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Radiological safety analysis of the SAMOP reactor experimental facility

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Abstract. The reactor subcritical assembly for 99Mo production (SAMOP) experimental facility is a nuclear reactor operating under subcritical conditions. SAMOP reactor is fueled by uranyl nitrate inside an annular tube surrounded by the TRIGA reactor fuels. The external neutron source required for the SAMOP reactor operation comes from the Kartini reactor radial beam-port. The aim of the analysis is to determine the estimated personnel radiation dose for normal conditions and severe postulated accident. The analysis methods is a calculation of source term, radiation propagation to the surrounding area, and estimation of personnel radiation dose rate for normal and severe accident conditions. The analysis result shows that the effective radiation dose received by personnel in the working area of the SAMOP reactor is below the dose limit for radiation worker at the Kartini reactor, i.e. 7 μSv/h. Whereas, in the postulated severe accident condition, the highest exposure rate at 1 m above and beside the reactor cooling tank are 4.19 mSv/h and 1.35 mSv/h respectively.

1. Introduction

The ^{99}Mo isotope is a ^{99m}Tc generator, of which ^{99m}Tc is the most widely used radioisotope for diagnostics in the nuclear medicine [1,2,3]. The need of the ^{99m}Tc isotope increased and can't be fulfilled because the ^{99}Mo commonly used as a ^{99m}Tc generator can only be produced in a nuclear reactor. The half-life of ^{99}Mo which only 66 hours is also a factor making it difficult to store and mobilize its spread. This problem can be solved by shortening the ^{99}Mo production process as well as shortening the mobilization of radioisotopes spreading distance.

In general, the ^{99}Mo production is done by splitting the ^{235}U nuclide, which ^{235}U is irradiated with thermal neutrons inside a nuclear reactor so that a critical reactor is required. The licensing issues for critical reactors are complex because they have to fulfill strict requirements. This problem requires a very huge funds to realize, excluding the cost of waste management products. To solve this problem BATAN or the National Nuclear Energy Agency designed a reactor that could produce ^{99}Mo but did not categorized as critical reactor and did not use high-enriched uranium. This reactor is simple and safer. The reactor applies subcritical reactor method known as SAMOP or subcritical assembly for ^{99}Mo production [4].

The subcritical reactor system needs an external neutron source which can be supplied by a neutron generator or an isotropic neutron source. The SAMOP experimental facility as a test facility which use an external neutron source from the beam-port of Kartini TRIGA reactor, which has been identified suitable for this purpose [5]. The analysis is a part of safety analysis report of SAMOP experimental facility [6,7].

SAMOP system is similar with an Aqueous Homogenous Reactor (AHR) but it is operated in a subcritical condition. AHR is a type of reactor in which uranium is dissolved in water. The fuel used is a mixture of coolants such as water which also act as a moderator and uranium salt, often referred to as homogeneous reactors. The high negative temperature reactivity coefficient makes the AHR more stable than conventional reactors. Another positive aspect of AHR is its small size and low total power [8,9].

The purpose of this research is to analyze the radiological safety of SAMOP reactor experimental facility and to determine the estimated personnel radiation dose for normal conditions and severe postulated accident. The tool used for this analysis is MCNPX software, which is a general purposes Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies [10,11].

2. Basic theory

2.1. Radiation dose and MCNP simulation

The effective dose calculation resulting from the radioisotope present in the reactor core is done by multiplying the radioisotope activity with the corresponding specific gamma-ray dose constants. The specific gamma-ray dose constants of the radioisotopes is derived from the ORNL technical document for dosimetry and radiological assessment [12]. The specific gamma ray dose constant is a useful quantity in radiation protection applications.

The dosage limit used according to the Indonesian Nuclear Energy Regulation Agency (BAPETEN) i.e. Act No. 4 of 2013 Article 15 [13]. The dose limit value for radiation workers, the effective doses should not exceed in averaged 20 mSv (twenty milisievert) per year within a period of 5 years, so doses accumulated in 5 years should not exceed 100 mSv. The other limitations are the effective dosage of 50 mSv in a year; the equivalent dose for an average eyepiece of 20 mSv per annum in a period of 5 years and 50 mSv in a year; and dose equivalent for the skin, hand or foot of 500 m Sv per year.

The dose calculation is using the MCNP (Monte Carlo N Particle) which is a software to complete the transport calculation of neutron /photon /electron particles. MCNP can also perform a combination of particle transport called transport mode, including a combination of neutrons, which means photons are produced from the interaction of neutrons with matter, the combination of neutrons-photons-electrons, photons-electrons or electrons-photons.

