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Determining the optimum thickness of a hot cell for safe processing of spent fuel plate and cobalt source

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HIGHLIGHTS

- Determining the optimum thickness of a hot cell for safe processing of spent fuel plate and cobalt source.
- The design of a hot cell with specific geometric dimensions and materials was simulated using MCNP code.
- The source intensity and the gamma spectrum of the spent fuel plate were obtained using ORIGEN code.

ABSTRACT

To design a hot cell, it is essential to consider all safety requirements and radiation protection acceptance criteria. In this research, the design of a hot cell with specific geometric dimensions and materials was simulated using MCNP code. Then, the gamma dose rate was calculated for a Co-60 source with 1.8×10^{13} Bq activity and a spent fuel plate with 90% burnup and a cooling time of 30 days to determine the appropriate shielding thickness. In these calculations, the source intensity and the gamma spectrum of the spent fuel plate were obtained using ORIGEN code. According to the references, the gamma dose rate criterion of 10 μSv.h^{-1} was used to determine the thickness of the hot cell wall, which is made of barite concrete with 3.35 g.cm^{-3} density and a combination of concrete and paraffin, in different orientations. The results indicate that the necessary optimal thicknesses of shielding for different locations are 80, 65 and 75 cm respectively regarding the irradiation safety criteria.

KEYWORDS
Hot cell
Shielding calculations
Co-60 source
Fuel burnup
Optimal thickness
MCNP and ORIGEN codes

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1 Introduction

Many shielding design safety requirements and acceptance criteria in nuclear facilities, such as hot cells are aligned with the radiation protection safety requirements and acceptance criteria. It is important to note that effective design, high-quality construction and proper operation will create safety through radiation protection (IAEA, 2016). The occupational dose limits in the safety standards of the IAEA are specify a maximum effective dose of 20 mSv per year with an averaged over a period of 5 consecutive years (IAEA, 2009).

The hot cell laboratory can consist of several hot cells, which are serve as protection for highly radioactive materials. The dose rate for expected radioactive materials is established based on working hours. For example, if working time of personnel is 2000 hours per year, the dose rate criterion for this facility is set at 10 μSv.h^{-1}. The hot cell laboratory has various applications, for example, in countries like Malaysia, there is a semi-permanent hot cell for the production of radioisotopes such as ^{99m}Tc and I-131. Also, this type of hot cell used for research employs lead as biological shielding. Furthermore, other hot cells are used to manage radioactive sources with high activity such as those used in radiation and teletherapy. This type of hot cell is employed for research activities such as spent fuel inspection and Post-Irradiation Evaluation (PIE). The main goal of developing the hot cell is to create conditions for research on the behavior of the fuel (Bahrin et al., 2019).

In the design of shielding, the choice of material and its thickness are fundamental principles, as the required shield thickness depends heavily on the energy and the type of the source. Lead or concrete materials are usually used for shielding gamma sources as in most cases, concrete shielding is used because of its cost-effectiveness, availability, and lighter weight. However, it is important to note that for situations where space is limited, it is better to use materials with high atomic number and density such as lead. Conversely, concrete can also be used,

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whose effective density can be increased by using special materials and additives.

There are some literatures considering hot cell shielding. For example, the design of hot cell shielding is carried out using various materials and thicknesses using the MCNP code. Gamma and neutron dose rates are calculated for a spent fuel source according to proposed criteria (Cho et al., 2004). The shielding calculations for the activated first wall of ITER, also using the MCNP code, are addressed in another research study (Yu et al., 2015). Also, the shielding structure of the hot cell door is presented for a specific nuclear power plant where the neutron shielding performance is evaluated through both experimental and simulation methods. Finally, the material and their thicknesses for the hot cell shielding door are detailed (Zhang et al., 2022).

