Analysis of radiation safety for Small Modular Reactor (SMR) on PWR-100 MWe type

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Abstract. Indonesia as an archipelago country, including big, medium and small islands is suitable to construction of Small Medium/Modular reactors. Preliminary technology assessment on various SMR has been started, indeed the SMR is grouped into Light Water Reactor, Gas Cooled Reactor, and Solid Cooled Reactor and from its site is group into Land Based reactor and Water Based Reactor. Fukushima accident made people doubt about the safety of Nuclear Power Plant (NPP), which impact on the public perception of the safety of nuclear power plants. The paper will describe the assessment of safety and radiation consequences on site for normal operation and Design Basis Accident postulation of SMR based on PWR-100 MWe in Bangka Island. Consequences of radiation for normal operation simulated for 3 units SMR. The source term was generated from an inventory by using ORIGEN-2 software and the consequence of routine calculated by PC-Cream and accident by PC Cosyma. The adopted methodology used was based on site-specific meteorological and spatial data. According to calculation by PC-CREAM 08 computer code, the highest individual dose in site area for adults is 5.34E-02 mSv/y in ESE direction within 1 km distance from stack. The result of calculation is that doses on public for normal operation below 1mSv/y. The calculation result from PC Cosyma, the highest individual dose is 1.92.E+00 mSv in ESE direction within 1km distance from stack. The total collective dose (all pathway) is 3.39E-01 manSv, with dominant supporting from cloud pathway. Results show that there are no evacuation countermeasure will be taken based on the regulation of emergency.

1. Introduction
Indonesia consists of around 17,500 big, medium and small islands, the big islands are: Java, Sumatra, Kalimantan, Sulawesi and Papua. Total area is about 1.9 million square miles including the ocean. Population is around 250 millions people with rate of 1.21% (2015). Based on that condition, Indonesia needs high demand and supply of energy in the future. National economic and energy assessment has shown that nuclear energy would be part of energy sources in national energy mix policy simultaneously with fossil energy source, gas, water and other renewable energy. Due to high demand and supply of energy in the future, thus it has been chosen that SMR technology is much more appropriate for small islands to support their development. Preliminary technology assessment on various SMR have been started, indeed the SMR is grouped into Light Water Reactor, Gas Cooled Reactor, and Solid Cooled Reactor and from its location it is group into Land Based reactor and Water Based Reactor. Based on Technology Assessment, the type of SMR that has a high probability to be
built in Indonesia is PWR (Pressurized Water Reactor) type. The SMRs have been designed as PWR type in the world such as CAREM-25 (100Mwe), ACP-100 (310MWt), FLEXBLUE (530MWt), KLT-40 (150MWt) and SMART-300 (100MWe) [1]. Research on various types of PWR have been done, which are about the reactor design, reactor core, and the reactor [2,3,4].

Fukushima accident made people doubt about the safety of Nuclear Power Plant (NPP), which impact the public perception on the safety of Nuclear Power Plants. Lesson learnt from Fukushima accident is that the safety of nuclear power plants need to be highlighted back to improve the public acceptance of nuclear safety and security in Indonesia. To prove that the operation of nuclear power plant in Indonesia required analysis of radioactive dispersion into the environment under normal conditions and postulated accident. The objective of this paper is to assess of radiation consequences on site for routine operation and DBA (Design Basis Accident) postulation of SMR PWR-100 MWe in Bangka Island. Consequences of radiation for normal operation were simulated for 3 units SMR. The source term was generated from an inventory activities released by using ORIGEN-2 software and the consequence calculated by PC-Cream (routine release) and PC Cosyma (accident release). The adopted methodology used in this research work was based on the predominant site-specific meteorological and spatial data.

2. Methodology

The assessment approach in this work was based on atmospheric dispersion calculation [5-14]. The reactor inventory used ORIGEN2.0 [15], the dose calculation used PC-CREAM 08 for normal operation [5,12] and PC Cosyma for DBA calculation [7-9]. Based on inventory, release assumption each compartment and postulation, the source term would be calculated. Through various pathways, source terms release will get into the human body depending on the type and behaviour of nuclides as well as meteorological and environmental conditions.

2.1. Inventory and Source term Calculation

The series of studies performed on the first core and transition core configurations 2 until the equilibrium core is reached. The initial step is to make an input of the first core configuration with composition consists of 57. Each Fuel Assembly (FA) consists of a cylindrical UO$_2$-Gd$_2$O$_3$ fuel in a 17x17 matrix. Fuel Assembly in the reactor core has 2 levels of uranium enrichment of 2.82% and 4.88%. The FA has average burn-up of 36 MWD/KGHM with the one reactor cycle is 1000 days operation [1].