Figure 1 shows a sampled neutron track as follows: the incident neutron starts in the void region and enters the fissionable material. According to the material properties (density, temperature) and the energy of the incident neutron, the Monte Carlo code generates a one-dimensional probability distribution function. The Monte Carlo code checks the energy dependent cross section data at any position in the fissionable material: 1, 2, 3, 5, and 7 and generates a reaction probability distribution [11]. This distribution creates the basis to determine a random number for the type of the reaction between the neutron and the particular isotope in the material.

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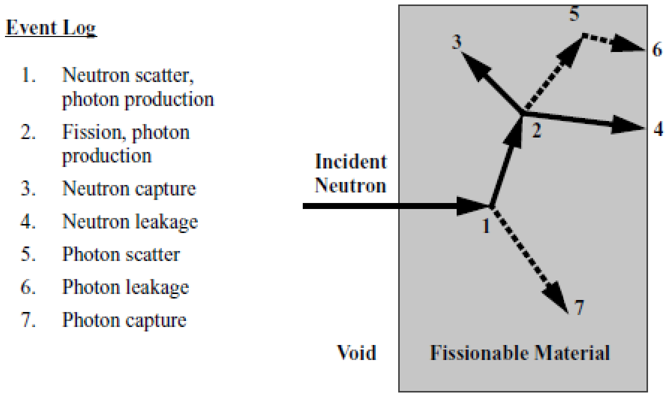

Figure 1. Neutron path in Monte Carlo code [12].

The first interaction in figure 1 is a neutron scatter- photon production reaction. The neutron is scattered in the direction shown and undergoes a fission reaction (position 2) and two additional neutrons are generated. The generated neutron at position 2 and photon at position 1 are absorbed and their histories are terminated at positions 3 and 7. The fission-produced photon has a collision at event 5 and leaks out of the material at position 6. Other terminations occur at the events 4 and 6. In case of absorption or leaving the volume of interest, any contribution made by the particle history is recorded. Secondary particles which result from events like (n, 2n) are then individually transported until they are exhausted, at which time a new source particle is started.

2.2. Description of SAMOP

SAMOP experimental facility is a test facility for ^{99}Mo production by using an external neutron source from the radial beam-port of Kartini reactor. The external neutron source has been identified as thermal neutron in order of 10^8 n/cm^{2}s [7]. The SAMOP core consists of annular cylindrical tube containing uranyl nitrate [UO_2 $(NO_3)_2$] or UN as fuels and target, surrounded by ring of UO_2 $(NO_3)_2$ tubes. The TRIGA fuel elements is loaded in the ring together with UO_2 $(NO_3)_2$ tubes to increase neutron multiplication factor. The enrichment of UN is 19.75% ^{235}U.

The SAMOP reactor core and reflector is a cylinder with 40.4 cm in diameter and 43 cm in height. The core and reflector is located in the cooling tank with diameter and height of 120 cm and 400 cm respectively. The radiation shielding of SAMOP consisted of paraffin and barite concrete. The shielding at the front side is a combination of paraffin and concrete both with have the same thickness i.e. 55 cm. Figure 2 shows the SAMOP experimental facility, dimensions, and the number of cell stating the area or location where the radiation exposure will be analyzed.

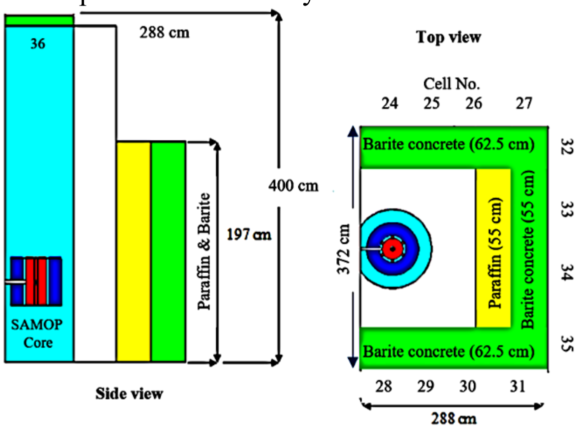

Figure 2. SAMOP experimental facility (side and top view).

