

Evaluate the Approximation of Geometry for Nuclear Fuel Assemblies Based on the Results of Gamma Radiation Absorption Rate Calculations

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ABSTRACT

Due to the differences in computer programs in the methods of calculating them, designing the geometric shapes, and distributing the material in the geometric shapes. We investigated the effect of these differences separately on the results of the calculations of the rate of absorbed dose of gamma rays in this research. The spent fuel assemblies (SFAS) of nuclear fuel acquired from the Serpent program served as the source of gamma-radiation in this investigation. The absorber gamma-dose rate was calculated using a number of methodologies, including the Serpent Monte Carlo method and approximation engineering techniques (BREZ). As a consequence, the possibility of calculating the absorbed dosage of gamma radiation in approximate geometry using engineering programs is studied.

Keywords: Serpent, BREZ, VVER-1200, the absorbed dose rate of gamma radiation, the linear attenuation coefficient,

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I. Introduction

To assess the radiation characteristics of nuclear fuel, computer programs such as Serpent [1], MCNP [2], MCU [3], SCALE [4], and BRIZ [5] are employed. The calculating algorithms, nuclear libraries (Endfb7 [6] and Jiff [7]), and geometry of example objects differ between these applications. And the changes in the outcomes of the computations are the result of these differences between programs [8-10].

The main explanations for the discrepancies in calculation results are the calculation method and the geometry of the representative material. The geometric shape of an object and the distribution of materials within it are referred to as approximate geometry (homogeneous or heterogeneous distribution). Real geometry is utilized directly in programs that implement the Monte Carlo method, while approximate geometry is employed in engineering applications (SCALE and BRIZ) (Serpent, MCNP, and MCU).

The aim of work

By using the Monte Carlo method, calculate the influence of the shape of the representative material for the Gamma-source on the calculation results of the absorbed dose of gamma radiation at various distances from the fuel assembly (Serpent program). In addition, assess the impact of the calculating technique on the absorbed dose rate of gamma radiation at various distances from the surface.

Model description

The VVER-1200 fuel assembly, with a burnup of 60 MWday/kgU, was the source of gamma radiation in the first investigation. Tables 1 and 2 show the geometric specifications and properties of the spent fuel assembly, respectively. The variation in the calculation results can only be due to the geometry of the representative material for the Gamma-source because a unified calculation method (Monte Carlo method using Serpent software) and a constant nuclear data library (ENDFB7) were used in this study.

Table 1. Geometric parameters of fuel assemblies of the VVER-1200 reactor.

Geometry	Hexagonal
Fuel assemblieslength (mm)	3750
Mass UO ₂ (kg)	534
Enrichment of ²³⁵ U (%)	4.9
Fuel density (g/cm ³)	10.5
Average fuel temperature (°K)	1000
Average density of water (g/cm ³)	0.72
Average temperature of the heating medium (°K)	587
The number of fuel rods in one fuel assembly: 312 rods	
Inner/outer diameter of the fuel pellet (mm)	1.4/7.57
Inner/outer claddingdiameter (mm)	7.73/9.1
Cladding material	alloy E110
Center tube	
Inner/outer diameter (mm)	11.0/13.0
Material	alloy E635
The number ofGuide tubein one fuel assembly: 18 tubes	
Inner/outer diameter (mm)	10.9/12.6
Material	alloy E635

Table 2.Characteristics of spent fuel assemblies of the VVER-1200 reactor.

Activity, (Bq)	4.6*10 ¹⁸
Burnup (MW·day/kgU)	60
Cooling time	0
Air volume, (cm ³)	105014.3
Spent fuel volume, (cm ³)	50877.8
Cladding volume, (cm ³)	23456.3
Air density, (g/cm ³)	0.00121
Spent fuel density, (g /cm ³)	10.497
Cladding density, (g/cm ³)	6.464

II. Result and discussion

As illustrated in Figures 1 and 2, the first scenario in the first analysis comprises diverse distributions of materials in the same geometric form (hexagonal geometry), such as heterogeneous distributions of materials (actual geometry (Figure 1)) or homogeneous distributions of materials (approximate geometry (figure 2)).The actual geometry is represented by the characteristics of spent fuel assemblies in table 2, and these characteristics are utilized to generate the approximation geometry. To ensure a uniform distribution of components, the spent fuel and air volume are blended. In the approximation geometry, the cladding volume of all fuel rods of fuel assemblies formed the outside boundary in the geometry.

