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Determination of dose rate during the inspection of spent fuel element in the testing cell

Abdelhady [a] A.

[a] *Reactors Dept., Nuclear Research Center, Atomic Energy Authority,13759 Cairo, Egypt*
amr.abdelhady@gmail.com

ABSTRACT

Testing cell is a multi-purpose shielded hot cell in open pool type reactor and one of its uses is inspecting the irradiated fuel element. Inspecting the irradiated fuel element may rise the radiation dose around the testing cell to level might be higher than the permissible limit. So, evaluating the predicted dose rate around the cell must be determined before the inspecting process. The dose rate level depends mainly on the decay time of the irradiated fuel element and the fuel burn-up. In this regard, a MCNP5 model was performed to simulate the irradiated fuel element inside the testing cell to estimate the radiation dose level around it during the inspection process. The dose rate would be estimated for different fuel burn-up and decay times. The calculations determine the minimal decay times required to manipulate the irradiated fuel element for burn-up ranging between 18745 and 101224.4 MWD/TU.

Keywords: **MCNP code; spent fuel inspection; dose rate; hot cell; ORIGEN2.1 code; open pool reactor.**

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1. INTRODUCTION

Testing cell is a multi-purpose hot cell in open pool type reactor and one of its important purposes is inspecting the spent fuel elements due to the high radioactivity associated with them [1]. Testing cell has heavy concrete walls that provided with lead glass window and tele manipulators to enable the worker to manipulate the irradiated fuel element (FE) during the inspection process.

The inspected fuel element is removed from the core and transported from the main pool to the testing cell via transfer channels, as shown in figure 1, to avoid increasing the dose rate level in the reactor hall.

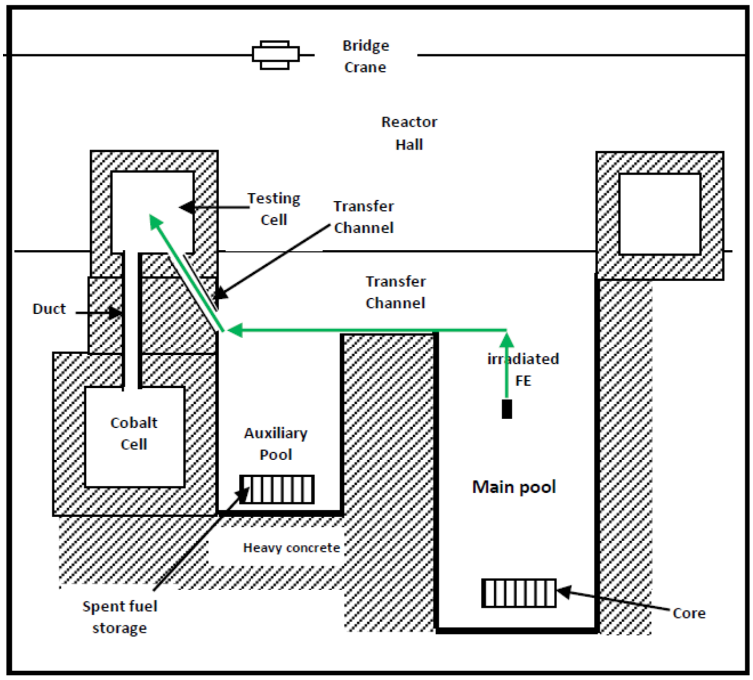


Figure 1: *Transporting the irradiated FE from the core to the testing cell.*

Inspecting fuel element may result in rising the radiation dose rate level around the testing cell depending on its burn-up (BU) and decay time. So, a previous calculation must be carried out to ensure the radiological safety for the workers who located around the cell during the inspection process.

The calculation would introduce a relationship between the dose rate and the decay time of the inspected fuel element for different burn-up. This relationship would determine the corresponding decay times that verify the radiological safety condition for the worker located outside the cell. MCNP5 [2] model was performed to simulate the irradiated fuel element inside the hot cell to carry out the relationship between the dose rate outside the cell with the decay time for different burn up of FE.

