Source Term Analysis of Long-Life 620 MWt BWR During Loss of Coolant Accident (LOCA)

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Abstract

Source term analysis on a long-life 620 MWt BWR has been performed. The study includes fuel cell calculations, multi-groups diffusion equation calculation with 8 groups of energy, source term calculation and dose calculation. The scope of the source term analysis includes fission product inventories calculation in core, their activity and their toxicity during the Loss of Coolant Accident (LOCA). The calculation based on the assumption that the reactor has been operating for 10950 days or 30 years with full power of 620 MWt. The total activity of the core is $1,12 \times 10^9$ Ci, and the higher inventory of the core is dominated by the trans-uranium elements. The release of fission product activity to the atmosphere is $1,11 \times 10^7$ Ci. And the toxicity analysis shows that noble gas has the higher toxicity degree. Whole body dose calculation result in exclusion zone is 7.17×10^{-04} rem and the thyroid dose calculation result in this zone is 7.21×10^{-02} rem. And for the whole body dose calculation result in low population zone is 2.69×10^{-05} rem and the thyroid dose calculation result in this zone resulted 1.80×10^{-03} rem.

Keywords: Long-life BWR, LOCA, Source term, Activity, toxicity, Whole body dose, Thyroid dose, Exclusion and low population zone

1. Introduction

The increase of human population in the world especially in Indonesia causes the increase of demand. Nuclear energy could become one of the promising alternative energy sources for near future in Indonesia. The application of long life reactor in Indonesia is very suitable especially for remote area. Because of Indonesia geographical condition which is archipelago. Long life reactor can solve transportations problem of fuels during electric power generation. Long life reactors have high utilization so that economical value can be improved.

This research has been done to Long Life BWR 620 MWt¹⁾ with cylinder core and the operation period of this reactor is 30 years.

2. Calculation Method

The scope of this study includes fuel cell calculations, multigroup diffusion equation calculation with 8 group constants, source term calculation and dose calculation. And the source term analysis includes fission product inventories calculation in core, their activity and their toxicity during the Loss of Coolant Accident (LOCA).

For fuel cell calculation we use SRAC (Standard Reactor Analysis Code) code system. The Fuel geometry for the present study is described in Figure 1. Here we use hexagonal cell form to obtain low moderation environment.

Figure 1. Cell Structure Geometry¹⁾

And detail of cell configuration is listed in Table 1.

Table 1. Fuel element specifications¹⁾

Percentage fuel pellet	60%
Percentage Moderator (Air)	20%
Percentage Cladding (Zircaloy)	20%
Percentage Void axial	0 – 40 %
Fuel element compositions	(Th-U-Pa)N
²³³ U Enrichment Range	8 – 11%
Composition Range of ²³¹ Pa	6 – 11%
Pitch	1.62 cm

The Cylindrical core was divided into 3 radial region and 6 axial region. The first radial region was inner fuel with 8% of U-233 enrichment, and the second radial region was 11% of U-233 enrichment, and the last region was reflector. We used water as the reflector. Upper and bottom of the axial region were reflector, and four others axial region were 0% void,

10% void, 25% void and 40% void. The detail of core composition is described in Figure 2.

reflector	reflector	reflector	32,4 cm
8% U-233 Enrichment 40% Void	11% U-233 Enrichment 40% Void	reflector	63,18 cm
8% U-233 Enrichment 25% Void	11% U-233 Enrichment 25% Void	reflector	63,18 cm
8% U-233 Enrichment 10% Void	11% U-233 Enrichment 10% Void	reflector	63,18 cm
8% U-233 Enrichment 0% Void	11% U-233 Enrichment 0% Void	reflector	63,18 cm
reflector	reflector	reflector	32,4 cm
74,52 cm	74,52 cm	32,4 cm	

Figure 2. Core Material Composition

Figure 3. Form and dimensions of the first core¹⁾

To calculate the neutron flux distribution we use the multigroup equation²⁻³⁾ which written below.

$$\frac{1}{v_g} \frac{\partial \phi_g(\vec{r}, t)}{\partial t} - \nabla \cdot D_g(\vec{r}) \nabla \phi_g(\vec{r}, t) + \sum_{i \in g} (\vec{r}) \phi_i(\vec{r}, t) = \sum_{g'=1}^G \sum_{s \in g'} (\vec{r}) \phi_{s'}(\vec{r}, t) + \frac{\chi_g}{k_{eff}} \sum_{g'=1}^G v_{g'} \sum_{s \in g'} (\vec{r}) \phi_{s'}(\vec{r}, t)$$

For the source term analysis, we assume that the accident was occurred because of the pipe breaking and the source term might release to the environment. There is no damage in containment but the safety system, such as ECCS, cavity and ventilating filter, works specifics to the radionuclide types. Noble gas was assumed that 100% released to the environment, 50% for the halogens released to the environment and 1% of the rest radionuclide released to environment. Table 2 shows the radionuclide groups.

