water tank  | 625 µSv/h (Gamma only) |

| WATER TANK |

This limitation is calculated by refereeing from the Dose Limitation value from the IAEA publication [1].
2.3. Measurement Equipment

**Personal Dosimeter:**
The personal dosimeter is usually light and small and can record the accumulated dose of radiation workers. During the collect of the data, the personal dosimeter used are pocket dosimeter (direct reading dosimeter) and the TLD Thermoluminescent dosimeter.

Those dosimeters must be worn at all times when carrying out radiation work and should be used in the position of the middle part of the body. It is recommended to keep away from the area when not carrying out the TLD during radiation work.

**Survey Meter:**
The survey meter equipment is used to know the results of direct measurement of radiation exposure or dose. It must be portable. The Detector for $\alpha$ and $\beta$ use thin filter and for the neutron detector the retaining material paraffin or polyethylene to distinguish fast and slow neutron. The instrument used for this research is the SUPER-IDENT RT-30 Gamma-Ray Spectrometer for the gamma dose measurement and the Thermo Scientific™ Radeye™ NL Personal Highly Sensitive Neutron Radiation Detectors for the directed neutron.

3. Simulation Method

3.1. MCNPX Calculation

The mapping of neutron and gamma dose rate in the TRIGA 2000 reactor can be simulated using Monte Carlo methods. The MCNPX and PHITS are computer code based on Monte Carlo Methods. The MCNPX program was used to determine the flux and neutron-gamma spectrum resulting from fission reactions on the reactor core. The neutron and gamma spectra on the reactor core were used as radiation sources for PHITS inputs. The PHITS code was used to simulate the neutron and gamma dose rate distribution in the reactor. One of the important things to do in modelling the MCNP is the geometry. The accuracy of the modeling results is also determined by the compatibility with the geometry of the object to be modeled. The most difficult step in modeling with MCNP is in making object geometry, especially complex objects. The TRIGA 2000 reactor and reactor generally have a very complex geometry which provides its own difficulties to model [2]. The geometry of the reactor modeled includes the reactor component, reactor building and cooling water.

All parts of the TRIGA 2000 reactor are included in geometric modeling because they all contribute to the absorption of neutron and gamma radiation produced from fission reactions on the reactor core.

To utilize TRIGA-MCNP in modelling radiation sources, modifications are made to the desired output. The modification needed is to add the normalized flux of the source and the energy tally. Tally is the MCNP language to mention the desired physical quantity. The addition of this tally will give an output in the form of an energy spectrum from the radiation source. [2]

To define the radiation source, source strength is needed. This value is then used to normalize the output values of MCNP. If the radiation source is radioactive, the strong value of the source is the activity value in units of Becquerel or Currie. Neutron and gamma radiation which will be calculated is the value of the dose which are neutrons and gamma results of fission reactions that occur in the fuel element. Therefore, the source strength of the radiation source model is the number of neutron and gamma particles per unit time resulting from the fission reaction at the 2 MW reactor power. The strong value of the source is obtained from the calculation of the number of fission reactions needed to produce the power per watt. To get this value, the conversion equation is used as follows: [2]
\[
\left( \frac{1 \text{joule/s}}{\text{watt}} \right) \left( \frac{1 \text{MeV}}{1.602 \times 10^{-13}} \right) \left( \frac{\text{fusion}}{180 \text{MeV}} \right) = 3.47 \times 10^{10} \text{fusion watt}^{-s}
\]

Based on the conversion in Equation (1), it is found that to produce a power of P (watts) and the number of fission reactions needed is \(3.47 \times 10^{10} \times P\). Because in one fission can occur 2.5 neutrons produced so 2 MW reactor will produce a neutron population:

\[
2.0 \times 10^6 \text{fusion watt} \times 7.72 \times 10^{10} \text{fusion/watt}^{-s} \times 2.5 \text{neutron/fusion} = 1.8 \times 10^{17} \text{neutron/s}
\]

This value of \(1.8 \times 10^{17} \text{neutron}^{-s}\) is called Normalisation value which is used to model the radiation source of TRIGA 2000 in 2 MW. (as the reactor is operated in a power of 1Mw or less then this value might be change on the calculation). To obtain the normalisation value of gamma radiation, the MCNP can be directly used. That value is obtained by providing MCNP input in the form of Tally photon flux in criticality calculations. To get the output in the form of dose rate, several types of tally are used including tally flux detector (F5), tally energy dose (DE), and tally dose function (DF). Tally F5 is used to provide MCNP output in the form of flux values in point and ring-shaped detectors. However, because the TRIGA 2000 reactor geometry is not symmetric, tally the ring detector is not used in the calculation, but only the tally point detector is used. The microscopic cross section data for simulation using ENDF/B-VII and Visual Editor for a visual creation of an MCNPX input.