2. MATERIALS AND METHODS

2.1. Feature of the testing cell and the fuel element

The reactor is open pool type of 22 MW power and fueled by flat-plate material testing reactor (MTR) fuel. It contains main and auxiliary pools connected with transfer channel to enable transporting radioactive materials and spent fuel elements between them under water surface. The core was built on a supporting grid having 6×5 positions available for placing fuel element or irradiation boxes. The core consists of 29 fuel elements and cobalt device box. Each fuel element has 19 aluminum fuel plates; each plate has a meat made by a dispersion of U_3O_8 particles with an enrichment of 19.7% (in weight of ^{235}U) in a matrix of pure aluminum [3]. The plate active zone is 80 cm×6.4 cm with a meat and cladding aluminum thicknesses of 0.7 and 0.4 mm respectively.

The testing cell is located at the second floor beside the auxiliary pool as shown in figure 1. It is connected to the auxiliary pool by means of a conduct provided with a sample holding cart. The testing cell, as shown in figure 2, has dimensions of 2.5 × 2.5 m and a height of 3 m. The shielding walls of the testing cell are made of heavy concrete of 80 cm thickness with a lead glass view window of 45.7 cm wide by 45.7 cm height and a thickness equivalent to 80 cm of heavy concrete. The testing cell is provided with an iron shielded door of 30 cm thickness for access of personnel and containers. It is provided with master and slave tele manipulators to enable the correct handling of fuel elements, as well as to accomplish the tasks that carried out in this cell [1]. Figure 2 shows a layout for the spent FE located vertically above the working table at distance of 100 cm from the lead glass window during the inspection process.

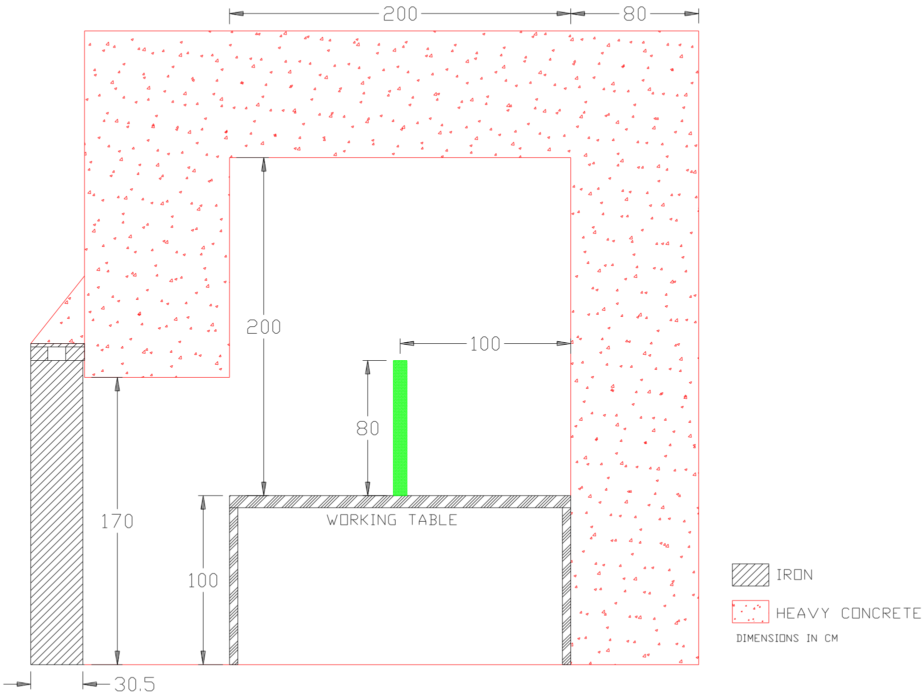


Figure 2: a *cross section of the testing cell shows the FE during the inspection.*

2. 2. Photons spectrum of the spent fuel element

ORIGEN2.1 [4] is a time dependent source term code was developed to determine the nuclides concentrations considering radioactive disintegration and neutron absorption (capture and fission) processes. The fission products and actinides photons source is one of the outputs of the ORIGEN2.1 code. The photons spectrum was determined for FE (^{235}U mass of 404.7 g) that irradiating to a power of 0.759 MW for a time ranging between 50 to 270 days equivalent to BU ranging between 18745 to 101224.4 MWD/TU. The cross sections for the fission products nuclides were obtained from ENDF/B-IV and the cross sections for the actinides and structural material elements were obtained from ENDF/B-V. The photons spectrum was determined using the ORIGEN2.1 18-group photon energy structure for decay times ranging to 5 years after irradiation as shown in figure 3. Then, the photon spectrum source data was introduced into an MCNP5 model to calculate the dose rate. The neutron source, that was contributed from spontaneous fission and (alpha,n) reaction, has intensity of 10^6 n/s per spent FE.

