The design of a Hot Cell with interlocking concrete wall

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Abstract. The Prototype and Plant Development Center (PDC) of Malaysian Nuclear Agency is developing a hot cell facility which is meant for research activities such as spent fuel inspection, fuel fabrication and post irradiation material examination/inspection. The hot cell is of alpha–gamma shielding hot cell type. The size of the hot cell is 4.4 x 3.9 x 3.825 meter. The 1-meter thickness hot cell wall consists of 89 pieces of interlocking curve design concrete. The 1-meter hot cell roof consists of 41 pieces of square shape concrete. The density of all concrete blocks is 2350kg/m³. The hot cell is designed to handle up to 227.5 Ci ⁶⁰Co activity. This paper will discuss the detailed design of the shielding wall, as well as the shielding evaluation/analysis against gamma and fission products of 50% burnup RTP 20wt% fuel type using MCNP.

1. Introduction

Isolation of radioactive material is the best method to protect the operator against hazardous radiation. This isolation is achieved by containing the radioactive materials in an enclosure, which can be in the form of a fume cupboard, glove box, shielded box or hot cell. A shielded enclosure fitted with suitable manipulation systems to allow the performance of operations without subjecting the operator to radiation beyond permissible dose rate is defined as a hot cell [1].

There are many types of hot cell such as alpha–gamma cells, beta–gamma cells, and alpha-beta-gamma cells. These hot cell types are identified by its biological shielding against alpha, beta, gamma and neutron radiation. In addition, the infrastructure of hot cells renders them capable to handle safely large quantities and types of hazardous radio-isotopes inside the enclosed space.

In order to design a hot cell, several specifications are required to be defined such as the type of hot cell, the types of radioactive material to be handled, the expected level of activity to be handled, types of operations to be performed, the viewing system, shielding material selection, size of hot cell and etc.

At Malaysian Nuclear Agency, the existing semi-permanent hot cell is used for radioisotope production such as Technetium-99m (⁹⁹ᵐTc), Iodine-131 (¹³¹I) and also for research works. Lead with chevron brick-built wall design is used as biological shielding [2]. On the other hand, Malaysian Mobile Hot Cell (MHC) is used to handle and manage high activity disused sealed radioactive sources
(DSRS) such as irradiators and teletherapy heads [3]. River sand with a thickness of 1.5 meter is used as biological shielding as it is easily available anywhere and can be procured nearby in the locality of MHC.

In general, a permanent hot cell facility is built next to the nuclear research reactor facility such as hot cell facility in Japan Material Testing Reactor (JMTR), Chalk River Laboratories (CRL) and HANARO (KAERI). These hot cells have been used for research activities such as spent fuel inspection, post irradiation material examination (PIE) and fuel fabrication. However, in Malaysia there is no hot cell facility that is capable of doing such activities. Thus, this paper will study the suitable design of hot cell to support research activities in Malaysian TRIGA PUSPATI Reactor (RTP).

The semi-permanent concept is selected due to the ease/ possibility of assembling and dismantling work within the Nuclear Malaysia site. Currently, no specific site has been selected yet but the possible site is Reactor Hall or “Khidmat Pembangunan” Building. The hot cell also can be assembled near the Thorium Lab if required depending on the user’s requirement. The basic design used is the Chevron brick-built wall and curved brick-built wall, the typical interlocking system in designing a semi-permanent hot cell.

![Figure 1. Chevron brick-built wall [1].](image1)

![Figure 2. Curved brick-built wall [1].](image2)

The main purpose for developing the hot cell is to provide the fuel fabrication preparation for thorium fuel research. The typical type of radiation emitted by thorium isotope are alpha particles and weak gamma rays [4]. Candidates of thorium fuel can be as Th-MOX pellets, Thorium oxide (ThO2) pellets, fuel in a molten state such as solvent mixtures of fluoride salts such as LiF-NaF-BeF2, LiF-NaFRbF, and LiF-BeF2 and etc [5].

According to H. M Glen [6], alpha-gamma hot cell is an enclosed space, in which radioisotopes are handled safely. The enclosing structure is a gamma-ray shield and is sealed to protect the exterior area from alpha contamination [6]. Master slave manipulator will be used to handle the processing or experimental activities remotely. The equipment setup and maintenance are done partially and manually by personnel entering into the hot cell once it is confirmed that hazardous/ radioactive material has been removed. In this study, Alpha-gamma shielding hot cell type with interlocking system design is selected.