3. Method

The analysis methods is a calculation of source term, radiation propagation to the surrounding area, and estimation of personnel radiation dose rate for normal and severe accident conditions. The worst accident is assumed to be a failure in loading-unloading of uranyl nitrate tube as SAMOP fuels or TRIGA reactor fuel from the SAMOP reactor tank causing 1 fuel to fall and break.

The calculation tool is used the MCNPX code 2.6.0. MCNPX is a general purposes Monte Carlo radiation transport code designed to track many particle types over broad ranges of energies. It is the next generation in the series of Monte Carlo transport codes that began at Los Alamos National Laboratory nearly sixty years ago. MCNPX 2.6.0 is the latest Radiation Safety Information Computational Center (RSICC) release of the code [11].

Reactor core modeling using MCNPX can be done using vised application provided by the MCNPX, or by creating its own input code using a text editor application. The MCNP input code structure consists of cell cards, surface cards and data cards. The cell card contains information about part of the reactor core that is bounded by surface defined on the surface card. Material information such as the type of material used, material density, cell volume (optional), and the type of particles tracked within those cells. The data card contains information about the details of the material information used in the previous cell card and what type of calculation that user wants MCNPX to solve.

Human healthy tissue on MCNPX is modeled by soft tissue. Soft tissue has elements similar to human healthy tissue and is used to record the dose rate that comes out of the SAMOP experimental facility. In this study, soft tissue is assumed to be radiation workers outside the radiation shield of the SAMOP experimental facility. The constituent elements of soft tissue can be seen in Table 1, where the most dominant elements are O, C, H and N. The organ-weight factor is the sensitivity level of healthy human or organ tissue to the stochastic effects of radiation. The soft tissue constituent elements used in MCNP are written using MCNP material code in ZZZAAA format, where Z is the atomic number and A is the mass number of the soft tissue element.

Table 1. Elements of *soft tissue*

Elements	MCNP material code	Organ weight factor
H	1001	0,10454
C	6012	0,22663
N	7014	0,02490
O	8016	0,63525
Na	11023	0,00112
Mg	12000	0,00013
Si	14000	0,00030
P	15031	0,00314
S	16032	0,00204
Cl	17000	0,00133
K	19000	0,00208
Ca	20000	0,00024
Fe	26000	0,00005
Zn	30000	0,00003
Rb	37087	0,00001
Zr	40000	0,00001

4. Result and discussions

The radiation exposure calculation results at the predetermined area surrounding the SAMOP reactor i.e. at the cell numbers such as shown in figure 2, is listed in table 1. It is shown that the maximum neutron dose rate is located at cell no. 24 and 28 i.e. 2.46 – 2.76 μSv/h. The values still bellows the radiation dose limit of 7 μSv/h for radiation worker at the Kartini research reactor as predefined. As indicated table 2 and figure 3, the higher radiation dose rate both neutron and gamma radiation is located at the side part of SAMOP facility.

Table 2. The radiation exposure calculation results at the area surrounding the SAMOP reactor.

Cell No.	Neutron dose rate (µSv/h)	Photon / gamma dose rate (µSv/h)
24	2.46	$3.51\ 10^{-11}$
25	$1.44\ 10^{-1}$	$2.88\ 10^{-11}$
26	2.24	$1.68\ 10^{-11}$
27	$2.81\ 10^{-4}$	$2.88\ 10^{-12}$
28	2.76	$4.73\ 10^{-11}$
29	$9.03\ 10^{-1}$	$4.73\ 10^{-11}$
30	$2.55\ 10^{-1}$	$1.34\ 10^{-11}$
31	10^{-5}	10^{-13}
32	10^{-5}	$1.26\ 10^{-12}$
33	10^{-5}	$2.63\ 10^{-12}$
34	10^{-5}	$1.29\ 10^{-12}$
35	10^{-5}	$2.33\ 10^{-12}$
36	1.32	$1.62\ 10^{-10}$

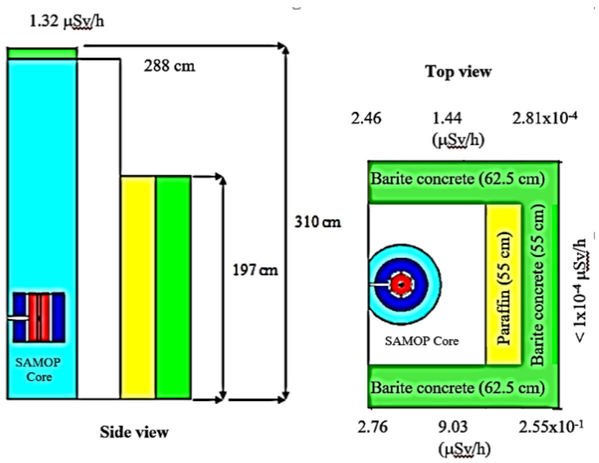

Figure 3. Calculated radiation exposure surrounding SAMOP reactor.