Particle and Heavy Ion Transport code System, PHITS (Sato et al.2013), is one of the most successful Monte Carlo particle transport simulation codes that are widely used in all over the world. It can deal with the transport of nearly all particles, including neutrons, protons, heavy ions, photons, and electrons, over wide energy ranges using various nuclear reaction models and data libraries [6]. More than 1500 researchers have been registered as PHITS users, and they apply the code to various research and development fields such as nuclear technology, accelerator design, medical physics, and cosmic-ray research. Not only optimization of neutronic performance but also shielding design, nuclear heat estimation, residual radioactivity estimation, and radiation damage estimation of materials were carried out. This full application of such a neutronic calculation code to the design of a large, high-energy accelerator facility was one of the first such trials in the world [5]. Protons, neutrons, photons, electrons, and other minor particles are tracked by PHITS code with detailed models. The transport is based cross section data library JENDL-4.0 for neutron and photon. For drawing particle track and visualization geometry of DLBS using ANGEL software.

3.2. PHITS (Particle and Heavy Ion Transport Code System):

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In this study, the PHITS code can be gotten by using the MCNP calculation. The data from the MCNP is converted to a graphical code in PHITS to get the distribution dose rate gamma and neutron.
The spectrum values of flux from MCNP is copied to the PHITS in form of source section to get the dose.

4. Result and discussion

4.1. Measurement

The measurement was carried out by using the equipment. The neutron read from the neutron survey meter is the dose from fast neutron and to get the dose rate for the thermal neutron, the value from the survey meter should be multiplied by the Factor of calibration 0.023 to not exceed the operation limited condition of 10 μSv/h especially in the control room. The background for the survey meter gamma is 0.04μSv/h and 0 for the neutron detector. The power is different in all data collect. It can be increase from 500kWh up to 800kWh until the reactor is in steady position.

4.1.1. Power at 0 kWh:

This measurement is carried out during the shutdown of the reactor TRIGA 2000. So the neutron measured at that power are 0 but as the reactor contain a source of radiation, the gamma is still measurable.

![Figure 3: Result of measurement dose at the power of 0 kWh](image)

It can be seen at this graph that the highest gamma dose is located in the point 29-32 where is the inner and outer transport of water coolant from the core is set. The gamma doses in that point is always greater than the other place.

4.1.2. Power 500kWh:

The measurement is using the power of 500kWh. If compared with the previous graph, the neutron has a value as already mentioned that the value read from the detector is multiplied to the multiplication factor 0.023. The background of the survey meter gamma is 0.04 μSv/h

![Figure 4: Result of measurement dose at the power of 500 kWh](image)
In this graph, the highest gamma and neutron dose is located in the point 32 with the value of 2.2 µSv/h. There are also some critical high level of neutron dose where the neutron radiation from the core goes directly through the shielding in the point 24. The neutronography and the neutron time of flight facility (TOF) are also located near that point.

4.1.3. Power 700kWh:

![Graph showing dose rate vs time for power 700kWh](image)

**Figure 5:** Result of measurement dose at the power of 700 kWh

The fact of increasing power can increase the gamma and neutron dose. The highest gamma dose is still located in the point 29-32 with the value of 306.4 µSv/h and the neutron is high in the point 24 with the value of 10.166 µSv/h. As already explained earlier, this is one of the critical point where the neutron dose is very high.

4.1.4. Power 800kWh:

![Graph showing dose rate vs time for power 800kWh](image)

**Figure 6:** Result of measurement dose at the power of 800 kWh

This is the highest power level measured in this study. Currently the reactor is operated with the power of 1000kWh or less after being update by the Regulation of the BAPETEN. The dose high is still located in the point 32 with the value of 170 µSv/h. The result of the power of 800 kW is lower than 750 kW because the reactor is already in the steady position. The highest neutron dose is located in the point 23 with the value of 6.946 µSv/h.
4.2. Simulation

The simulation result was gotten by using the MCNP and PHITS software. The code used for this research is already established by other researcher and the author changed some parameters to get the favorable result. The graphical code is using PHITS and the number of particle show the distribution in the geometry of reactor area from the core to all places but as this research is still ongoing, the result is not enough yet to have the whole reactor dose distribution. From this simulation shown that the gamma dose rate at reactor is higher than neutron dose. In Fig. 7(a) shown gamma dose map at the TRIGA reactor. The gamma dose rate at the radiation worker places like deck and BSF about $1 - 100 \, \mu\text{Sv/h}$. As neutron dose map, statistical error for gamma dose rate also high. neutron dose rate at the radiation worker places like deck and BSF about $1 - 10 \, \mu\text{Sv/h}$ as shown in Fig. 7 (b). But statistical error in that area still high.

