A study of radiation dose for the anticipated accident condition in the SAMOP reactor experimental facility

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Abstract. A study of radiation dose for the accident condition in the Subcritical Assembly for 99Mo Production (SAMOP) reactor experimental facility has been done. The main purpose of this study is to estimate the radiation dose received by worker as well as society during accident condition in the SAMOP reactor experimental facility. In this study, one of the worst accident scenario that has the highest probability is postulated as one of TRIGA fuel fall down in to the SAMOP reactor core during loading-unloading process. That accident inflicts one of this TRIGA fuel broken and radioactive materials are released. The radiation dose is estimated based on the radioactive source-term from ORIGEN 2.1 computer code calculation. Furthermore, the total effective dose equivalent (TEDE) received by the society is calculated using dispersion model of Hotspot 3.0 computer code. From this study, the total radiation dose received by the radiation worker during this accident condition is 4.5 mSv and the maximum TEDE received by society during this accident condition is 0.031 mSv. This study proves that the radiation dose received by worker as well as society during this accident condition is still below the limit appointed by regulatory body (BAPETEN).

1. Introduction
The 99Mo radioisotope is indispensable as a 99mTc generator, of which 99mTc radioisotope is the most widely used radioisotope for diagnostics in the nuclear medicine [1,2,3]. A Subcritical Assembly for 99Mo Production (SAMOP) reactor experimental facility as a utilization of Kartini TRIGA reactor, is being developed at the Center for Accelerator Science and Technology. Recently, the license for operating SAMOP reactor experimental facility has been granted by the Regulatory Body (BAPETEN) on October 2018. The design of SAMOP reactor is based on fission reaction of 235U which is occurred in the subcritical condition. This method is simpler compared with the 99Mo production using a thermal neutron from the critical reactor. The licensing issues for critical reactors are complex because they have to fulfil strict requirements. This problem requires a very huge funds to be realized [4,5]. Furthermore, the radioactive waste that produced from critical reactor is more complex than SAMOP reactor. The SAMOP core consists of annular cylindrical tube containing uranyl nitrate [UO2(NO3)2] as fuels and target, surrounded by ring of UO2(NO3)2 fuel rods. Uranyl nitrate that is used in this research has been enriched less than 20% of 235U [4,5]. The TRIGA fuel elements is loaded in the ring together with UO2(NO3)2 tubes to increase neutron multiplication factor. The number of TRIGA fuel loaded into SAMOP reactor are limited to keep k-eff value below 1. The SAMOP reactor uses an external neutron source from the beam-port of Kartini TRIGA reactor, i.e. radial beam-port. This beam-port was directly connected with Kartini TRIGA reactor core.
By the fact that the SAMOP reactor is a nuclear experimental facility that using a fission reaction of $^{235}$U, in which this reaction inflicts an external radiation exposure caused by gamma ray from fission and activation products. In other hand, this reaction also inflicts an internal exposure because several fission and activation products that produced from this reaction are volatile. The study of radiation dose received from the SAMOP reactor experimental facility is very important to ensure radiation safety for the worker as well as society, especially during accident mitigation condition. In this paper, the worst accident scenario that has highest probability was postulated [4].

Based on act 04/2013 of Indonesia Nuclear Energy Regulatory Agency (BAPETEN), the utilization of nuclear experimental facility should comply a radiation protection and safety requirement [6]. One of radiation protection principle that should be applied in the nuclear experimental facility was optimization, in which the radiation dose that received by the worker and the society during normal as well as abnormal (accident) condition should be reduced as long as reasonably applicable. For this reason, the Center of Science and Accelerator Technology has established a dose constraint value of 15 mSv/year [7]. In other hand, in order to ensure radiation protection and safety requirement of SAMOP reactor, it is necessary to calculate the radiation dose that received by the worker and also the society in the vicinity of SAMOP reactor experimental facility for both normal and abnormal condition.

