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Source Term Assessment for 100 MWe Pressurized Water Reactor

Pande Made Udiyani* **and M. Budi Setiawan**

Center for Nuclear Reactor Technology and Safety, National Nuclear Energy Agency, Bldg. 80 Puspiptek Area, Setu, South Tangerang, 15310, Indonesia

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A B S T R A C T

One of the barriers on the implementation of nuclear energy in Indonesia is public perception towards the safety of nuclear power plants (NPPs). Therefore, it is necessary to perform a study about the radiation impact of normal and abnormal operations of an NPP. In accordance to the program of Ministry of Research and Technology period 2020-2024, concerning the plan to build a small modular reactor (SMR)-type NPP, a radiation safety study has been performed for the 100 MWe Pressurized Water Reactor (PWR-100MWe). Source term release of radioactive substances into the environment from PWR-100MWe is a starting point in the study of the radiological consequences of reactor operation. Therefore, this paper will examine the PWR-100MWe source term under normal and abnormal operating conditions, according to the design and the design basis accident (DBA). The initial trigger of the DBA is Lost of Coolant Accident (LOCA) such as Small LOCA and Large LOCA. Due to the limitations of available SMR data, the study of PWR-100MWe source term refers to the assumption of the release fraction of fission products per subsystem in a larger 1000MWe PWR. It is expected from this assumption that pessimistic source term will be obtained. The study begins with calculation of PWR-100MWe core inventory using ORIGEN2 code based on PWR-100MWe reactor parameters. Through the mechanistic source term model and PWR-1000MWe release parameters, source terms will be obtained for normal operation and abnormal conditions i.e. DBA. Normal source term is used to calculate the consequences of normal operation, which will be used for environmental monitoring and environmental safety analysis of the site. Whereas accident source term is the basis for calculating the radiological consequences of accidents used for SAR documents and nuclear preparedness.

1. INTRODUCTION

Based on the directions for policy and the National Research Priority with the focus on energy sector during 2020-2024 period [1], PTKRN as a center of technology and safety of the nuclear reactor conducted a study of technology and safety of SMR reactor, which is planned to be built in Indonesia. In this period, it is expected that we will obtain the results of the studies on SMR radiation safety analysis. Radiation safety documents for normal conditions and postulated accidents are needed to support the licensing, operating, commissioning and decommisioning requirements.

Based on the reference design PWR-100MWe, radiation safety of normal conditions and accidents can be estimated, as well as ensuring that radiation exposure in the installations and communities to be as low as reasonably achievable (ALARA), below

* Corresponding author. Tel./Fax.: +62-8128958248
E-mail: pmade-u@batan.go.id
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the regulation limit. As the first step for calculation of radiological consequences at normal operations and postulated accidents from PWR-100MWe, source term data is needed.

Calculations on source term of normal operations for PWR-1000MWe, as well as SMR-type PWR and HTGR, have been carried out using similar methods in several studies [2-4], along with the environmental consequences [3-6]. Calculations for postulated accident conditions, both Design Basis Accident (DBA) and Beyond Design Basis Accident (BDBA), for PWR-1000MWe using the NUREG and EUR references have also been obtained [7-11]. Source term calculations using a combination of experimental, deterministic, and probabilistic processes have also been carried out [12-15]. After Fukushima Daiichi accident, pessimistic source term was preferred to estimate radiological consequences in the site environment [16-19]. Source term is used for estimation of consequences [20-24], as it is more precise in determining the nuclear emergency zone as evidenced from the research of Udiyani et al [25]. Source term calculations for large nuclear power plants have been carried out because the fleets are already existing. As for small and medium-power nuclear power plants, there is not as much development as larger plants. Especially for SMR, there is much less data available for source term.

This study aims to examine the source term generated in the operation of an SMR under normal operations and accident conditions. Due to the data limitations, the study of PWR-100MWe source term refers to the assumption of the fission products release fraction of PWR-1000MWe. Estimation is done by calculating and analyzing the release of nuclides for each safety sub-system during normal operations or accidents that cause a significant radiation impact to human and the environment. For accident conditions, postulations are made for Small LOCA and Large LOCA accidents. The release of fission products to the sub-system for normal operation includes core inventory, cooling systems, clean-up filters, steam generators through volume control systems, hoods, external gas systems, to and from the chimney through various filters. Whereas for accidents, it includes the inventory of core, gaps, cladding, cooling systems, hoods, and chimneys.