In this research, the optimal thickness of a hot cell for a Co-60 source with an activity of 1.85×10^{13} Bq was calculated using the MCNP code (Denise et al., 2013). The feasibility of using it for as irradiated fuel plate was investigated for the first time in our country that could be an innovation in comparison to other research works. Additionally, in this feasibility study, an irradiated fuel plate with 90% burnup and 30 days cooling time is considered as the most pessimistic scenario for the gamma source calculations using the ORIGEN code (Croff, 2000). Finally, the results of the gamma dose rate have been calculated and the optimal thickness values have been determined with the preset value of 10 μSv.h^{-1} in different parts of the outer of the hot cell.

2 Materials and Methods

The spent fuel plate of the Tehran Research Reactor (TRR) is considered as one of the two sources in the shielding calculations. Therefore, some of the main parameters of the TRR are given in Table 1 (AEOI, 2018). Also, the specifications of the barite concrete as the main shielding material are given in Table 2.

Table 1: Main characteristics of the TRR and fuel plate.

Quantity	Value
Nominal power	5 MW
Coolant and moderator	Demineralized light water
Cooling regime	Forced and Downward
Fuel material	U_3Si_2-Al
Clad	Al
Meat density	4.8 g.cm^{-3}
Control rods	4 Ag-In-Cd and 1 stainless steel
SFE dimensions	$8.01 \times 7.71 \times 89.7$ cm^3
Number of plates in SFE	19
CFE dimensions	$8.01 \times 7.71 \times 161.5$ cm^3
Number of plates in CFE	14
Meat dimensions	$0.07 \times 6.0 \times 60.0$ cm^3
Maximum burnup	60%
U per fuel plate	76 g
Fuel enrichment (%)	19.75
Average neutron flux	1.10×10^{14} cm^{-2}.s^{-1}

Table 2: Composition of radioisotopes and their weight fraction values in barite concrete.

Element	Weight fraction
H	0.003585
O	0.311622
Mg	0.001195
Al	0.004183
Si	0.010457
S	0.107858
Ca	0.050194
Fe	0.047505
Ba	0.463400

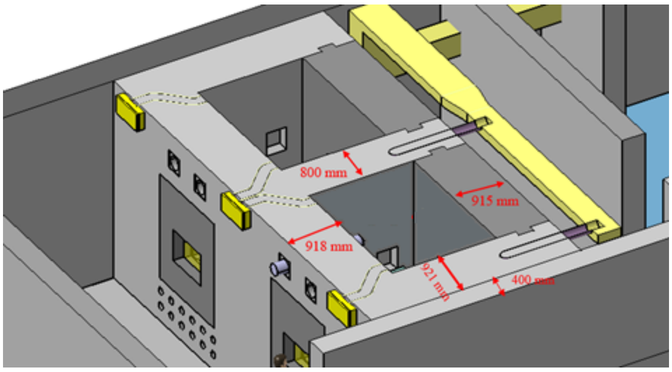

Figure 1: Overview of hot cell with thicknesses.

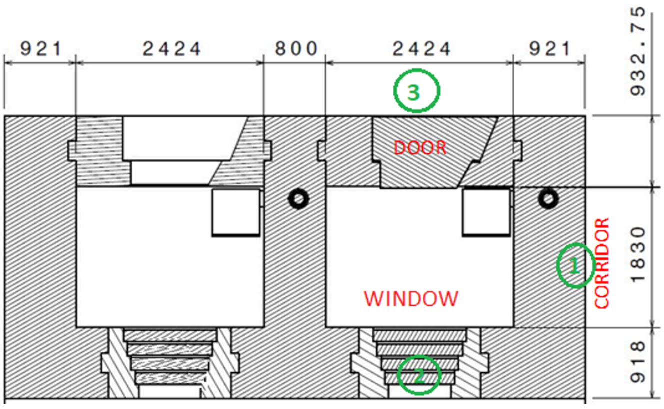

Figure 2: Dimensions and specifications of the hot cell.