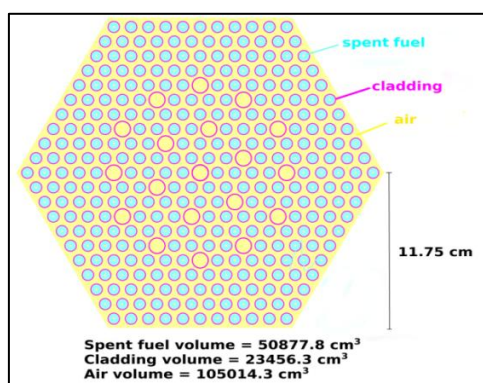


Figure 1. Real geometry of fuel assembly (heterogeneous distribution of materials).

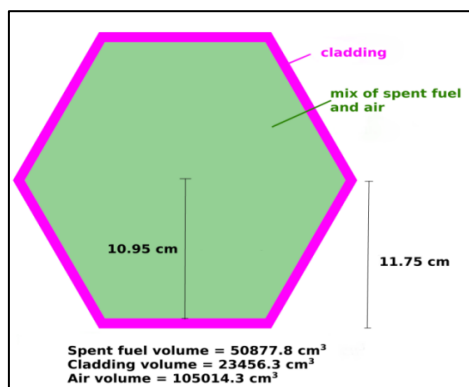


Figure 2. Approximate geometry of fuel assembly (homogeneous distribution of materials).

In these circumstances, the source's activity is the same as in table 2. The absorbed dose rate of gamma radiation (D) was computed in actual (D. R) and approximate geometries (D. A) at various distances from the surface of the fuel assembly. Table 3 and Figure 3 summarize the findings.

Table 3. The absorbed dose rate of gamma radiation at different distances from the surface of fuel assembly (d) in real (D. R) and approximate (D. A) geometries and the standard deviation between the results (σ).

d, cm	D, R 10^4 Gy/h	Δ D.R, %	D, A 10^4 Gy/h	Δ D.R, %	σ , %
10	13.0	0.4	10.6	0.5	18
20	8.50	0.5	6.97	0.5	18
30	6.22	0.5	5.19	0.5	17
40	4.90	0.5	4.07	0.6	17
50	4.00	0.5	3.34	0.5	16
100	1.93	0.6	1.65	0.6	15

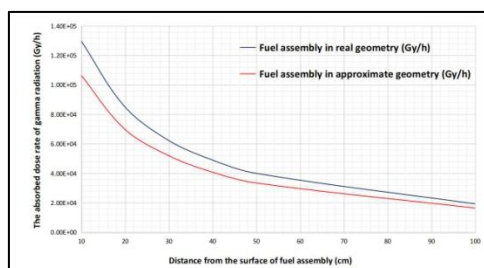


Figure 3: The absorbed dose rate of gamma radiation at different distances from the surface of fuel assembly in real and approximate geometry.

The approximation geometry of the spent fuel assemblies, as shown in table 3 and figure 3, decrease the density of the gamma-source (spent fuel), and it has the effect of further reducing the flux of emitted photons from spent fuel, reducing the absorbed dose rate from the fuel assembly. From the other hand, because the radius of the fuel element is only 0.38 cm in the case of real geometry, the attenuation of the released photons is lower, and thus the absorbed dose rate is larger. The largest difference between the findings is not more than 18%.

Furthermore, the thickness of the cladding in the approximation geometry was (0.8 cm) for the fuel assembly, but the thickness in the actual geometry was (0.137 cm) for each fuel element in the fuel assembly, resulting in a discrepancy in the amount of attenuation for the same material.