The source terms were calculated for routine and accident releases. The calculation of source terms was begun with calculation of reactor inventory. After the inventory was obtained, the calculation of source term that release from inventory to environment were conducted. The routine release is assumed that the release of fission products from pinholes reached 0.1%. In addition to the fission products of the porosity of the cladding, the fission products in the primary coolant contamination also comes from impurities natural uranium and enriched uranium on the outer surface of the cladding. Uranium contaminants in surface cladding can reach 10 microns uranium weight [12].

For the accident cases, scenarios were postulated by Large Break Loss of Coolant Accident (LB-LOCA). The source term was calculated based on reactor inventory with the assumption core damaged reaches approximately 33% which is the optimal damaged based on results of the study and an agreement for LB-LOCA accident [9,13-14]. Estimation of source term were calculated by using the following assumptions: 33% cores damaged, the gap release of noble gases; Kr = 7.5% and Xe = 2.15%, for Iodine = 0.65%, other nuclides = 0.0051. For release core inventory: Iodine is 0.22%, Cs-137 is 0.5%, and other nuclides 0.06%. Reduction in containment for nuclide Iodine is 0.46. The efficiency of the filter in the chimney of the reactor was taken to 0% noble gases, iodine (organic) 90%, and other nuclides (Br, Te, Cs, Rb) 99% [7,9].

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2.2. Doses calculations

The mathematical models used to get the generic data sets are atmospheric dispersion using a Gaussian plume model, dry and wet deposition. A Gaussian Modification (Segmented Gaussian) atmospheric dispersion code PC Cream and PC Cosyma were used to estimate the radiological doses of the released radionuclides [5-9]. The segmented Gaussian plume model is accurate on the 0-50km scale. The model requires the following input data and parameters: time and location, atmospheric data (wind speed and wind direction, air temperature, relative humidity, cloud cover, mixing layer height), stability class of the atmosphere (proposed by the model based on meteorological data), roughness length and information about the release (air pollutant, source height). The distribution of Pasquille Guifford atmospheric stability classes as an annual base of the West Bangka site were processed hourly. Radionuclides are released through the ventilation stack of 60 meters height to the environment. The dose were calculated by area within 20 km around the site for nine distances (0.5, 1, 2, 3, 4, 5, 10, 15, 20, 30, 40 and 50 km) and 16 directions/sectors at different wind speed.

3. Result and Discussion

3.1. Inventory and source terms

| Nuclide | Core Inventory (Bq) | Routine source term (Bq/y) | Accident source term (Bq) | Nuclide | Core Inventory (Bq) | Routine source term (Bq/y) | Accident source term (Bq) |
|---------|---------------------|---------------------------|--------------------------|---------|---------------------|---------------------------|--------------------------|
| Kr-85m  | 8.76E+16            | 2.89E+13                  | 1.97E+15                 | Cs-137  | 3.38E+16            | 2.79E+07                  | 1.02E+09                 |
| Kr-88   | 2.45E+17            | 8.09E+13                  | 5.52E+15                 | Te-132  | 4.85E+17            | 1.60E+08                  | 3.71E+10                 |
| Xe-133  | 3.76E+17            | 1.24E+14                  | 2.43E+15                 | Ba-139  | 6.19E+17            | 5.11E+08                  | 4.74E+10                 |
| I-131   | 3.37E+17            | 1.67E+09                  | 6.66E+10                 | Ba-140  | 6.19E+17            | 5.11E+08                  | 4.78E+10                 |
| I-132   | 4.93E+17            | 2.44E+09                  | 9.74E+10                 | Sr-90   | 2.55E+16            | 2.11E+07                  | 1.95E+09                 |
| I-133   | 6.99E+17            | 3.46E+09                  | 1.38E+11                 | Mo-99   | 6.36E+17            | 5.25E+08                  | 4.87E+10                 |
| I-134   | 7.77E+17            | 3.85E+09                  | 1.53E+11                 | Ru-106  | 1.53E+17            | 5.04E+07                  | 1.17E+10                 |
| I-135   | 6.67E+17            | 3.30E+09                  | 1.32E+11                 | Rh-105  | 3.23E+17            | 1.07E+08                  | 2.47E+10                 |
| Rb-88   | 4.80E+14            | 3.96E+05                  | 3.68E+07                 | Y-90    | 2.66E+16            | 2.20E+07                  | 2.04E+09                 |
| Cs-134  | 4.48E+16            | 3.69E+07                  | 3.43E+09                 | Y-91    | 4.51E+17            | 3.72E+08                  | 3.45E+10                 |