The geometry of the modeled hot cell consists of two side-by-side chambers with heavy concrete walls, concrete entry and exit doors on one side, and leaded glass windows and holes for equipment access to the inside of the hot cell on the other. The general view of the hot cell is given in Fig. 1 consisting of two chambers with leaded glass and holes for equipment access.

Figure 2 shows the specifications of the components and dimensions of the system and the materials used in the hot cell. Also, the locations of dose points are given with numbers 1, 2 and 3 in this figure where these points height is in the middle of the window.

Figure 3 shows two different views of the simulated hot cell using the MCNP code with overall approximate dimensions where the side view is given form the middle

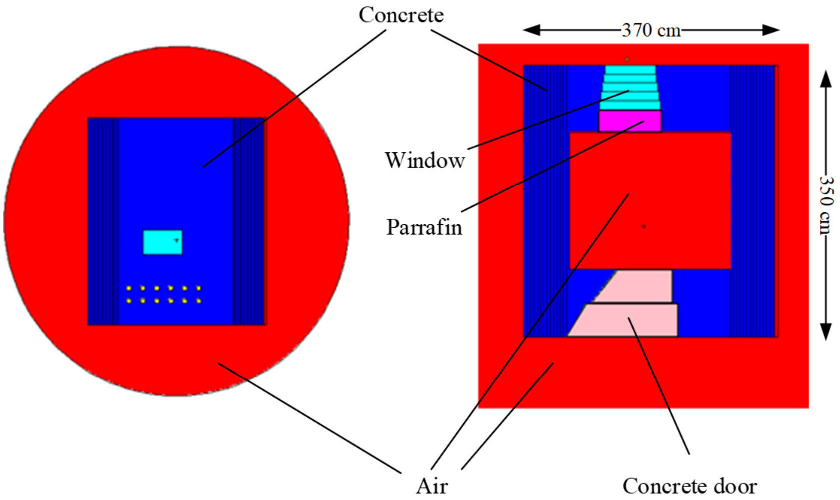

Figure 3: Upper (right) and side view (left) of the simulated hot cell.

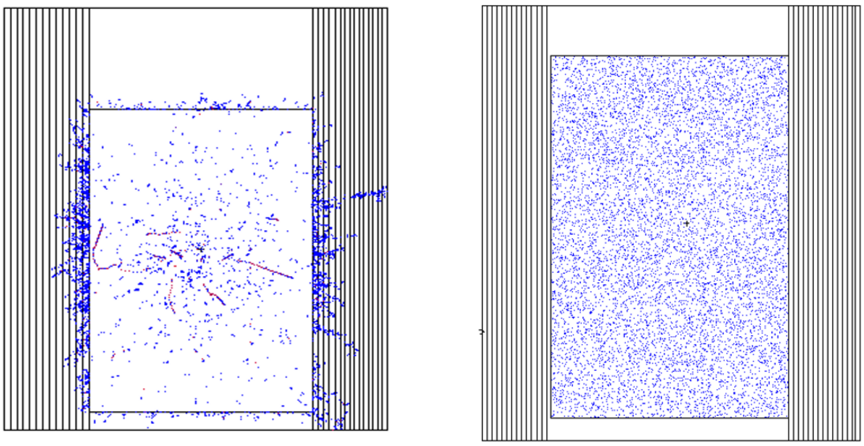

Figure 4: Gamma sources distribution (right) and gamma tracks (left).

of the concrete wall. It highlights the characteristics of the components and the materials used in the hot cell.

The concrete enclosure is constructed from barite concrete in which two rows of six holes are designed for the passage of specialized equipment. The holes are filled with lead components, each measuring 45.5 centimeters in length. The window is composed of lead glass with a density of 6.22 g.cm^{-3}. Compositions and weight percentages of all materials mentioned in this report for the MCNP code input are based on validated references (McConn et al., 2011).