The radiation source-term analysis of SAMOP reactor has been done by using MCNP computer code [8,9], but this analysis was performed by calculating radiation source-term just in a normal condition. In this paper, the source-term analysis for abnormal (accident) condition was performed using ORIGEN 2.1 computer code that developed by Oak Ridge National Laboratory (ORNL). The main function of ORIGEN 2.1 is to calculate the composition of radionuclide and also several nuclear material characteristics such as build-up, decay and processing of radioactive material [10]. The result of this source-term analysis is used to calculate a radiation dose that will be received by the worker during an accident condition.

In this paper, the total effective dose equivalent (TEDE) that received by the society is calculated using dispersion model of Hotspot 3.0 computer code. The Hotspot 3.0 codes were created to provide a fast emergency actions planner. The Hotspot atmospheric dispersion models codes are a first-order approximation of the radiation effects associated with the short-term atmospheric release of radioactive materials [11].

2. Methodology
The radiation dose calculation in the SAMOP reactor experimental facility for the accident condition was performed using two main steps. First step is radiation source-term analysis and the second step is personal radiation dose calculation by using that source-term analysis result.

2.1. Radiation source-term analysis of SAMOP reactor
The source-term analysis of SAMOP reactor is performed using ORIGEN 2.1 computer code. This calculation is performed by assuming that the worst accident scenario is one of TRIGA fuel fall down in to the SAMOP reactor core during fuel loading-unloading process and fuel cladding is ruptured. This accident inflicts one of the TRIGA fuel broken and radioactive materials released at 1 meter from the bottom of SAMOP reactor tank position [4].

This calculation uses several data input such as neutron flux [12,13], fuel material composition and also operation time of Kartini reactor. The output from this calculation is radioactivity of several fission and activation products. The flowchart of radiation source-term analysis is described on Figure 1.

2.2. Radiation dose calculation for the radiation worker
Personal radiation dose calculation is performed by considering both of internal radiation dose and external radiation dose that received by worker during mitigation of accident. The external radiation dose is calculated from the exposure rate constant ($\Gamma$), where it relates the activity of a point isotropic radiation source to the exposure rate in air above the SAMOP reactor. The correlation between exposure rate at a given distance and the exposure rate constant ($\Gamma$) is given by Equation 1.
\[
\dot{X} = I_\delta \frac{A}{d^2} \cdot e^{-\mu x}
\]  

(1)

Where A is the source activity, \(d\) is distance to the source, and \(\delta\) is a minimum cut-off energy, which determines the minimum energy photon that can contribute to the exposure, \(\mu\) is linear attenuation coefficient and \(x\) is the thickness of shielding material [14].

The internal radiation dose is calculated from the radionuclide activity that has a probability to enter the body by inhalation process, where the correlation between effective radiation dose and inhalation coefficient is given by equation 2 [15].

\[
E_{t,inh} = \sum_j e(g)_{j,inh} I_{j,inh}
\]

(2)

\(E_{t,inh}\) : Radiation effective dose (Sv).

\(e(g)_{j,inh}\) : Effective dose per activity of radionuclide \(j\) as intake by inhalation of group \(g\), as published on ICRP 119 (Sv.Bq\(^{-1}\)).

\(I_{j,inh}\) : The activity of radionuclide \(j\) that entered by inhalation (Bq).

2.3. **TEDE calculation using Hotspot 3 computer code**

The radiation dose (TEDE) received by the society in the vicinity of SAMOP reactor experimental facility during accident condition was calculated using dispersion model of Hotspot 3.0 computer code [11]. In order to obtain a maximum TEDE, several input data needed among other meteorological data, source-term of radionuclides released through the reactor stack, and dispersion model that is in accordance with postulated accident conditions. The source-term of radionuclides that used in this calculation are obtained from Origen 2.1 calculation. Figure 1 describes the flowchart to calculate TEDE using Hotspot 3.0.

![Figure 1](image-url)  

Figure 1. The flowchart of radiation source-term analysis and personal dose calculation during accident condition, (A) TEDE calculation for the society; (B) Radiation dose calculation for the worker.