Table 2. Radionuclide Groups⁴⁾

Group	Elements
Noble Gases	Xe, Kr
Halogens	I, Br
Alkali Metals	Cs, Rb
Tellurium Group	Te, Sb, Se, Ba, Sr
Noble Metals	Ru, Rh, Pd, Mo, Tc, Co
Lanthanides	La, Zr, Nd, Eu, Nb, Pm, Pr Sm, Y, Cm, Am
Cerium	Ce, Pu, Np

The activity inside the containment after the accident can be written as:

$$c_0 = 8.46 \times 10^5 F_p F_b P \gamma_i (1 - e^{-\lambda_i t}) c_i$$

The degree of toxic in the human body due to the amount of radioactive energy called toxicity. It is weighting by dose constrain which can be written as:

$$\text{Toxicity} = \text{Activity} / \text{Dose Limit}$$

The value of dose limit for each radionuclide is refers to annual limit intake for inhalation. The data is collected from 10 CFR 20 US-NRC Appendix B⁵⁾

The whole body and thyroid dose from gamma rays using the following equation³⁾

$$H = \frac{0.262 \bar{E}_y \lambda_i}{\pi \bar{V} \sigma_y \sigma_z \lambda_c} (1 - e^{-\lambda_c t_0}) \text{rem}$$

for whole body

$$H = \frac{592 b \xi q \bar{E}_y \lambda_i C_0}{\pi \bar{V} \sigma_y \sigma_z M \lambda_c \lambda_e} (1 - e^{-\lambda_c t_0}) \text{rem}$$

for thyroid.

The parametric used in this study are: the breathing rate R is 0.000347 m/s, $\sigma_y = 30$ meters, $\sigma_z = 11$ meters (in exclusion area zone) and R = 0.000231 m/s, $\sigma_y = 220$ meters, $\sigma_z = 40$ meters (in low population zone).

3. Calculation Result and Discussion

3.1 Neutron flux distribution

Figure 4 and 5 describes two dimension neutron flux distribution, radial and axial region. Neutron flux decrease in the higher region enrichment because the fissile material was absorbs this neutron to fission. The flux neutron increases in the reflector region because this region contains material that reflects the neutron.

Figure 4. neutron flux distribution of each group in radial region

Figure 5. Neutron flux distribution of each group in axial region

3.2 Source term analysis

To analyze the source term in reactor core, we need to calculate the inventory and the activity of each radionuclide. To calculate the transuranium elements inventory we use SRAC output data, and for the fission product we use data from JNDC (Japan Nuclear Data Center). Figure 6 to 9 are the results of the source term calculation. Figure 6 describes the top 30 radionuclide inventory inside the core in 10950 days or 30 years, most of it is transuranium elements and the rest are fission product elements.

The total activity in the core is 1.12E+09 Ci. The top 30 activity is dominated by fission product elements. And the highest activity is I-134 with 1.759E+07 Ci.

The activity inside the containment in 2 hours after the accident describes in figure 8. The figure shows the top 30 radionuclide and the noble gas has the highest activity inside the containment.

Krypton-88 was the radionuclide with the biggest toxicity degree.

Figure 6. Radionuclide inventory inside the reactor core (Top 30)

Figure 7. Radionuclide activity inside the reactor core (Top 30)

Figure 8. Radionuclide activity inside the containment (Top 30)

Figure 9. Source term toxicity degree (Top 30)

The clearance time of each radionuclide has been calculated. The result shows that Plutonium 239, Neptunium 236 and Neptunium 237 have long clearance time.

Figure 10. Clearance time for Pu-239, Np-236 and Np-237

The dose calculation results describes in Table 3.

Table 3. the total dose exposure

Zone	Body organ	Dose (rem)	Dose Limit (rem) ⁶⁾
Exclusion Area Zone	Thyroid	7.21E-02	≤1
	Whole body	7.17E-04	1.25
Low Population Zone	Thyroid	1.80E-03	<1
	Whole body	2.63E-04	<1

4. Conclusion

From above discussion we can conclude that the calculation based on the assumption that the reactor has been operating for 10950 days or 30 years with full power of 620 MWt. The total activity of the core is 1.12×10^9 Ci, and the higher inventory of the core is dominated by the trans-uranium element. The release of fission product activity to the atmosphere is

1.11×10^7 Ci. And the toxicity analysis shows that noble gas has higher toxicity degree than the other elements. Whole body dose calculation result in exclusion zone is 7.17×10^{-04} rem and the thyroid dose calculation result in this zone is 7.21×10^{-02} rem. And for the whole body dose calculation result in low population zone is 2.69×10^{-05} rem and the thyroid dose calculation result in this zone resulted 1.80×10^{-03} rem.

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