To calculate the gamma dose rate for calculation of the shield thickness in hot cell, a Co-60 point-source with an activity of 1.85×10^{13} Bq and energies of 1.17 MeV and 1.33 MeV along with an irradiated fuel plate have been simulated, separately. The sources are considered homogeneous in all directions.

Fuel plate with density of 4.8 g.cm^{-3} and 90% burned fuel plate is used. The maximum operating power of the reactor, which is 5 MW, was assumed. The cooling time was assumed to be 1 month (AEOI, 2018).

Subsequently, the gamma dose rate values at various locations outside the concrete shield of the hot cell have been calculated, ensuring compliance with the preset criterion of 10 μSv.h^{-1} (Akbar and van Rooyen, 2012; Chilton et al., 1984).

Figure 4 shows Co-60 gamma sources distribution (on the right) and gamma tracks (on the left) in the hot cell, as simulated using the MCNP code. The uniform distribution of gamma sources in entire volume of hot cell and also photon transport in the initial concrete wall towards the out of hot cell clearly could be seen.

The gamma dose rate calculations were performed using F5 Tally of MCNP code. This tally utilized response functions for pointwise dose rate calculations based on flux obtained through IC and IU cards. The MCNP code employed the ENDF/B-VI cross-section library for dose rate calculations.

It is noteworthy that the intensity of the source is 1.60×10^{14} Bq. The dimensions of one fuel plate are $7.7 \times 0.15 \times 61.5$ cm^3 containing meat encased in aluminum cladding. In this research, variance reduction methods based on statistical population control employed, utilizing multiplication techniques and Russian Roulette, along with spatial meshing and energy cutting methods to minimize results errors and execution times. The average error of the calculations in all results was less than 10%.

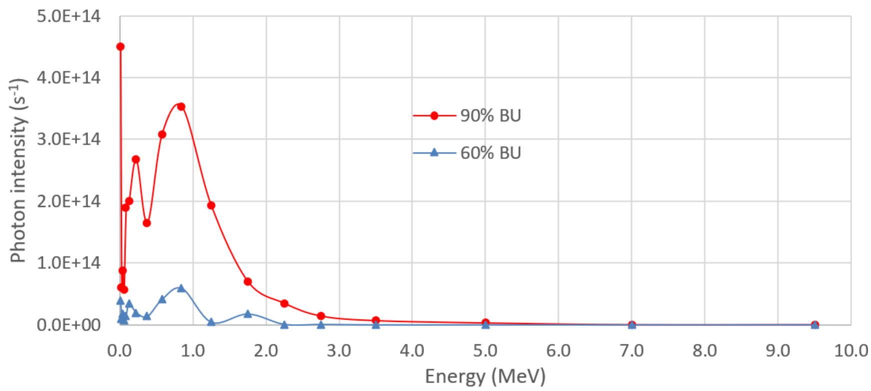

Figure 5: Gamma spectrum of a spent fuel plate.

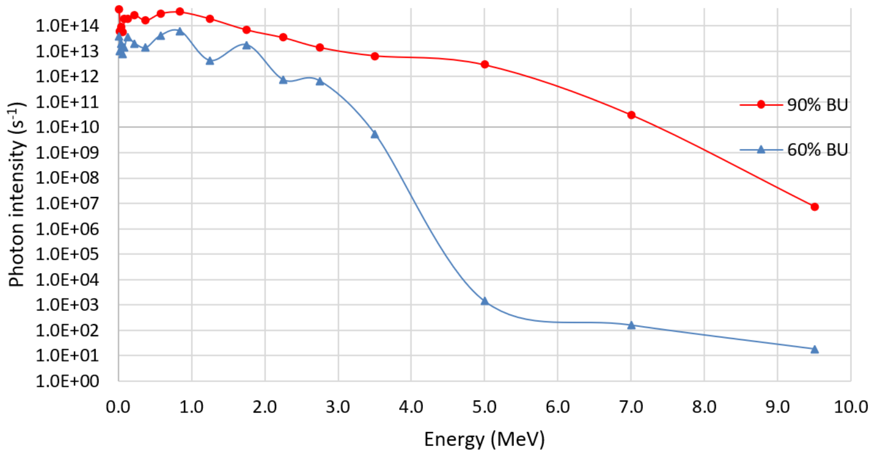

Figure 6: Gamma spectrum of a spent fuel plate in logarithmic scale.