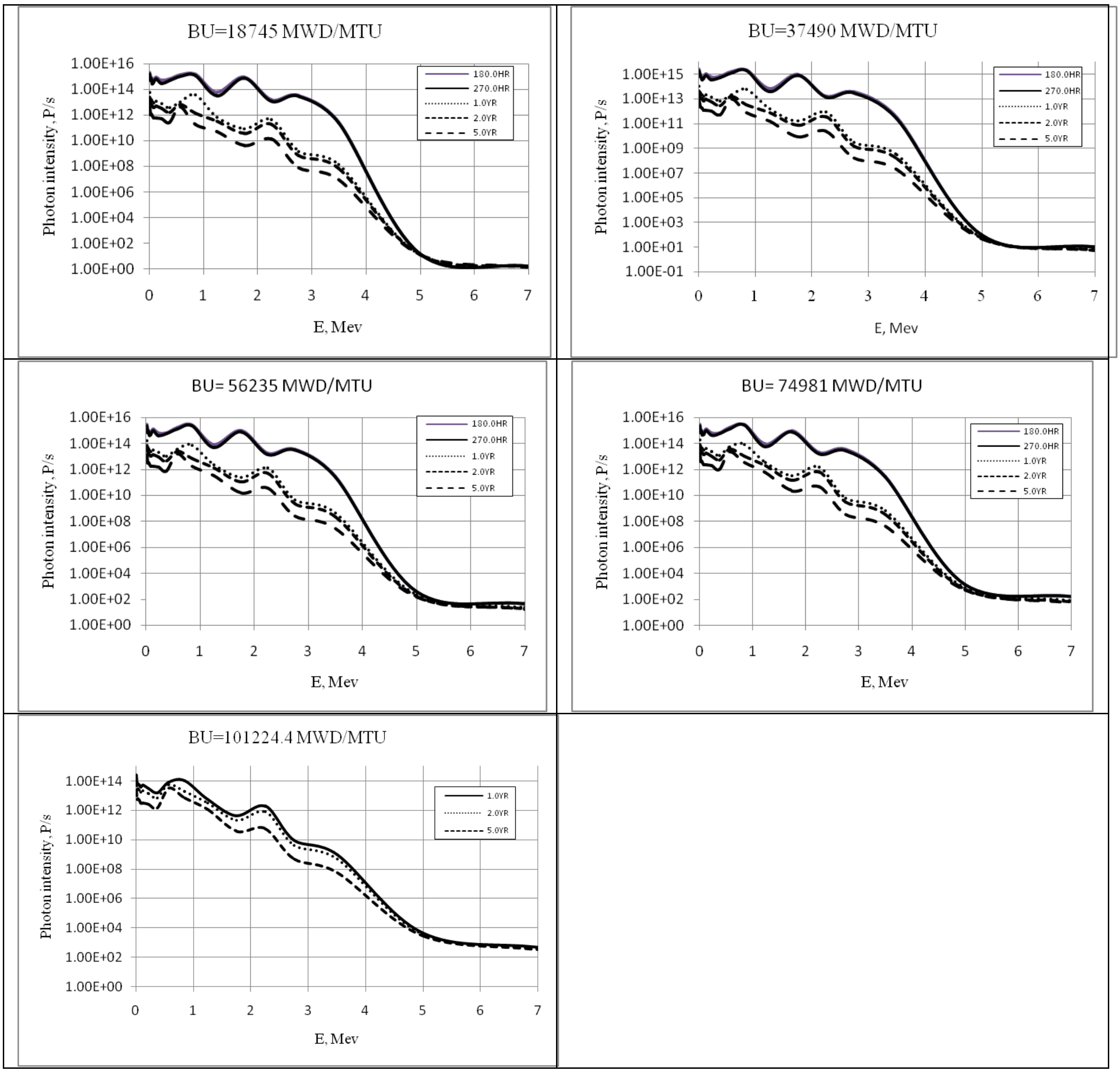


Figure 3: *Photon spectrums at different FE burn-up.*

2.3. MCNP model of the testing cell and the spent fuel element

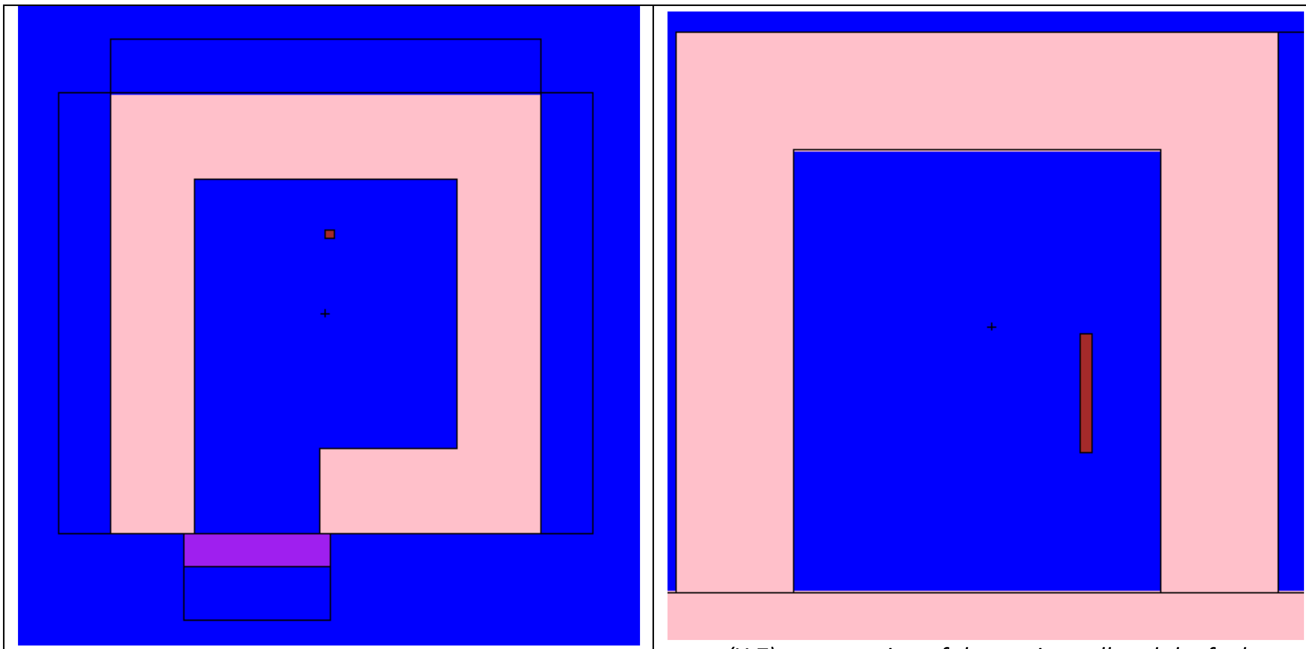
As shown in figure 4, MCNP5 code [2] was used to simulate the testing cell and the spent fuel element during the inspection process, implementing point detector tallies in the four directions around the cell, and applying ICRP-74 [5] gamma rays flux-to-dose conversion coefficients to determine the effective dose rate. ICRP-74 was used instead of the updated ICRP-116 because there is no significant difference in the gamma conversion coefficient values between them. The testing cell was represented by rectangular parallelepiped surfaces and the SDEF, SI, and SP cards were used together to define a volumetric source inside the testing cell. The source is defined as a rectangular parallelepiped with dimensions, of 8 cm×8 cm×80 cm, inside the testing cell. The source is isotropic with a tabulated energy distribution resulting from the ORIGEN2.1 code calculations. The densities of air, heavy concrete, lead glass, and iron are 0.00121, 3.2, 6.22, and 7.86 g/cm^3 respectively. The compositions of heavy concrete, iron, lead glass, and air are shown in table 1.

Table 1: the compositions of heavy concrete, iron, lead glass, and air.

Element	**atomic density (at/b.cm)**	**Weight fraction (%)**	**Weight fraction (%)**	**Weight fraction (%)**
	Heavy concrete	**Iron**	**Lead glass**	**Air**
H	9.893×10^{-03}	---	---	---
O	4.564×10^{-02}	---	15.6453	23.1781
N	---	---	---	75.5267
Mg	6.242×10^{-05}	2.14	---	---
Ca	1.178×10^{-03}	0.29	---	---
Fe	1.448×10^{-02}	94.052	---	---
Si	1.643×10^{-03}	0.046	8.0866	---
Al	4.438×10^{-04}	0.051	---	---
S	2.002×10^{-05}	0.011	---	---
Ar	---	---	---	1.2827
Ti	7.717×10^{-03}	---	0.8092	---
C	9.280×10^{-06}	3.41	---	0.0124
As	---	---	0.2651	---
Pb	---	---	75.1938	---

The F5 tallies were modified by ICRP-74 [5] gamma rays flux-to-dose conversion dose function and a FM multiplier card representing the total photon intensity of the spent FE was used to determine the dose rates. The spent FE would be located vertically at a distance of 100 cm from the lead glass window.