Table 1 gives the activity of core inventory, source term routine and accident for SMR-100 MWe with PWR type. Based on inventory data, the source term for routine released and for LB-LOCA accident have been calculated. The result estimation is shown on the same table. The source term calculation in the normal/routine operations are activity for a year. Based on activity, and a half-life nuclide, and toxicity then are selected 20 nuclides as the source term. The routine source term were used as input data for PC CREAM, and accident source term were used for PC Cosyma.

3.2. Routine doses

The results of the mean effective individual dose for adult (all pathway and nuclide) from sums stack of three SMR calculations are given in Fig 1 and Fig.2. The term dose for normal dose used in the paper is therefore the sum of the annual external and internal effective doses to individuals received over 1 y. The individuals doses at each sector were assessed assuming that the individuals occupancy indoors is 80-90%, depend on habit and occupancy of population. Trend of radiation dose in Fig. 1 showed that the individual dose decreases with increasing distance from the stack. The data also show
that the highest doses for all distances is \(3.78 \times 10^{-2}\) mSv/y within 1km distance from all stacks. The dominant nuclide is Kr-88 for Gamma from Plume pathway and Rb-88 for Beta from Ground pathway.

Figure 2 show the mean effective dose for adult (all pathway and nuclide) from sums stack of three SMR calculations by wind direction or sectors. The wind directions divided into 16 sectors. According to calculation by PC-CREAM 08 computer code, the highest mean effective individual dose in terrestrial for adults is \(5.34 \times 10^{-2}\) mSv/y in ESE direction within 1 km distance from stack. (Fig.1). It can also be concluded that the estimated effective doses are lower than the dose constraint of 0.3 mSv/y (BAPETEN, 2014) associated with this plant.

### 3.3. Accident Doses
The dose from accident was calculated by LB LOCA accident postulation. Based on accident source term on Table 1, dose was calculated by PC. Cosyma. The calculation result is shown on Fig. 3 and Fig.4.

Mean effective individual dose for accident by distance is shown on Fig 3, and Mean effective individual dose for accident by wind direction is shown on Fig 4. The dose was calculated by 20 source term nuclide which were dispersed in atmosphere and deposed on ground surface. The
dominant dose from air is supported by noble gases Xe-133 and Kr-88. Nuclides of I-131, Sr-90, and Te-132 support dose from ground surface.

Fig. 4. Mean effective individual dose for accident by wind direction (sector)

Fig. 3 and Fig. 4 show the highest individuals effective dose for LBLOCA is 1.92E+00 mSv in ESE direction within 1km distance from stack. Results show that there are no evacuation countermeasure will be taken based on the regulation of emergency. Evacuation shall be taken if the dose have accepted by public is \( \geq 50 \) mSv (BAPETEN). From Fig. 3 that the individual dose decreases with increasing distance from the stack.

Collective dose is based on individual dose and population. Fig. 5 shows on the collective dose for LBLOCA accident by distance. The total collective dose (all pathway) is 3.39E-01 manSv, with dominant supporting from cloud pathway. The results show that for LBLOCA postulated accident at 100 PWR-MWe, it does not lead to evacuation countermeasures on public in site. The assessment
have been done to prove that the construction of SMR is safe and meets the safety standard for normal operation and the impact in DBA conditions.

4. Conclusion
The result of analysis radiation safety for SMR with PWR-100 MWe: the highest individuals dose for routine discharge of normal operation for adults is 5.34E-02 mSv/y in ESE direction within 1 km distance from stack. It can also be concluded that the estimated effective doses are lower than the dose constraint of 0.3 mSv/y associated with this plant (BAPETEN, 2014). The highest individual effective dose for LBLOCA is 1.92.E+00 mSv in ESE direction within 1km distance from stack. The total collective dose (all pathway) is 3.39E-01 manSv, with dominant supporting from cloud pathway. Results show that there are no evacuation countermeasure will be taken based on the regulation of emergency. The assessment have been done to prove that the construction of SMR -100MWe in site is safe and meets the safety standard for normal operation and the impact in DBA conditions.

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