# RADIATION RELEASE CHARACTERIZATION OF PWR SPENT FUEL ASSEMBLIES GENERATED FROM KOREAN NUCLEAR POWER PLANTS

by

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**Spent nuclear fuel should be kept under safe management until it is disposed of permanently. Because of this, it is important to understand its radiation release characteristics. In this paper, the Monte Carlo method is applied to evaluate the radiation release characteristics of two types of PWR spent fuel assembly generated from the operating plants in Korea: Westinghouse and Korea Standard Nuclear Power Plant. The source terms were calculated using ORIGEN-ARP. The neutron and photon (or gamma) dose distributions along the vertical and horizontal directions of each spent fuel assembly were evaluated using MCNPX code. Compared with the two dose distributions, the photon dose was found to be about $10^5$ times higher than the neutron dose.**

*Key words: spent fuel assembly, radiation release characteristics, source term, neutron dose, gamma dose*

## INTRODUCTION

Spent nuclear fuel contains high concentrations of radioactive material that is generated due to nuclear reactions. Since the radioactive material can have deleterious effects on humans and on the environment, spent nuclear fuel should be kept under safe and strict management until it can be disposed of permanently. For the purpose of managing it safely, it is important to know its properties, particularly the radiation release characteristics.

In this study, the Monte Carlo method was applied to evaluate the radiation release characteristics of two types of PWR spent fuel assembly generated from the operating plants in Korea: Westinghouse type and Korea Standard Nuclear Power Plant (KSNP) type. The neutron and photon dose distributions along the vertical and horizontal axes of each spent nuclear fuel assembly were calculated and compared. This comparison has produced the ratios of the two dose distributions, which can be used as criteria to determine if the fuel assembly is intact, with the help of an appropriate instrument, assuming that the ratio changes at the locations where the defects occur.

## RADIATION RELEAS CHARACTERISTICS EVALUATION

A Westinghouse 17 17 fuel assembly of Kori units 3 & 4 [1] and a KSNP 16 16 fuel assembly of Ulchin units 5 & 6 [2] were chosen to be evaluated. The Westinghouse type fuel assembly consists of 248 fuel rods, 16 burnable absorbers ($UO_2$-$Gd_2O_3$, $UO_2$ 2.6%, $Gd_2O_3$ 6%), and 25 guide tubes. The KSNP type fuel assembly consists of 236 fuel rods and 5 guide tubes whose size is equivalent to 4 fuel rods. Table 1 summarizes the main parameter values of each fuel assembly. From the preliminary study [3], the axial burn-up profiles of the two fuel assemblies exhibit a similar shape, that is, the burn-up decreases at the top and at the bottom of the fuel assembly where a flux gradient is caused by neutron leakage.

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**Table 1. The main parameter values of each fuel assembly**

| Parameter | Value | |
|---|---|---|
| | Westinghouse type | KSNP type |
| Fuel temperature [°C] | 600 | 687.8 |
| Coolant temperature [°C] | 310 | 312.2 |
| Fuel pin radius [cm] | 0.4096 | 0.4015 |
| Air gap thickness [cm] | 0.0082 | 0.0165 |
| Cladding thickness [cm] | 0.0572 | 0.0570 |
| Fuel pin pitch [cm] | 1.2598 | 1.2850 |
| Total fuel rod length [cm] | 409.3667 | 409.54 |
| Active fuel length [cm] | 397.50 | 381.5 |
| The storage rack pitch [cm] | 27.40 | 27.40 |
| Fuel enrichment [%] | 4.1 | 4.5 |

For the source term evaluations, ORIGEN-ARP [4], which is a part of the 5.1 version of the SCALE (Standardized Computer Analysis for Licensing Evaluation) package, was used. Though it does not include the nuclear data library for 16 16 fuel assembly, the ORIGEN-ARP was applicable to the source term evaluation of the KSNP 16 16 fuel assembly since it is very similar to the Westinghouse 17 17 fuel assembly, regarding the size, material composition, and geometry. The common assumptions for the source term evaluation are as follows: burn-up of 48 000 MWD/MTU, average power of 38.54 MW/MTU, and cooling time of 9 days.