The smaller values is found at cell numbers located in front of SAMOP facility. This phenomenon is reasonable according to geometry of the system where cell no. 24 and 28 are located at the front side of the beam-port neutron source. At this location, the probability of neutron and gamma scattering radiations is the higher comparing to the other areas. Although, the highest radiation dose values calculated are all still below the specified limit. It shown in table 2 or figure 3 that the maximum effective radiation dose received by personnel in the working area of the SAMOP experimental facility is 2.76 µSv/h, this value is below the dose limit for radiation worker at the Kartini reactor, i.e. 7 µSv/h.

The analyses results have met the criteria determined by national nuclear regulatory body (BAPETEN) i.e. the dose rate of all soft tissue not exceeding 10 µSv/h [13]. The calculation results is also shown that paraffin is good neutron radiation shielding, in fact, this is because it has a mass of particles similar to that of neutrons. When crashing paraffin, the neutron will lose a lot of kinetic energy because the energy is transferred to the shielded material particles. While barite concrete has a heavy

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density and high atomic number so it has the ability to absorb the photon radiation beam at the SAMOP reactor experimental facility.

SAMOP experimental facility operation period is 6 days continuously (per batch) and then shut-down for ^{99}Mo isotope extraction. For the conservative analysis, the calculation of source term i.e. isotopes radiation sources generated in the SAMOP reactor core is done after 30 days operation or 5 batches. The result shows that the total radioactivity in the in the reactor core until day 30th reaches 516.681 Ci with the effective gamma-ray dose of 1557.523 mSv/h. The largest radioactivity is come from of ^{239}Np which is 89.32 Ci, followed by ^{133}Xe 54.68 Ci and then ^{99}Mo that can reach 51.29 Ci. The radioactivity of radioactive substances in the reactor core increases exponentially, and the effective dose of gamma radiation increases significantly after 24 days of operation, the result is in accordance with research previously done by other researcher [14].

The worst or severe accident is assumed 1 uranyl nitrate (UN) tube as SAMOP fuel fall and break. The source term calculation result in an UN tube located in the center of the SAMOP reactor core shows that the total activity reach 7.00 Ci with the effective gamma-ray dose of 22.33 mSv/h. Therefore, in the postulated severe accident condition, the highest exposure rate at 1 m above and beside the reactor cooling tank can be calculated i.e. 4.19 mSv/h and 1.35 mSv/h respectively.

It is also possible that volatile radioactive materials will be released from the reactor confinement, which may lead to an internal dose (Iodine dose) due to inhalation and precipitation in the thyroid and external dose caused by the release of noble gases into the environment. The volatile radioactive materials arev dominated by gas fission products mainly Kr, I, and Xe isotopes. The estimation of internal dose rate in the severe accident condition have been calculated and the result is presented in table 3. It is shown that the total radiation dose rate at 30 m from SAMOP is 0.2004 mSv/h. The distance of 30 m from SAMOP reactor facility is considered due to the presence of radiation workers.

Table 3. The radioactivity calculation result of gas fission product (Kr, I and Xe isotopes) in an UN tube.

Isotopes	Radio-activity (Ci)	Total (Ci)	Total (Ci)	*Total rad. rate (mSv/h)
Kr-85m	6.123 10^{-1}	1.189	49.74	0.2004
Kr-88	4.159 10^{-1}			
I-131	3.450	25.48		
I-132	5.664			
I-133	1.046 10^{1}			
I-135	5.911			
Xe-133	8.211	23.07		
Xe-	2.511 10^{-1}			
Xe-135	1.363 10^{1}			
Xe-	9.468 10^{-1}			

*Total radiation rate at 30 m from SAMOP reactor

5. Conclusion

The analysis results shown that the operation of SAMOP experimental facility is safe from radiology point of view. It can be concluded that the the maximum effective radiation dose received by personnel in the working area of the SAMOP experimental facility is 2.76 μSv/h, this value is below the dose limit for radiation worker at the Kartini reactor, i.e. 7 μSv/h, as well as the dose limit predetermined by the national nuclear regulatory body of 10 μSv/h.

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