2. THEORY

Fuel failure fraction

The magnitude of the fuel failure fraction is based on the postulation of core damage and reactor safety systems such as ECCS (Emergency Core Cooling System). For DBA, it is based on the results of experiments in EU countries, ranging from 1-33% depending on postulation and safety features on PWR-100MWe [8,18].

Fission Product (FP) release fraction from gaps

The release of nuclides from the inventory gap through two phases are the dry phase that occurs when the core is filled with steam, and the wet phase when the core is re-filled with water from ECCS. For noble gaseous nuclides, all are released in the dry phase, while for other nuclides, 10% are released in dry phase, and the rest in wet phase [13].

FP released fraction of fuel

The fraction of FP releases from fuel can be calculated using (1) [13].

$$F = F_0 \lambda^{bf} \qquad (1)$$

where,

λ : nuclide decay constant

F_0 : noble gas: $4.0\text{x}10^{-2}$; volatile gases: $3.8\text{x}10^{-2}$

bf : noble gas: - 0.29 ; Volatile gas: - 0.17

Transfer of fission products from the core to the cooling system

The detachment rate of fission products is affected by radioactivity in the inventory, the fraction of fuel failure, and the release rate coefficient. Activation products are affected by corrosion rate as shown in (2) [13].

$$C_i(t) = \frac{P_{i,n}}{\lambda_i WP}\{1 - \exp(-\lambda_i t)\} + C_i(t_{n-1})\exp(-\lambda_i t) \qquad (2)$$

where,

C_i : nuclide activity in reactor coolant, Ci/kg

P_i : the rate of release of nuclei i from the core to the reactor cooler, Ci/s

WP: reactor cooling capacity, kg

t : time, dt

λ_i : the nuclear transformation constant of nuclide i, dt^{-1}

Transfer of nuclides from RCS to primary containment

The transfer of nuclides from the Reactor Cooling System (RCS) to the primary hood is influenced by the operation of the safety features of the Emergency Core Cooling System (ECCS) through the two phases of its operation, namely the dry and wet phases [13,19].

Retention in the containment room

The concentration of fission products (noble gases, halogens, volatiles, etc.) carried to the hood depends on the depletion mechanism, design of safety features such as spray systems, and the rate of release from the process of fuel damage. Besides this, the reduction in radioactivity occurs by natural mechanisms such as agglomeration, condensation, gravity, deposition, and adsorption [13,19].

3. METHODOLOGY

3.1. The PWR-100MWe core is calculated using ORIGEN-2 computer code, based on the reactor data of the PWR-100MWe SMR [4].

3.2. Calculation of the characteristics of the PWR-100MWe reactor for each safety sub-system under normal operating conditions uses the assumption for discharge radionuclides from the fuel sub-system due to the presence of cladding pinholes will result in about 0.1% of radionuclides release from inventory to the reactor coolant. Release factor from the core to the reactor coolant for iodine are 0.3-0.5%, Cs and Rb are 0.25-0.3%, Ru and Te are 0.01%, and other nuclides are 0.25%. Filter efficiency in the cleaning system and Chemical and Volume Control System (CVCS) for iodine is 95%, other fission products are 90%, and noble gases are 0%. Filter efficiency in the reactor chimney taken for noble gases are 0%, iodine (organic) is 90%, and other nuclides (Br, Te, Cs, Rb) are 99% [4,13,25]. The normal operating source term calculation mechanism is shown in Fig. 1.

3.3. Calculation of the characteristics of the PWR-100MWe in Small LOCA accident postulation was using the assumptions that core failure is 3%. Gap release for noble gases of Kr is 7.5% and Xe is 2.15%. Gap release for I is 0.65%, while other nuclides are 0.0051. Release from core inventory for I is 0.02%, Cs-137 is 0.06%, and others nuclides are 0.06%. Reduction in containment for nuclides I is 0.05%. Filter efficiency in the reactor chimney taken for noble gas is 0%, iodine (organic) is 90%; and other nuclides (Br, Te, Cs, Rb) are 99% [4, 13,25].