Under the above assumptions, the neutron and photon source terms necessary for the Monte Carlo simulation were evaluated using ORIGEN-ARP. For the neutron source terms, the neutrons generated from ($\alpha$, $n$) reaction, spontaneous fission neutrons, and delayed neutrons were considered. The photons generated from the light elements, actinides and fission products were used as the photon source terms. Table 2 summarizes the neutron and photon source terms evaluated by ORIGEN-ARP, respectively.

For the Monte Carlo simulation using MCNPX code [5], considering the characteristics of two fuel assemblies where the same geometries are repeated, universeal fill cards were used to construct the input geometry of the fuel assembly, as shown in fig. 1.

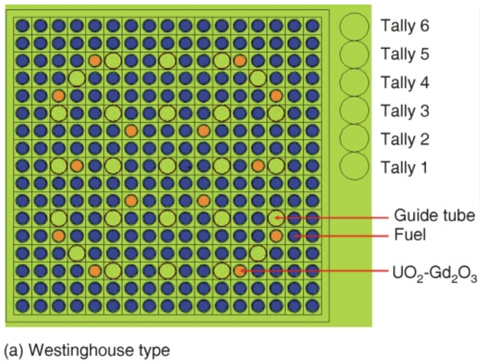
In the Monte Carlo calculation, tally cards were used to specify the type of information that the user requires from the calculation. F4 tally (flux averaged over a cell) and segmentation tally were used to assess the neutron and photon dose in this study, respectively. F4 tally calculates the average number of particles entering a unit cell, so its unit is particles/cm$^2$.

The segmentation tally subdivides a cell or a surface, which is the object of a dose assessment, into segments for tallying purposes, and assesses the radiation dose in each segment. In this study, this tally assessed the radiation dose distribution along the axial direction of the fuel assembly, as shown in fig. 2. Each section was at 5 cm intervals.

In order to assess the radiation dose distribution along the horizontal direction of the fuel assembly, each different location was designated and shown in fig. 3 as tally 1 to tally 6 marked in the upper right-hand side in fig. 3. The tally was assumed to be a cylinder with a radius of 1 cm.