As illustrated in Figures 4 and 5, the second case in the first analysis comprises diverse geometric shapes, such as hexagonal and cylindrical geometry, with the same homogenous distribution of materials. The estimated geometry was created using the parameters of fuel assemblies listed in table 2. The volume of expended spent fuel was spread throughout hexagonal and cylindrical geometries, excluding air. The cladding volume of all fuel rods in the fuel assembly was spread throughout a hexagonal and cylindrical shape, and the exterior boundary was formed. Table 4 and picture 6 show the absorbed dose rate of gamma radiation at different distances from the surface of the fuel assembly in hexagonal and cylindrical shape.

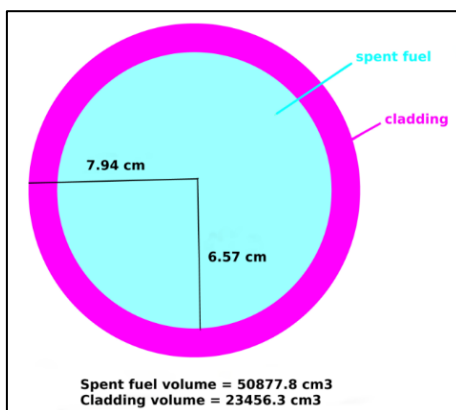


Figure 1: Approximate geometry of fuel assembly in hexagonal geometry.

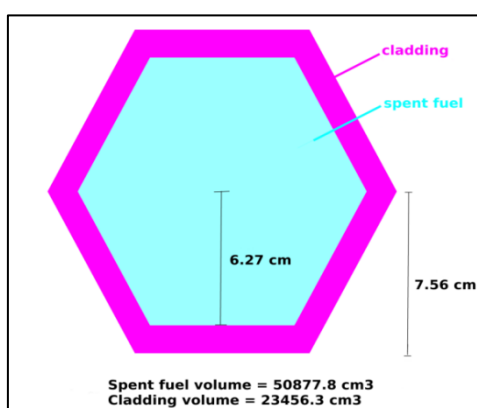


Figure 2: Approximate geometry of fuel assembly in cylindrical geometry.

Table 4: The absorbed dose rate of gamma radiation at different distances from the surface (d) of the fuel assembly in hexagonal (D.H) or cylindrical (D.C) geometry and the standard deviation between the results (σ).

d, cm	D.H, 10^4 (Gy/h)	Δ D.H, %	D.C, 10^4 Gy/h	Δ D.C, %	σ , %
10	6.51	0.8	6.51	0.7	0.1
20	4.11	0.7	4.15	0.8	1.0
30	2.93	0.8	3.01	0.7	2.8
40	2.28	0.8	2.35	0.8	3.2
50	1.82	0.8	1.91	0.8	4.8
100	0.881	0.8	0.925	0.9	5.1

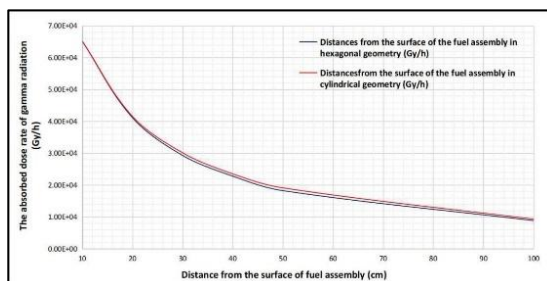


Figure 3: The absorbed dose rate of gamma radiation at different distances from the surface of the fuel assembly in hexagonal or cylindrical geometry.

The attenuation of radiated photons in different geometric shapes (hexagonal or cylindrical geometry) with the same distribution of materials turns out to be the same, as shown in table 4 and figure 6, and the absorbed dose rate almost does not vary. The highest difference between the two sets of findings is 5.1 percent.