3. **Result and discussion**

In this paper, the worst accident scenario is determined by assuming one of Kartini TRIGA fuel falls down and broken at 1 meter from the bottom of SAMOP reactor tank. In this scenario, during loading-
unloading process the SAMOP reactor tank was fully filled by water as reactor coolant. The source-term analysis result was described on the Table 1. In this source-term analysis, the neutron flux that used was calculated by using MCNP computer code [12]. Another input that used i.e. the material composition of Kartini TRIGA fuel and time of fuel operation in the TRIGA reactor as described in the flowchart on the Figure 1.

In this worst accident, the highest gamma exposure rate in 1 meter above SAMOP reactor tank is 419 mR/h. This value is obtained by considering a water as a gamma radiation shielding. Furthermore, the highest gamma exposure rate at the point 1 meter beside of SAMOP reactor tank is 135 mR/h. For this accident scenario, the mitigation action that will be carried out is closing the SAMOP reactor tank using lead with 7.7 half value thickness (100 mm with $\bar{E}_y$~2.5 MeV). By this action, the radiation dose at 1 meter above SAMOP reactor tank is reduced to 20 μSv/h. During this mitigation action, the external radiation dose that will be received by worker is 3.1 mSv. This value is obtained by assuming that those mitigation action take about 45 minutes.

In the Table 2. The internal radiation dose during accident condition is described on the Table 2.

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### Table 1. Radiation source-term of TRIGA reactor nuclear spent fuel.

| Radionuclide | Gamma Factor (R.cm$^2$/mCi.h)$^{[14]}$ | Activity (Bq) | Radionuclide | Gamma Factor (R.cm$^2$/mCi.h)$^{[14]}$ | Activity (Bq) |
|--------------|--------------------------------------|--------------|--------------|--------------------------------------|--------------|
| Cr-51        | 0.178                                | $3.23 \times 10^{17}$ | I-133       | 3.47                                | $2.47 \times 10^{18}$ |
| Kr-85m       | 0.79                                 | $3.10 \times 10^{16}$ | I-135       | 8.04                                | $8.77 \times 10^{19}$ |
| Sr-91        | 3.86                                 | $4.87 \times 10^{10}$ | Xe-133      | 0.568                               | $2.35 \times 10^{10}$ |
| Y-91         | 0.0163                               | $1.54 \times 10^{11}$ | Xe-133m     | 0.639                               | $1.46 \times 10^{11}$ |
| Y-91m        | 3.04                                 | $3.09 \times 10^{11}$ | Xe-135      | 1.38                                | $3.11 \times 10^{11}$ |
| Y-92         | 1.34                                 | $4.06 \times 10^{10}$ | Xe-135m     | 2.54                                | $1.41 \times 10^{11}$ |
| Y-93         | 0.488                                | $2.33 \times 10^{9}$  | Cs-137      | 3.43                                | $7.87 \times 10^{9}$  |
| Zr-95        | 4.12                                 | $3.33 \times 10^{11}$ | Ba-140      | 1.14                                | $3.63 \times 10^{11}$ |
| Nb-95        | 4.29                                 | $3.17 \times 10^{11}$ | La-140      | 11.7                                | $4.16 \times 10^{11}$ |
| Nb-97        | 3.77                                 | $2.90 \times 10^{10}$ | Sb-126      | 15.6                                | $1.83 \times 10^{10}$ |
| Mo-99        | 0.917                                | $5.26 \times 10^{10}$ | Sb-127      | 3.96                                | $4.16 \times 10^{10}$ |
| Tc-99m       | 0.795                                | $5.07 \times 10^{10}$ | Sb-129      | 7.81                                | $1.59 \times 10^{10}$ |
| Ru-103       | 2.87                                 | $5.40 \times 10^{11}$ | I-131       | 2.2                                 | $1.45 \times 10^{11}$ |
| Ru-105       | 4.44                                 | $3.65 \times 10^{10}$ | Ce-141      | 0.453                               | $4.14 \times 10^{11}$ |
| Rh-103m      | 0.15                                 | $4.87 \times 10^{11}$ | Ce-143      | 1.85                                | $3.13 \times 10^{11}$ |
| Rh-105       | 0.44                                 | $6.92 \times 10^{9}$  | Ce-144      | 0.135                               | $1.54 \times 10^{11}$ |
| Ag-110m      | 15                                   | $4.60 \times 10^{17}$ | Ce-144      | 0.0994                              | $2.87 \times 10^{12}$ |
| Ag-111       | 0.15                                 | $2.42 \times 10^{10}$ | Pr-145      | 0.0994                              | $2.87 \times 10^{12}$ |
| Te-127       | 0.0287                               | $7.13 \times 10^{17}$ | Nd-147      | 0.931                               | $1.29 \times 10^{11}$ |
| Te-127m      | 0.448                                | $7.28 \times 10^{17}$ | Pm-149      | 0.0659                              | $7.03 \times 10^{10}$ |
| Te-129       | 0.523                                | $5.71 \times 10^{10}$ | Pm-151      | 1.9                                 | $2.77 \times 10^{10}$ |
| Te-129m      | 0.497                                | $8.77 \times 10^{9}$  | Eu-156      | 6.21                                | $1.18 \times 10^{10}$ |
| Tc-131       | 2.36                                 | $3.30 \times 10^{10}$ | Pu-236      | 0.209                               | $2.37 \times 10^{11}$ |
| Tc-131m      | 8.1                                  | $1.47 \times 10^{10}$ | Am-242      | 0.479                               | $3.48 \times 10^{10}$ |
| Tc-132       | 1.93                                 | $5.86 \times 10^{10}$ | Am-242m     | 0.392                               | $3.50 \times 10^{10}$ |
| I-132        | 12.5                                 | $6.04 \times 10^{10}$ | Crn-242m    | 0.161                               | $1.57 \times 10^{10}$ |