**Table 2. Source terms of two types of spent nuclear fuel assemblies**

| | Neutron source term (#/fuel rod) | | | Photon source term (#/fuel rod) | | |
|---|---|---|---|---|---|---|
| | Energy interval [MeV] | Westinghouse | KSNP | Energy interval [MeV] | Westinghouse | KSNP |
| 1 | 1.00E-11 1.00E-08 | 2.51E-06 | 2.40E-06 | 1.00E-02 5.00E-02 | 3.25E+14 | 3.39E+14 |
| 2 | 1.00E-08 3.00E-08 | 5.24E-06 | 4.84E-06 | 5.00E-02 1.00E-01 | 1.21E+14 | 1.27E+14 |
| 3 | 3.00E-08 5.00E-08 | 6.62E-06 | 6.06E-06 | 1.00E-01 2.00E-01 | 1.67E+14 | 1.73E+14 |
| 4 | 5.00E-08 1.00E-07 | 2.14E-05 | 1.95E-05 | 2.00E-01 3.00E-01 | 6.79E+13 | 6.97E+13 |
| 5 | 1.00E-07 2.25E-07 | 7.57E-05 | 6.87E-05 | 3.00E-01 4.00E-01 | 6.21E+13 | 3.49E+13 |
| 6 | 2.25E-07 3.25E-07 | 7.81E-05 | 7.08E-05 | 4.00E-01 6.00E-01 | 1.93E+14 | 1.99E+14 |
| 7 | 3.25E-07 4.00E-07 | 6.75E-05 | 6.12E-05 | 6.00E-01 8.00E-01 | 3.00E+14 | 3.17E+14 |
| 8 | 4.00E-07 8.00E-07 | 4.63E-04 | 4.20E-04 | 8.00E-01 1.00E+00 | 5.07E+13 | 5.28E+13 |
| 9 | 8.00E-07 1.00E-06 | 2.83E-04 | 2.57E-04 | 1.00E+00 1.33E+00 | 1.49E+13 | 1.49E+13 |
| 10 | 1.00E-06 1.13E-06 | 2.00E-04 | 1.81E-04 | 1.33E+00 1.66E+00 | 8.83E+13 | 9.31E+13 |
| 11 | 1.13E-06 1.30E-06 | 2.79E-04 | 2.53E-04 | 1.66E+00 2.00E+00 | 1.96E+12 | 1.93E+12 |
| 12 | 1.30E-06 1.77E-06 | 8.63E-04 | 7.82E-04 | 2.00E+00 2.50E+00 | 3.90E+12 | 3.90E+12 |
| 13 | 1.77E-06 3.05E-06 | 2.93E-03 | 2.65E-03 | 2.50E+00 3.00E+00 | 2.75E+12 | 2.90E+12 |
| 14 | 3.05E-06 1.00E-05 | 2.58E-02 | 2.33E-02 | 3.00E+00 4.00E+00 | 2.32E+10 | 2.44E+10 |
| 15 | 1.00E-05 3.00E-05 | 1.30E-01 | 1.18E-01 | 4.00E+00 5.00E+00 | 9.25E+04 | 8.40E+04 |
| 16 | 3.00E-05 1.00E-04 | 8.31E-01 | 7.53E-01 | 5.00E+00 6.50E+00 | 3.71E+04 | 3.37E+04 |
| 17 | 1.00E-04 5.50E-04 | 1.18E+01 | 1.07E+01 | 6.50E+00 8.00E+00 | 7.27E+03 | 6.60E+03 |
| 18 | 5.50E-04 3.00E-03 | 1.50E+02 | 1.36E+02 | 8.00E+00 1.00E+01 | 1.54E+03 | 1.40E+03 |
| 19 | 3.00E-03 1.70E-02 | 2.03E+03 | 1.84E+03 | | | |
| 20 | 1.70E-02 1.00E-01 | 2.83E+04 | 2.57E+04 | | | |
| 21 | 1.00E-01 4.00E-01 | 1.92E+05 | 1.74E+05 | | | |
| 22 | 4.00E-01 9.00E-01 | 4.19E+05 | 3.80E+05 | | | |
| 23 | 9.00E-01 1.40E+00 | 4.21E+05 | 3.82E+05 | | | |
| 24 | 1.40E+00 1.85E+00 | 3.41E+05 | 3.11E+05 | | | |
| 25 | 1.85E+00 3.00E+00 | 6.63E+05 | 6.05E+05 | | | |
| 26 | 3.00E+00 6.43E+00 | 6.02E+05 | 5.50E+05 | | | |
| 27 | 6.43E+00 2.00E+01 | 5.26E+04 | 4.75E+04 | | | |
| Total | | 2.72E+06 | 2.48E+06 | | 1.40E+15 | 1.46E+15 |