In the third analysis, the source of gamma radiation was spent fuel from reactor VVER-1200, which had a burnup of 60 MW day/kgU and had been cooled in a cooling pool for three years. The properties of the spent fuel

are shown in Tables 5, which computed by the Serpent software. The engineering method (BRIZ) and the Monte Carlo method (Serpent) were used in this analysis, both with the same homogeneous distribution of spent fuel (approximate geometry) in the same geometric shape (cylindrical geometry), and thus the difference in the calculation results can only be due to the calculation method. The actual geometry of the fuel assembly characteristics in table 5 was utilized to produce the approximate geometry in cylindrical geometry displayed in picture 7. The volume of spent fuel was dispersed throughout the cylindrical shape, omitting air and cladding. Table 6 and picture 8 show the absorbed dose rate of gamma radiation estimated using the Serpent and BRIZ software at various distances from the surface of the spent fuel in cylindrical geometry.

Table 5: Characteristics of fuel assemblies of a VVER-1200 reactor.

Activity, (Bq)	2.6×10^{16}
Burnup, (MW-day/kgU)	60
Cooling time, (years)	3
Spent fuel volume, (cm ³)	50877.8
Spent fuel density, (g /cm ³)	10.497

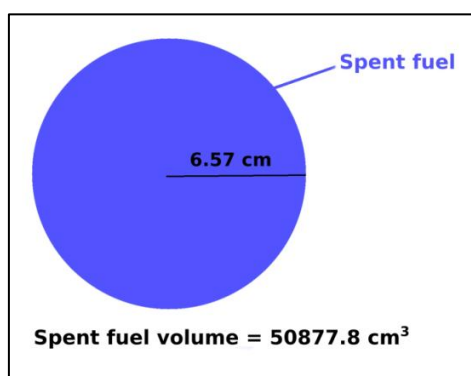


Figure 7. Spent fuel in cylindrical geometry.

Table 6. The absorbed dose rate of gamma radiation was calculated using the Serpent (D.S) and BRIZ (D.B) software at different distances from the surface of the spent fuel in cylindrical geometry and the standard deviation between the results (σ).

d, cm	D. S, 10 ² Gy/h	Δ D.S, %	D.B, 10 ² Gy/h	σ , %
10	2.93	0.8	2.72	7.1
20	1.77	0.8	1.68	4.8
30	1.30	0.8	1.21	7.0
40	1.01	0.8	0.94	6.3
50	0.83	0.8	0.77	7.4
100	0.40	0.8	0.37	6.5

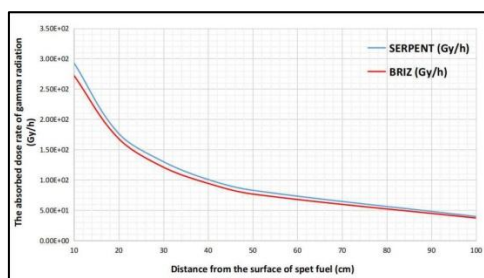


Figure 8. The absorbed dose rate of gamma radiation was calculated using the Serpent and BRIZ software at different distances from the surface of the spent fuel in cylindrical geometry.

The findings of the absorbed dose rate acquired using the Serpent program at all distances from the surface of the spent fuel are somewhat greater than the results of the absorbed dose rate obtained using the BRIZ program. Furthermore, the largest difference between the findings was not greater than 7.4%. And this variance is due to the mathematical technique used.

III. Conclusion

The absorbed dose rate of gamma radiation from the spent fuel with actual geometry is larger than approximate geometry in the event of varied distribution of materials in the same geometric shape. This indicates that changes in the distribution of materials in the spent fuel assembly have an impact on the absorbed dose calculation findings. Whereas, given the same uniform distribution of materials and a different geometric shape or calculating technique, the absorbed dose rate of gamma radiation for both virtually does not vary. The discrepancy is due to a variation in the linear attenuation coefficient of gamma rays in the spent fuel material due to differences in density between the two models, as well as a difference in the cladding thickness.

The systematic nature of the data demonstrates that the total error in using engineering calculation techniques (BREEZ) to evaluate dose quantities cannot exceed 7.4%, which is smaller than the error caused by variances in material distribution in geometry (20 percent). Given the reported error in the calculation of the approved engineering software complex "Breeze," it is recommended for use in resolving problems of maintaining radiation safety at nuclear power plants with complex geometries, with recommendations to introduce multiple geometric shapes to describe the distribution of materials in different radiation sources to broaden its scope of application.

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