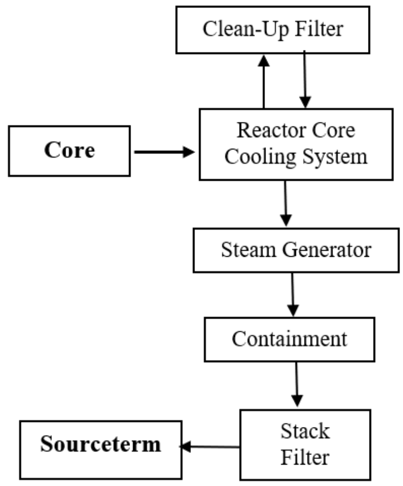

Fig. 1. Mechanism Transport of source term at normal operation of PWR-100MWe

3. 4. Calculation of PWR-100MWe source term for the Large Break LOCA using the following assumptions: Core failure is 33%, gap release for noble gases are 7.5% for Kr and 2.15% for Xe. Gap release for I is 0.65% and other nuclides are 0.0051%. Release core inventory for I is 0.22%, Cs-137 is 0.5%, and others nuclides are 0.06%. Reduction in containment for nuclide I is 0.46%. Filter efficiency in the reactor chimney taken for noble gases are 0%, iodine (organic) is 90%, and other nuclides (Br, Te, Cs, Rb) are`99% [4,13,25].

The source term calculation mechanism in LOCA accident conditions is shown in Fig. 2.

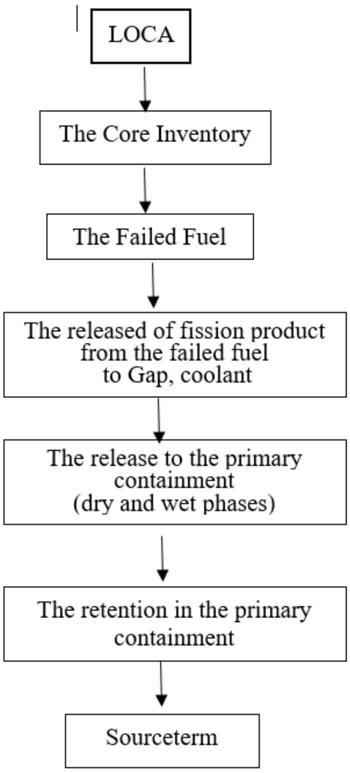

Fig. 2. Transport mechanism of source term at LOCA of PWR-100MWe

4. RESULTS AND DISCUSSION

Source term for Normal Operation

The radionuclides in the reactor core are classified into 8 groups, which are noble gases group (Xe, Kr), halogen (I), alkali metal (Cs, Rb), tellurium group (Te, Sb, Sc), barium-strontium group (Ba, Sr), noble metals (Ru, Rh, Pd, Mo, Tc, Co), lanthanides group (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am), and cerium group (Ce, Pu , Np).

The fission products in the fuel inventory are influenced by the type of fuel, fuel storage level, uranium loading, fuel fraction, reactor power, core configuration, and fuel irradiation time in the core. These parameters will affect the amount of fission product activity confined in the fuel cladding. The fission products in fuel are noble gas nuclides such as Xe and Kr, as well as halogen such as I, Cs, Ru, Ce, La, Ba, both volatile and non volatile. Due to the diffusion process in fuel pellets, the fission products will pass to the fuel gap with the release rate which is affected by the diffusion coefficient, temperature, and fuel fraction. The presence of porosity in the fuel cladding that has been irradiated on the core for more than 3 years has resulted in the formation of pinholes which make fission products in the fuel gap escape from the cladding to the primary coolant. The release of fission products from pinholes reaches 0.1-1%. The rate at which fission products are released to the primary coolant is affected by activity in the fuel gap, pinholes diameter, and fuel fraction. In addition to fission products from cladding porosity, fission products inside the primary coolant also come from natural uranium contamination and enriched uranium on the outer surface of the cladding. Uranium contaminants on the surface of the cladding can reach 10 microns by weight of uranium [3-5]

Radioactivity for fission products and activation products resulting from the operation of PWR-100MWe for normal operating conditions is shown in Fig. 3. It contains the results of the calculation of the radiation activity of the components and sub-systems of the PWR-100MWe. This is by using the assumption that there is a release of fission products from uranium contaminants found in the fuel surface due to fabrication defect. Another assumption is that there are fission products that escape the pinhole cladding through diffusion. Fission products pass to the RCS and through the reactor safety features, released to the system or components in very small quantities with insignificant activity to become radiation exposure that exceeds the exposure limit of the regulatory body.

