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# Effect of Concrete Mixture Design on Radiation in Biological Shielding Wall

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## 1. Introduction

Concrete is considered to be composed of coarse materials that is embedded in a hard matrix of material filling the space between the aggregate particles in the mixture. It has been used for different purposes including support for buildings, for shielding purposes in nuclear installations, and in the construction of NPP containment [1]. However, although the effect of nuclear radiation on the behavior of concrete mechanical properties was demonstrated in recent times, little attention has been paid to the effect of concrete mixture design on the degradation of biological shield along its depth. In this paper, the biological shielding wall in APR1400 Pressurized Water Reactor (PWR) was considered on irradiation, specifically considering several types of concrete mixture design.

## 2. Concrete Mixture Design

Concrete mix types incorporating three types of cementitious materials were investigated for radiation in biological shielding wall: Ordinary Portland Cement (OPC), Flyash (FA), and Slag (GGBS). The proportion of the mix types analysed is based on the Korean standard used in the construction of ShinKori 3 & 4 units with the following modification:

Concrete mix 1: 100% OPC
Concrete mix 2: 80% OPC + 20% FA
Concrete mix 3: 60% OPC+ 40% GGBS

The composition of the three concrete mix is summarized in Table I. All mix types have the same amount of coarse and fine aggregates. 20% and 40% of OPC were partially replaced with FA in Mix 2 and slag in Mix 3 respectively. Table II shows the chemical composition of each cementitious material in the mix. The compositions given in Tables I and II was analyzed carefully to determine the percentage and mass compositions of each element in the mix which was then used as input in the simulation for radiation analysis.

Table II: Chemical Properties of FA and GGBS (%)

| Chemical composition | FA | GGBS |
|---|---|---|
| Specific gravity | 2.43 | 2.79 |
| $SiO_2$ | 63.5 | 34.4 |
| $Al_2O_3$ | 11.1 | 9.0 |
| $Fe_2O_3$ | 5.2 | 2.58 |
| CaO | 14.7 | 44.8 |
| MgO | 1.98 | 4.43 |
| $SO_3$ | 0.35 | 2.26 |
| $Na_2O$ | 0.48 | 0.62 |
| $K_2O$ | 0.4 | 0.5 |
| LOI | 2.1 | 1.32 |

## 3. Neutron Distribution Analysis

The neutron fluence for the APR1400 Reactor Pressure Vessel was calculated using the DORT code and BUGLE cross section library as contained in the journal published by Korea Hydro and Nuclear Power Company (KHNP) [2]. The results of the fluence calculation at different radial positions in the RPV as well as the flux values is presented in Table III. The values represents the peak values of the fast neutron flux (E > 1 MeV) over 60 years of operation with plant capacity of 93% with a total uncertainty of 20% [3].

Table I: Concrete mixture design

| Concrete | Water | Cement | GGBS | FA | crushed Sand | Sand | Total(g/cm3) |
|---|---|---|---|---|---|---|---|
| MIX 1 | 0.16 | 0.402 | 0 | 0 | 1.043 | 0.684 | 2.289 |
| MIX 2 | 0.16 | 0.32 | 0 | 0.082 | 1.043 | 0.684 | 2.289 |
| MIX 3 | 0.16 | 0.241 | 0.161 | 0 | 1.043 | 0.684 | 2.289 |

Table III: Neutron fluence for APR1400 RPV

| thickness(cm) | Fluence (E>1MeV, 55.8 EFPY) | Flux |
|---|---|---|
| 0 | $9.5 \times 10^{19}$ | $5.4 \times 10^{10}$ |
| 5.7531 | $5.1 \times 10^{19}$ | $2.9 \times 10^{10}$ |
| 17.2593 | $1.0 \times 10^{19}$ | $5.68 \times 10^{9}$ |

In order to evaluate the flux value at outer surface of RPV, fluence and flux distribution was evaluated through a regression analysis which was expressed as following:

$$\varphi = \varphi_0 e^{-\Sigma_t x} \qquad (1)$$

where $\varphi$ is the attenuated flux at a thickness, x, $\varphi_0$ is the initial value of flux before attenuation, $\Sigma_t$ is the removal cross section of the material.

Table IV and Fig. 1 shows the flux values evaluated from the regression model, which is used to evaluate flux at the outer surface of the RPV.

Table IV: Regression analysis of flux for APR1400

| Radius (cm) | Flux (Regression) | Flux(calculated) |
|---|---|---|
| 231.775 Inside surface | $6.00 \times 10^{10}$ | $5.4 \times 10^{10}$ |
| 237.528 ¼ thickness location | $2.82 \times 10^{10}$ | $2.9 \times 10^{10}$ |
| 249.034 ¾ thickness location | $6.30 \times 10^{9}$ | $5.68 \times 10^{9}$ |

Using the expression in (1), the flux level at the surface of bio-shielding concrete wall was evaluated as it was assumed to the flux level at the outer surface of RPV. Then, the distribution of flux levels along the depth in the bio-shielding concrete wall was evaluated with consideration of the removal cross section for concrete. Table V and Fig. 2 shows the distribution of the flux along the depth in bio-shielding concrete wall.