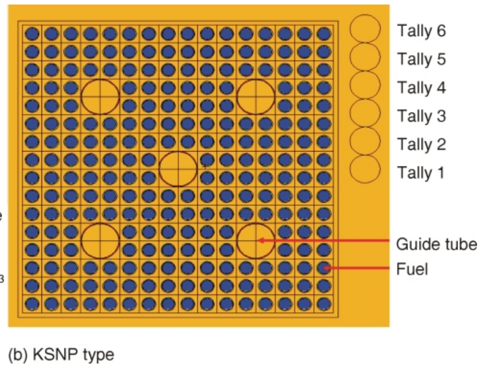




**Figure 1. Sectional view of two fuel assemblies**

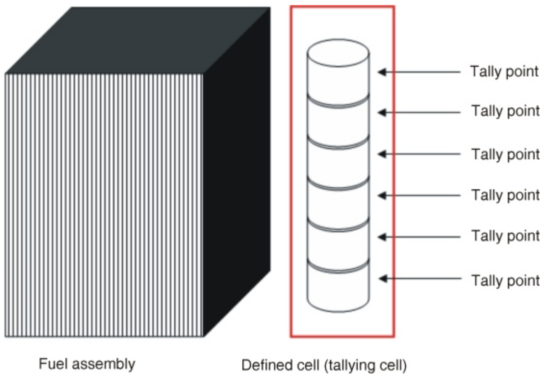



**Figure 2. Conceptual diagram of the segmentation tally**

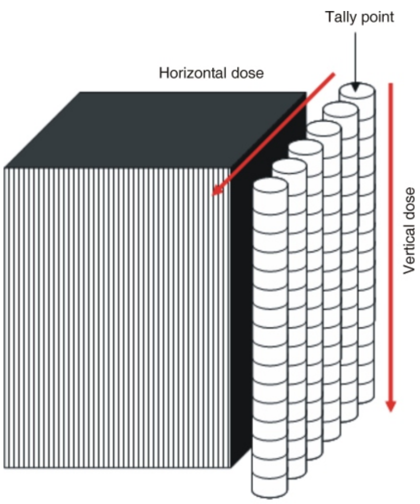



**Figure 3. Conceptual view of the tallies along the vertical and horizontal directions**

## RESULTS AND DISCUSSION

To calculate the radiation dose at the outside surface of the fuel assembly, the tallies assessed previously were multiplied by the initial particle numbers of the source terms evaluated by ORIGEN-ARP. The neutron and photon dose can be obtained by multiplying the tally value converted considering the dose conversion factors of ICRP-74 [6] by the initial particle number per fuel rod and fuel rods per assembly.

The vertical dose distributions of a fuel assembly were evaluated at 6 different locations along the horizontal direction. Figures 4 and 5 show the results of the neutron dose assessment and photon dose assessment of each fuel assembly, respectively.

As shown in tab. 3, the photon dose (~$10^7$cSv/h) was about $10^5$ times higher than the neutron dose (~$10^2$ cSv/h) for both two cases. This was attributed to the difference between the penetrating powers of neutrons and photons in material and the difference between the initial particle numbers per fuel rod, *i. e.* the source term.

The simulation showed that the neutron tally per particle was $10^4$ times higher than the photon tally. In general, a fuel assembly is composed of high atomic number materials. When neutrons and photons pass