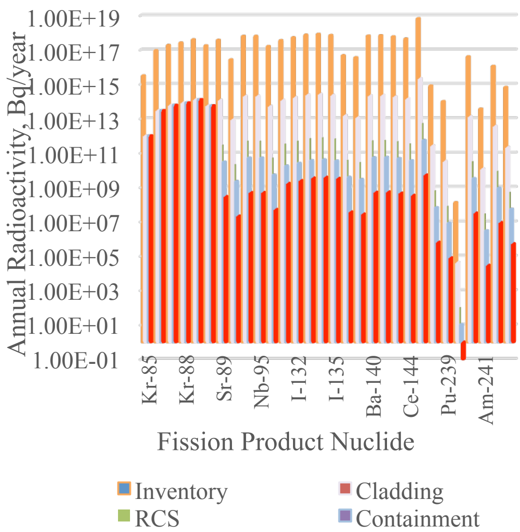

Fig. 3. Radiation activity of the components and sub-systems on the PWR-100MWe at normal operation

The estimated results with the aforementioned assumptions acquired for source term activity of PWR-100MWe on normal operation are as follow: Xe-133 activity 1 is 1.24E+14 Bq/year (the reference of OPR-1000 of is 1.67E+14 Bq/year and AP1000 is 1.28E+14 Bq/year [5]). Other important nuclide such as I-131 is 1.67E+08 Bq/year (reference 2.85E+09 Bq/year from OPR-1000, and 2.94E+09 Bq/year from AP1000) [5]. Source term activity for Cs-137 is 2.79E+07

Bq/year (reference 4.81 E+07 Bq/year from OPR-1000 and 1.44E 07 Bq/year from AP 1000 [5]).

Source term for DBA

Fission product activities release from the inventory through fuel cladding cracks is caused by conditions that allow the fuel to fail. As for accident conditions, which allows the fuel to fail, are because the LOCA results in core cooling and cladding integrity cannot be maintained. The condition of nuclides migration from inventory to the hood is based on the postulation of core damage and the reactor safety system such as the ECCS. If ECCS still functions, accidents that occur are still at the level of the basic DBA, such as Small Break LOCA and Large Break LOCA.

Small LOCA

Radioactivities for fission products and activations products resulting from the operation of the PWR-100MWe for the Small LOCA accident conditions are shown in Fig. 4. It contains the calculation results of the radiation activity of the existing components and sub-systems of the PWR-100MWe. Using the assumption that fission products release occurs from the damaged core so that they pass into the cooling system through the reactor safety features off to other systems leading to the containment.

The fission product in the DBA source term (Small LOCA or Large LOCA) is divided into 3 main classes, they are noble gases (Xe and Kr), volatiles (I, Cs, Ru), and other non-noble and volatile gases (Ba, Ce). Source term activity of Small LOCA for Xe-133 is 2.43E+14Bq, I-131 is 1.32E+09 Bq and 3.11E+07 Bq for Cs-137.

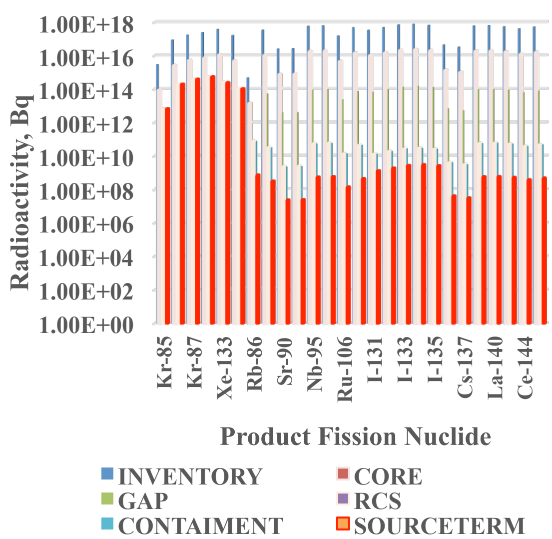

Fig. 4. Radiation activity of the components and sub-systems on the PWR-100MWe at Small Break LOCA

Large Break LOCA

Radioactivities for fission products and activation products resulting from Large Break LOCA accident of the PWR-100MWe are shown in Fig. 5. The assumption is the same with the previous, that there is a release of fission products from the damaged core, so that fission products pass into the cooling system and through the safety features to other systems, and then leads to containment.