![Figure 7: Gamma and neutron map in TRIGA 2000 using PHITS](image)

4.3. Optimization of the radiation protection

The measurement of any radiation sources should be referred to the Radiation protection principle. Before going to the radiation field the person or officer have to make sure that the appliance is on and measure the background radiation. The survey around the source should be carried out if possible with always keeping an eye meter with attention to the screen or listen to audio. To stay safe, the officer should reduce as low as possible the time for the measurement and do not stay too long in the radiation field.

Beam port conditions and neutronography facilities have been closed. In additions a Pb shields and concrete in the reactor tank have been made to maintain exposure in control room that remain below the operational limited condition and the exposure derivative per day set. Besides that, on the surface the BSF (Bulk Shielding Facility) which is also one of the place that has highest gamma and neutron dose has been closed by Pb sheet to minimize exposure. Until radiation exposure at the control room at the time the is smaller dan $10 \, \mu\text{Sv/h}$ for external exposure (gamma and neutron exposure). [3]
Based on the calculations in the previous LAK, the shield of the reactor structure was designed for a maximum power of 250 kW, it is made of reinforced concrete with a thickness of 2.44 meters. Radiation levels around the reactor, above and around the reactor, are kept as low as possible so that the accepted dose of the operational limit condition applies.

5. Conclusion:

The measurement has been carried out in the TRIGA 2000 at BATAN Bandung by using personal dosimeter for the individual protection and the survey meter during the data collect. In generally the dose measured didn’t exceed the operation limited condition dose nor the limitation dose set by the IAEA standard for the gamma and neutron dose. The highest dose observed was during the power of 700kWh with the value of 306.4 µSv/h at the point 32 but referred with the operation limited condition which is 625 µSv/h for gamma dose it still not exceed the limit. For neutron dose the limitation is 10 µSv/h and the dose measured by the detector after multiplying with the multiplication factor 0.023 but there is a point at 24 with the 700 kWh, the maximum value is 13.8 µSv/h. As it is not in the control room yet, it can be acceptable but might be mitigate by approving the shielding. The simulation part of the research is still ongoing to be compared with the measurement results. The research is still ongoing until the whole reactor distribution dose will be complete. The radiation protection is already optimized by using the Pb shield in the Bulk Shielding Facility to mitigate the radiation dose through the control room. The technique will be improved for the optimization of the radiation protection system with the Individual protection of the worker and the public surrounding the reactor.

6. References:

[1] IAEA, Safety Standards Series, Occupational Radiation Occupational Radiation Protection Safety Guide No. RS-G-1.1
[2] Rasito Tursinah, Putu Sukmabuana Pemodelan Dosis Neutron Dan Gamma Di Reaktor Triga 2000 Dengan Metode Monte Carlo Mcnp5, January 2009
[3] BATAN Safety analysis Report (2016)
[4] Masahide Harada, Fujio Maekawa, Kenichi Oikawa, Shin-ichiro Meigo, Hiroshi Takada and Masatoshi Futakawa Japan Atomic Energy Agency, 2-4 Shirakata, Tokai-mura, Naka-gun, Ibaraki-ken, 319-1195, Japan Application and Validation of Particle Transport Code PHITS in Design of J-PARC 1 MW Spallation Neutron Source, Progress in Nuclear Science and Technology, Vol. 2, pp.872-878 (2011)
[5] Tatsuhiko Sato, Koji Nita, Norihiro Matsuda, Shintaro Hashimoto, Yosuke Iwamoto, Takuya Furuta, Shusaku Noda, Tatsuhiko Ogawa, Hiroshi Iwase, Hiroshi Nakashima, Tokio Fukahori, Keisuke Okumura, Tetsuya Kai, Satoshi Chiba, Lembit Silver Overview of particle and heavy ion transport code system PHITS