3 Results and discussion

3.1 MCNP Code Validation

The dose rate calculation was conducted for a point location in the TRR (AEOI, 2018). The reactor containment is a right cylinder domed with reinforced concrete exterior walls, and a steel-domed roof that which primarily prevents gamma radiation from escaping outside the containment. It has an integrated wall with 1.80 meters thick from elevation 90.43 to 96.48 meters from where it tapers to 1.20 meters at elevation 108.98 meters at which point it reduces to 0.85 meters for connection to the steel dome. The KCODE card of the MCNP code is used for dose calculation in criticality condition of one equilibrium core. The strength of the fission neutron source for this benchmark is 3.97×10^{17} n.s^{-1} where the energy from the fission of U-235 is released at 5 MW with a recoverable energy per fission of 200 MeV (Lamarsh, 1975).

There is no gamma leakage from the core to the under-containment region due to the full water in the beam tubes and the core being completely submerged. The calculated and measured dose rates at selected points are presented in Table 3. The universal radiation protection survey meter LB 123 UMO as gamma dose and dose rate monitor is used (Boustani et al., 2021). This point is inside the TRR containment and near the concrete wall of the TRR containment as the diameter of reactor is 30 m.

Table 3: MCNP Validation using TRR Measured Gamma Dose Rate.

Point location (x, y, z) (cm, cm, cm)	Dose rate (μSv.h^{-1}) Measurement	MCNP	Relative Diff. (%)
$(1250, 0, 60)$	2.4 ± 0.4	2.3 ± 0.2	0.04

Table 4: The gamma source intensity for a silicide fuel plate.

Cooling (days)		Bu (%)
30	0	
9.97×10^{13}	2.80×10^{14}	60
1.60×10^{14}	2.79×10^{15}	90

As shown in Table 3, there are some differences between the calculated and experimental results. Based on the conducted simulations and their comparison with experimental measurements, it appears that the simulations are reliable and can be utilized for calculating gamma dose rates.

3.2 Dose rate calculations

The gamma dose rates in the hot cell were calculated for 60% and 90% burned fuel plate with a 30 days cooling time using the ORIGEN code and given in Fig. 5.

The logarithmic scale of spent fuel plate gamma spectrum is obtained using the code and is presented in Fig.

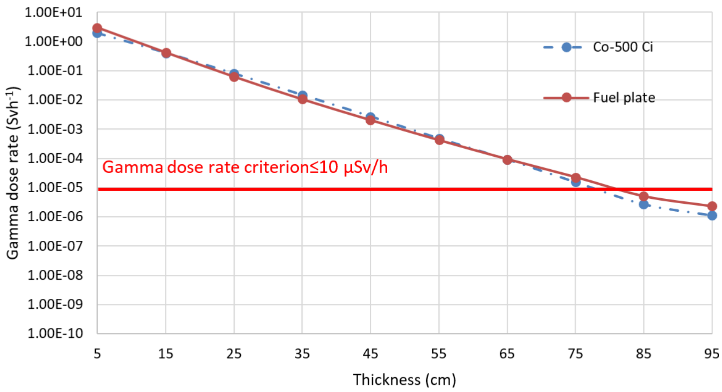

Figure 7: Gamma dose rate for concrete wall for point 1.