The composition of the radionuclides released to the primary coolant is influenced by the characteristic (number, type, and properties) of radionuclides, primary cooling physics conditions, interactions with cladding and or structural components, burn-up, material morphology, release fraction, the barrier, and the assumption of release time. It is also influenced by temperature, pressure, and retention time in the RCS during the degradation of the core [13]. The high pressure in the RCS will lead to a longer time for aerosol fission products to be removed from the core, and will reduce the number of short-lived fission products release to the containment [13,19]. This affects the number of fission products that released from the containment, if the pressure in the RCS is low. The rate of fission products release from the fuel is influenced by the phenomenon of fuel melting progress, such as the heating rate of the core, the location, the width of melting area, the oxidation location in Zircaloy cladding, the pace and location of accumulated melted fuel, shape of the melting area, the thickness of the ceramic covering the melting pot, timing and location of the crust and melting, the time of the disconnection of the core is entered into the lower areas, and the interaction with water.

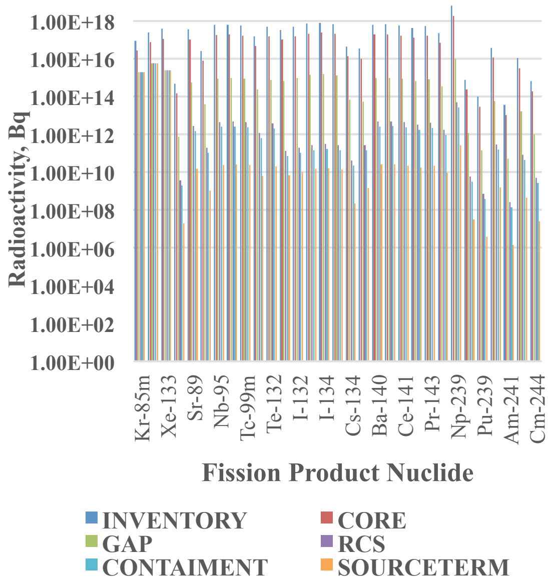

Fig. 5. Radiation activity of the components and sub-systems on the PWR-100MWe at Large Break LOCA

The concentration of fission products (noble gases, halogens, volatiles, etc.) carried to the hood depends on the following: depletion mechanism, design of safety features such as spray systems, and the rate of release from the process of fuel damage. These are apart from radioactivity reduction in natural mechanism, such as agglomeration, condensation, gravity, deposition, and adsorption. For nuclides that have non-volatile, heavy metal, and water-soluble properties, this natural mechanism significantly reduces the release of nuclides from the chimney. The deposition of the nuclide, apart from adsorption of the containment wall, is also aided by the effects of gravity, especially for non-volatile heavy metals. The mechanism of reduction of iodine in air containment, specifically for organic iodine, can reach 0.4 to 0.5. Spray mechanism also has a function to reduce the activity of nuclides, by making the nuclide not separated out of the containment. This spray system generally works when accident including BDBA or severe accident occurs.

The estimated results for source term activity of PWR-100MWe on Large LOCA are 2.43E+15 Bq for Xe-133 (the reference AP1000 is 1.16 E+16 Bq [13]). Source term activity for I-131 is 7.10E+09 Bq (reference 4.2E+10 Bq from AP1000) and activity for Cs-137 is 1.40E+09 Bq (reference 5.50E+09 Bq from AP 1000 [13]).

5. CONCLUSION

Radionuclide characteristics for each safety subsystem in the PWR-100MWe are influenced by reactor operating conditions (normal or accident), safety features, and radionuclide inventory in the reactor core. Radionuclide activity calculation data is obtained for each sub-system, are the release of radionuclides to the sub-system to the reactor cooler from the reactor core, radionuclide release from the reactor cooling system, radionuclide release to containment, and radionuclide release to environment.

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