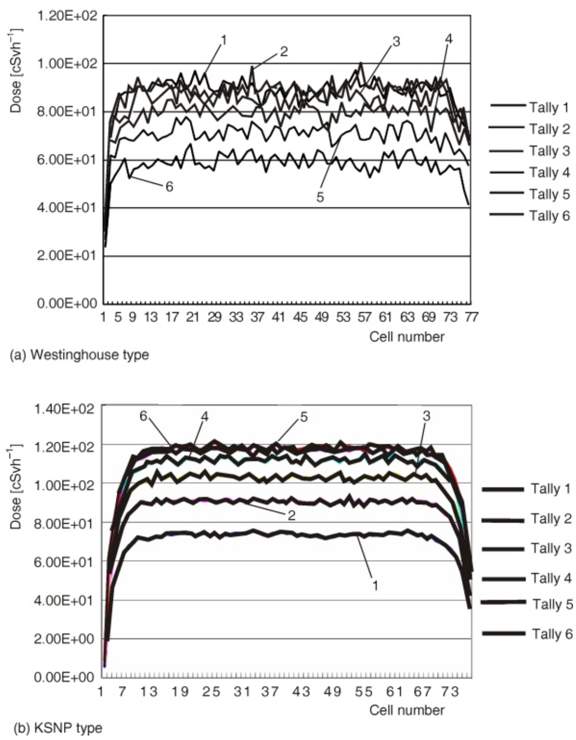



**Figure 4. Evaluation results for the neutron dose [cSvh$^{-1}$]**

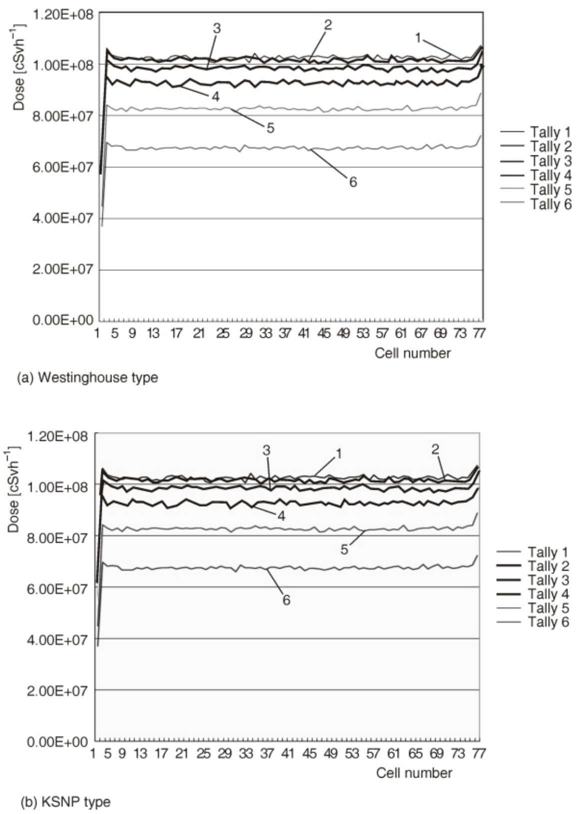



**Figure 5. Evaluation results for the photon dose [cSvh$^{-1}$]**

**Table 3. Ratio of the photon to neutron dose**

| Tally # | Fraction (photon dose/neutron dose) | |
|---|---|---|
| | Westinghouse | KSNP |
| 1 | 2.188E + 05 | 1.979E + 05 |
| 2 | 2.175E + 05 | 2.016E + 05 |
| 3 | 2.191E + 05 | 2.025E + 05 |
| 4 | 2.227E + 05 | 1.997E + 05 |
| 5 | 2.272E + 05 | 1.983E + 05 |
| 6 | 2.187E + 05 | 1.976E + 05 |

through the structure of the fuel assembly, the photons whose penetrating powers are lower than those of neutrons are more reduced inside it. Hence, the neutron dose outside the fuel assembly would be $10^4$ times higher without considering the initial flux. Since the initial number of photons is much higher than that of neutrons, however, considering all these factors, the photon dose was found to be $10^5$ times higher than the neutron dose.

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## REFERENCES

[1] ***, Korea Electric Power Co., 1985, Kori Units 3 & 4, Final Safety Analysis Report (in Korean), Korea Electric Power Co.
[2] ***, Korea Hydro & Nuclear Power Co., 2002, Ulchin Units 5 & 6, Final Safety Analysis Report (in Korean), Korea Hydro & Nuclear Power Co.
[3] Moon, J. H., *et al*., Calculation of the Axial Burn-up Profiles for the PWR Spent Fuel Assemblies, Interim Report of the R&D Project of "Development of the Methodology to Detect the Position of Defects on Nuclear Spent Fuel" (in Korean), Dongguk University, 2009
[4] Gauld, I. C., *et al*., ORIGEN-ARP: Automatic Rapid Processing for Spent Fuel Depletion, Decay, and Source Term Analysis, Oak Ridge National Laboratory report ORNL/TM-2005/39, 2006
[5] ***, MCNPX User's Manual Version 2.5.0 (Ed. D. B. Pelowitz), Los Alamos National Laboratory Report, LA-CP-0369, 2005
[6] ***, International Commission on Radiological Protection (ICPR), 1996, Conversion Coefficients for Use in Radiological Protection against External Radiation, ICRP Publication 74, Elsevier, London

**Џу Хјан МУН**

**КАРАКТЕРИЗАЦИЈА ЕМИСИЈЕ ЗРАЧЕЊА ИЗ ИСЛУЖЕНОГ PWR ГОРИВА НУКЛЕАРНИХ ЕЛЕКТРАНА У КОРЕЈИ**

Ислуженим нуклеарним горивом, пре његовог коначног одлагања, потребно је управљати на сигуран начин. Отуда је значајно познавати особине емисије зрачења из њега. У овом раду примењена је Монте Карло метода да се оцене особине емисије зрачења две врсте ислужених PWR горивних ансамбала насталих у Корејским нуклеарним електранама у раду: Вестингхаусовим нуклеарним електранама и корејским стандардним нуклеарним електранама. Изворни чланови рачунати су коришћењем ORIGEN-ARP програма. Расподеле неутронских и фотонских (или гама) доза у вертикалном и хоризонталном правцу сваког ансамбла ислуженог горива оцењене су MCNPH кодом. Поређењем расподела доза утврђено је да је доза фотона око $10^5$ пута већа од дозе неутрона.

*Кључне речи: ислужено гориво, особине емисије зрачења, извор зрачења, доза неутрона, доза гама зрачења*