Beside of the external exposure, the internal radiation dose also will be received by the SAMOP worker due to inhalation of radioactive material. It is due to several fission products of TRIGA nuclear spent fuel are volatile. The internal radiation dose during accident condition is described on the Table 2.

In the Table 2. The internal radiation dose received by the worker during mitigation action for one hour is 1.88 mSv. By the fact that the mitigation action is estimated about 45 minutes so the internal radiation dose received by worker is 1.4 mSv.

The total effective dose that received by the worker during accident condition that-obtained from both external and internal radiation dose is 4.5 mSv. This value is still below the dose constraint that determined in The Center of Science and Accelerator Technology which is 15 mSv/year[7].
During accident condition, the radiation dose received by the society in the vicinity of SAMOP reactor experimental facility was obtained by calculating TEDE using Hotspot 3.0. The meteorological data used in this calculation is obtained from the Meteorological Station located about 3 kilometers from the SAMOP reactor experimental facility. The results of the TEDE calculation were described on Table 3. The maximum TEDE at 100 meters from reactor stack is 0.031 mSv. This value is still below the dose limit that appointed by regulatory body i.e. 1 mSv/year. The result from this study shows that the design of SAMOP reactor satisfy with radiation protection and safety requirement during the accident condition. It can be proved from the radiation dose that received by the radiation worker as well as the society in the reactor vicinity are still below the dose limit that designated by BAPETEN as regulatory body in Indonesia.