Table V: Flux distribution in concrete

| Concrete depth (cm) | Flux level |
|---|---|
| 0.000 | 2.94E+09 |
| 7.505 | 1.49E+09 |
| 15.01 | 7.62E+08 |
| 22.515 | 3.91E+08 |
| 30.02 | 2.00E+08 |

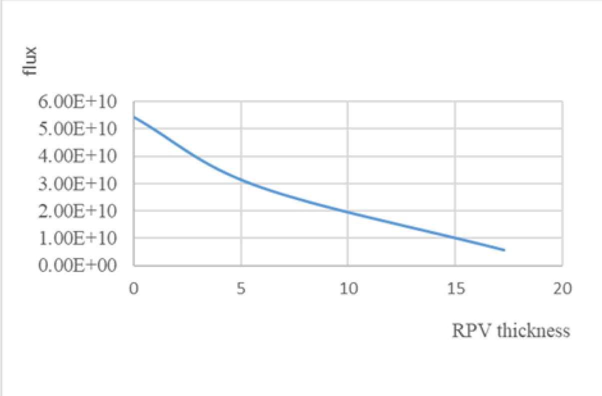


Fig. 1. Neutron attenuation in RPV

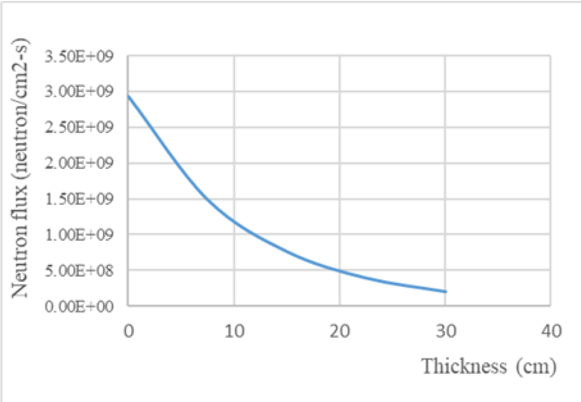


Fig. 2 Neutron distribution in concrete

## 4. Concrete Irradiation Analysis

To simulate concrete irradiation in the bio-shielding wall, it was assumed that APR1400 was operated for 60 years and a cooling down time of 10 years. The irradiation calculation was done with ORIGEN computer code, a part of the SCALE 6.21 computer program developed at Oak Ridge National Laboratory ORNL), USA. It contains the ORIGEN-ARP that performs point depletion calculations [4]. The activation produced after irradiating each concrete mix was analyzed to determine the effect of concrete mixture on radiation. Then, the waste level was analyzed through comparing the activity of radionuclides after 10 years cooling period. With the value of the concentration, the waste level can be evaluated with the following equation:

$$\sum \frac{Ci}{Cl} < 1 \qquad (2)$$

where $Ci$ = concentration obtained from the ORIGEN simulation in Bq/g after 10 years cooling period and $Cl$= concentration of radionuclides given by the regulatory body in Bq/g.

For the three concrete mixture design, the waste level analysis results are presented in Table VI and Fig. 3 along distribution of flux level.

Table VI: Concentration ratio for concrete mix

| flux | OPC | FLYASH | GGBS |
|---|---|---|---|
| 2.94E+09 | 2.72E+00 | 2.97E+00 | 2.72E+00 |
| 1.49E+09 | 1.38E+00 | 1.50E+00 | 1.38E+00 |
| 7.62E+08 | 7.06E-01 | 7.13E-01 | 7.02E-01 |
| 3.91E+08 | 3.61E-01 | 3.94E-01 | 3.61E-01 |
| 2.00E+08 | 1.85E-01 | 2.01E-01 | 1.85E-01 |

As presented in the table and figure, at a flux value of 2.94E+09, the concentration ratio was 2.72 for the OPC and GGBS mix type while 2.97 was obtained for the flyash mix. As can be seen in the table and figure, GGBS and OPC showed a better attenuation accordingly, specifically flux level is relatively higher.

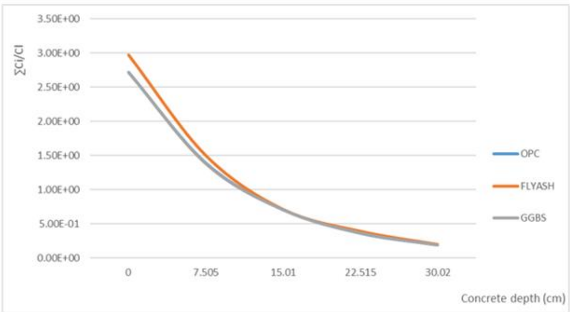


Fig. 3. Concentration ratio against concrete thickness

## 5. Conclusion

The activation products obtained after a 60 years operation and 10 years cooling of the concrete biological shield for three different concrete mix, FA, GGBS, and OPC for the APR1400 PWR was analyzed in this project. The result showed that the concrete mixture design with flyash showed higher concentration. Through further study, the results in this paper can be useful for decision on how much of concrete should be decontaminated during decommissioning.

## 6. Reference

[1] Eyup Zola, Cagatay Ipbuker, Alex Biland, Radiation Shielding of High performance reinforced concrete with basalt fibres, University of Tartu, Institute of Physics, Estonia, 2016.

[2] KEPCO, KHNP, Neutron Fluence Calculation Methodology for APR1400 RPV, APR1400-Z-A-NR-14015-NP, Rev0, Nov.2014.

[3] KEPCO, KHNP, Neutron Fluence Calculation Methodology for APR1400 RPV, APR1400-Z-A-NR-14015-NP, Rev0, Nov.2014.

[4] SCALE 6.21 computer program, Reactor and systems Division, Oak Ridge National Laboratory ORNL, USA, August 2016.