Because the testing cell has shielded with heavy concrete (hydrogenous material) of 80 cm thickness and the aforementioned neutron intensity from the spent FE is 10^6 n/s, the neutron flux would be attenuated before reaching the outer surface of the cell depending on previous calculations. So, the neutron dose calculation was ignored to avoid the waste of time.



(X-Y) cross section of the testing cell and the fuel element. *(Y-Z) cross section of the testing cell and the fuel element.*

Figure 4: *MCNP model of the inspected FE inside the testing cell.*

RESULTS AND DISCUSSION

All the dose rates calculations were of highest accuracy required for proper MCNP code results. Dose rate was calculated for spent FE with the maximum discharge of burn-up (270 days) because it represents the majority of the spent FEs. Figure 5 shows the dose rate distributions versus the decay time for the four directions around the testing cell. The horizontal line represents the permissible dose rate limit (10 μSv/h) derived from the annual permissible dose for the worker (20 mSv/yr) as referenced in [6]. The intersecting points between the dose rate distribution curves and the horizontal line represent the minimal decay times for each direction. The minimal decay times would be 270, 300, 720, and 1800 d for the following directions; in front of the cell window; in the west of the cell; in the east of the cell; and in front of the iron door of the cell.

Since the worker, who stands in front of the cell window, would spend a proper time to manipulate with the inspected fuel element, the dose rate in this location must be carried out as a function of decay time for different burn-up. Figure 6 shows a relationship between the dose rates, in front of lead glass window, with the decay time for different FE burn-up. The intersecting points between the dose rate distribution curves and the horizontal line, in figure 6, would introduce the minimal decay times that verifying the permissible dose rate limit. It is shown that the minimal decay times would be 120, 150, 210, 240, and 270 days for BU of 18745, 37490, 56235, 74981, and 101224.4 MWD/TU respectively.