**Table 2. Internal radiation dose calculation from Krypton, Iodine and Xenon isotopes**

| radionuclide | Activity (Bq) | Total Activity by inhalation (Bq) | Inhalation Coefficient\(\text{e}(\text{g})_{\text{inh}}\) (Sv.Bq\(^{-1}\)) | Effective dose\(\text{e}(\text{g})_{\text{inh}}\) (mSv) |
|-------------|---------------|----------------------------------|-------------------------------------------------|---------------------------------|
| Kr-85m      | 3.1 x 10\(^{-06}\) | 1.57 x 10\(^{-09}\) | -                                              | -                               |
| I-131       | 1.45 x 10\(^{11}\) | 1.69 x 10\(^{08}\) | 1.1 x 10\(^{8}\) | 1.86 x 10\(^{00}\) |
| I-132       | 6.04 x 10\(^{10}\) | 7.03 x 10\(^{04}\) | 2.0 x 10\(^{-10}\) | 1.41 x 10\(^{02}\)   |
| I-133       | 2.47 x 10\(^{08}\) | 2.88 x 10\(^{12}\) | 2.1 x 10\(^{09}\) | 6.05 x 10\(^{04}\)   |
| I-135       | 8.77           | 1.02 x 10\(^{05}\) | 4.6 x 10\(^{-10}\) | 4.70 x 10\(^{-12}\) |
| Xe-133      | 2.35 x 10\(^{11}\) | 1.19 x 10\(^{08}\) | -                                              | -                               |
| Xe-133m     | 1.46 x 10\(^{09}\) | 7.40 x 10\(^{05}\) | -                                              | -                               |
| Xe-135      | 3.11 x 10\(^{-04}\) | 1.57 x 10\(^{-11}\) | -                                              | -                               |
| Xe-135m     | 1.41           | 7.11 x 10\(^{04}\) | -                                              | -                               |
| **Total**   |                | 1.88                             |                                                |                                 |

\(^*\) 1 TRIGA nuclear spent fuel

\(^*\) Internal radiation dose by inhalation in 1 hour.

**Table 3. Target organ committed dose equivalent calculation (sv), at 100 meters using Hotspot 3.0.**

| Organ            | Committed dose equivalent (Sv) | Organ            | Committed dose equivalent (Sv) |
|------------------|-------------------------------|------------------|-------------------------------|
| Skin             | 1.3 x 10\(^{-06}\)           | LLI Wall         | 6.5 x 10\(^{-07}\)           |
| Surface bone     | 1.4 x 10\(^{-06}\)           | Esophagus        | 1.3 x 10\(^{-06}\)           |
| Spleen           | 7.7 x 10\(^{-07}\)           | Testes           | 7.8 x 10\(^{-07}\)           |
| Breast           | 1.1 x 10\(^{-06}\)           | Brain            | 7.1 x 10\(^{-07}\)           |
| ULI Wall         | 6.7 x 10\(^{-07}\)           | Thyroid          | 1.0 x 10\(^{-03}\)           |
| Thymus           | 1.4 x 10\(^{-07}\)           | Liver            | 7.6 x 10\(^{-07}\)           |
| Kidneys          | 7.3 x 10\(^{-07}\)           | Adrenals         | 7.3 x 10\(^{-07}\)           |
| Pancreas         | 7.0 x 10\(^{-07}\)           | SI Wall          | 6.5 x 10\(^{-07}\)           |
| Lung             | 3.3 x 10\(^{-06}\)           | Bladder Wall     | 6.6 x 10\(^{-07}\)           |
| Red Marrow       | 8.8 x 10\(^{-07}\)           | Muscle           | 9.5 x 10\(^{-07}\)           |
| Ovaries          | 6.3 x 10\(^{-07}\)           | Uterus           | 6.3 x 10\(^{-07}\)           |
| Stomach Wall     | 1.0 x 10\(^{-06}\)           |                  |                               |
| **Total equivalent dose effective (Sv)** | 3.1 x 10\(^{-05}\) |                  |                               |

**4. Conclusion**

A study of personal radiation dose received by the worker of SAMOP reactor during accident condition was carried out. The total radiation dose for both internal and external dose is 4.5 mSv, which is still below the dose constraint that appointed in the Center for Accelerator Science and Technology i.e. 15 mSv/year. The maximum TEDE that will be received by society during this accident condition is 0.031 mSv, which is still below the designated dose limit for the society i.e. 1 mSv/year. This study proves that the radiation dose received by worker as well as society during accident condition are still below the limit that appointed by regulatory body. The result of this calculation will be used to determine an
appropriate action due to accident in the SAMOP reactor and will be used as a reference to develop emergency preparedness program.

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