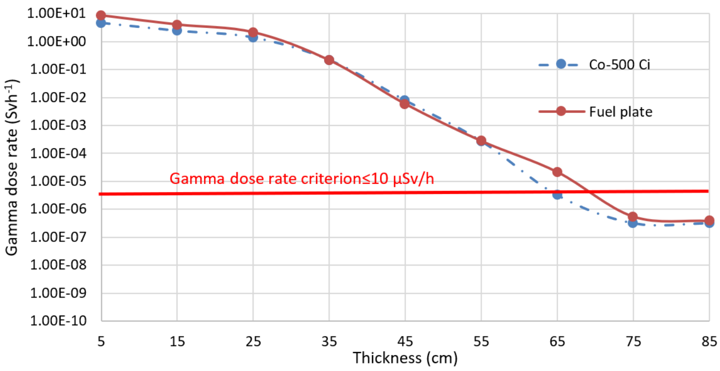

Figure 8: Gamma dose rate for concrete wall for point 2.

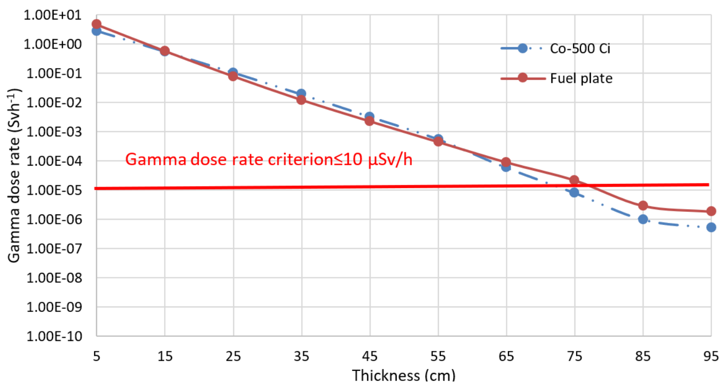

Figure 9: Gamma dose rate for point 3.

6. As shown in this figure, the source intensity is much higher at low energies.

The changes in the intensity of the gamma source are given in Table 4 for immediately and after 30 days cooling for both 90% and 60% burnup scenarios.

In these calculations, the gamma dose rate was obtained for three different locations behind the concrete wall in the corridor (point1), behind the window (point 2) and behind the door (point 3) of the hot cell as shown in Figs. 7 to 9.

As shown in Figs. 7 to 9 and based on the defined criterion of less than 10 μSv.h^{-1}, the optimal thickness of the hot cell concrete wall in different sides are given in Table 5.

As shown in the results, the gamma dose rates outside the hot cell are lower than the preset criterion where the spent fuel plate exhibits higher dose rate values compared to Co-60 in all of the investigated cases.

Table 5: The optimal thickness of the hot cell concrete wall.

Measurement Location	Radioactive Source	Optimal Shield Thickness (cm)	Gamma Dose Rate (μSv.h^{-1})
Point 1	Irradiated fuel plate	85	5.1
	Co-60	85	2.7
Point 2	Irradiated fuel plate	75	0.5
	Co-60	65	3.2
Point 3	Irradiated fuel plate	85	2.9
	Co-60	75	8.1

4 Conclusions

The first time, a hot cell is designed for the test and evaluation of spent fuel in the TRR. In this research, the gamma dose rates for a Co-60 point-source with 1.85×10^{13} Bq activity and an irradiated fuel plate with 90% burnup after 30 days cooling to determination of the optimum shielding thickness of the hot cell using the MCNP code is performed. The source intensity and gamma spectrum of the irradiated fuel plate were generated using the ORIGEN code. It is important to note that the criterion of 10 μSv.h^{-1} was used to determine the optimum thickness of the hot cell wall in various directions. Finally, the results indicate that the gamma dose rate of irradiated fuel plate is slightly higher than the cobalt source. It is concluded that the necessary optimal thicknesses of shielding for point locations of 1, 2 and 3 are 80, 65 and 75 cm respectively regarding the irradiation safety criteria. Finally, it should be noted that these types of calculations are very important for assessing the safety of hot cell.

Conflict of Interest

The authors declare no potential conflict of interest regarding the publication of this work.

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