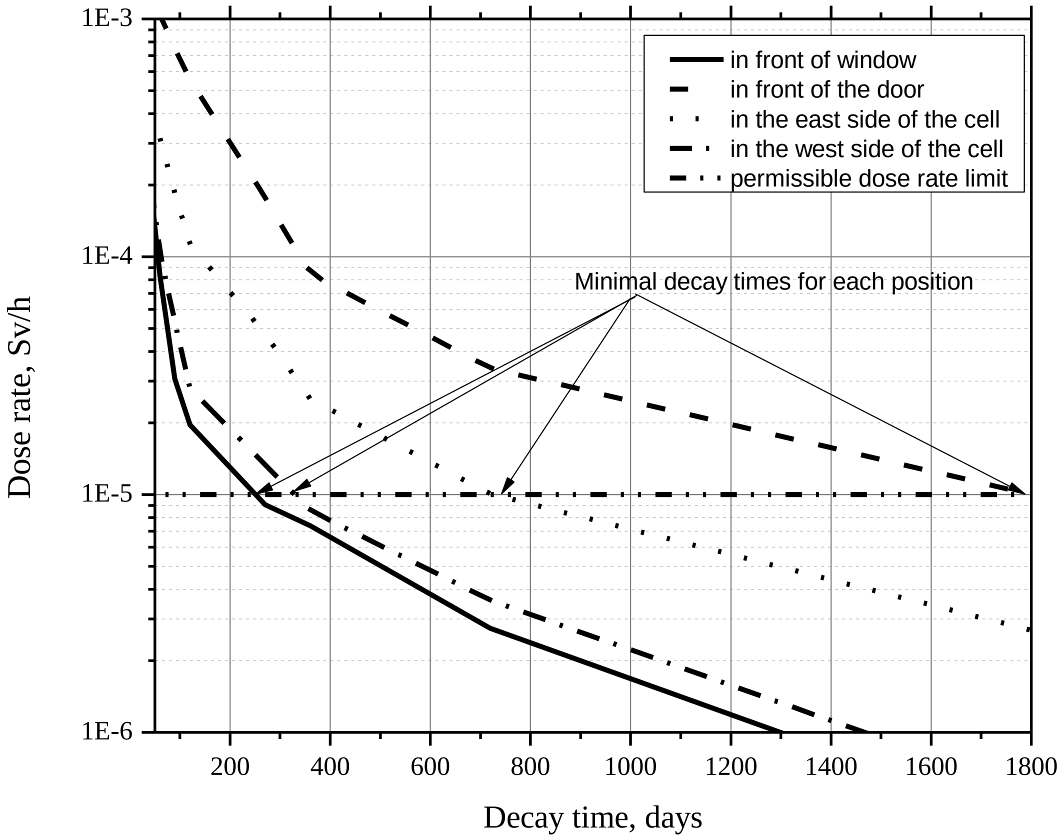


Figure 5: *dose rate around the testing cell versus decay time for FE of the maximum BU of discharge.*

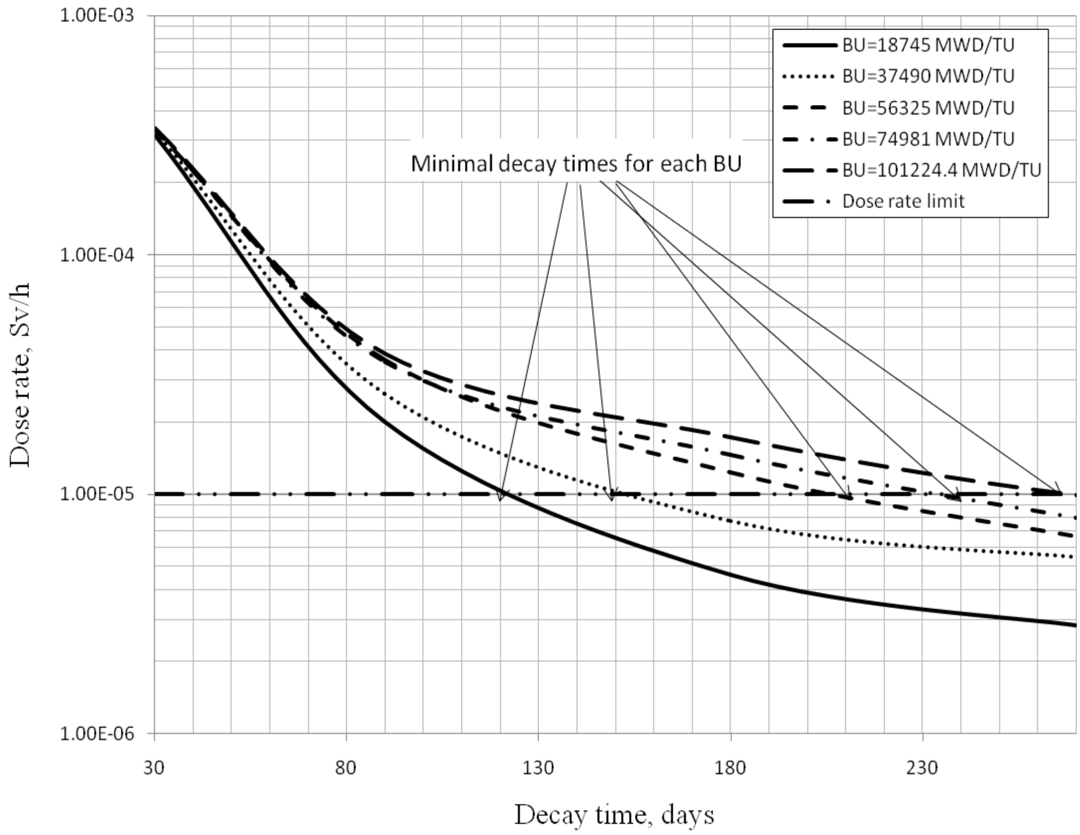


Figure 6: *dose rate in front of the cell window versus decay time for different BU of FE.*

3. CONCLUSIONS

Ensuring the radiological safety condition for the worker located outside the testing cell during inspection of irradiated fuel element is a priority principle. So, the aim of this study was determined a relationship between the irradiated FE activity (representing in decay time and burn-up) and the predicted dose rate around the cell. The relationship was developed in order to determine the minimum decay time, that verifying the radiological permissible limit for the inspector, as a function of FE burn-up as presented in the following table:

Table 2: Minimum decay time as a function of FE burn-up

BU of FE (MWD/TU)	**Decay time (days)**
18745	120
37490	150
56235	210
74981	240
101224.4	270

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