2. Methodology

2.1. Radioactive handling capacity

The expected radioactive handling capacity value of hot cell is defined based on the literature of other hot cell designs. For example, the Process Mechanical Cell (PMC) and General-Purpose Cell (GPC) at West Valley Demonstration nuclear fuel reprocessing plant recorded a general dose rates of 100 to 300
R/hr up to 2000 R/hr hot spots [7]. In addition, other values from studies of RTP spent fuel are also taken into consideration as a future planning on research of RTP spent fuel inspection [8][9]. Thus, the value of 300 R/hr (equivalent to 227.5 Ci of Cobalt-60 ($^{60}\text{Co}$) and fission products of 50% burnup RTP 20wt% fuel type are used to simulate the hot cell shielding design. The target operating hour for the hot cell is 2000 hour per year. The design of hot cell will ensure that the hot cell operator is safe and the operator will not receive dose higher than 20mSv per year, following the Radiation Protection (Basic Safety Standards) Regulation 1988.

2.2. Shielding material

It is very important to select a suitable shielding material for hot cell walls, viewing window and door as it will determine the hot cell capability in handling high activity sources. The common shielding materials of hot cell wall are concrete, lead and steel. In JMTR hot laboratory, cells with different shielding materials are used to conduct different post irradiation examination (PIE) activities, accordingly [10]. The dismantling of irradiated capsules and examination of fuels are conducted in concrete cells. Therefore, concrete is selected as a shielding material for the hot cell wall as it is cheaper and easy to construct. The normal concrete density is about 2300 ~ 2400kg/m$^3$. The high density concrete will provide better radiation shielding based on the concrete radiation linear attenuation coefficient ($\mu$) value.

In order to design an interlock concrete wall, the suitable concrete grade should be selected according to the concrete grade application. It is important as the concrete does not just act as hot cell wall shielding, but it must also be able to withstand the load in interlocking concrete design of the hot cell. Thus, concrete grade 40 with density of 2350kg/m$^3$ is selected as shielding material for the hot cell wall as it provides a good radiation shielding and is suitable for interlocking concrete design. Table 1 shows the concrete grade in construction in Malaysia [11]. Table 2 shows the linear attenuation coefficient of different concrete grade [12].

| Grade No. | Application            |
|-----------|------------------------|
| 7~10      | Concrete without reinforcement |
| 15        | Lightweight concrete    |
| 20~25     | Normal concrete         |
| 30        | Post stressed concrete  |
| 40~60     | Pre-stressed concrete   |
| 70~100    | High performance concrete |

### Table 1. Concrete Grade in Construction.

| Radioactive material | Linear attenuation coefficient (cm$^{-1}$) |
|----------------------|------------------------------------------|
| Co 60                | Grade 15 | Grade 20 | Grade G25 | Grade G30 | Grade G35 | Grade G40 |
|                      | 7.78     | 7.62     | 6.79      | 6.73      | 7.22      | 7.0       |
| Cs 137               | 25.87    | 25.31    | 22.57     | 22.36     | 23.99     | 23.26     |

The shielding window provides a short observer - to - object distance which permits good depth perception. There are two types of transparent viewing windows - solid type and liquid filled type [13]. The solid type window such as lead glass and phosphate glass are susceptible to discoloration due to irradiation. In addition, solid type is heavy and brittle, causing difficulties on mobility. Thus,
liquid shielding is selected for hot cell with interlocking wall. Liquid shielding provides advantages such as easy to be replaced, recovered and transported. However, extra care is needed to prevent leakages. After considering the cost and related problems on handling liquid shielding material, colourless zinc bromide (ZnBr₂) 77% w/w Nuclear Optical Grade with density of 2.55 g/ml @ 20°C is chosen as the most practical shielding liquid for viewing window.

Lead door is selected for the hot cell interlocking wall design because of its easiness of handling and the size of the door. The door is designed with lead and layered with boronated HDPE which act as gamma and neutron shielding, respectively. The lead density is 11.35g/cm³ and boronated HDPE density is 1.08g/cm³. Boronated HDPE is used in the design of hot cell door for future upgrading as neutron shielding in the hot cell. The attenuated intensity of the gamma radiation that passes through a shielding material can be quantified according to the Lambert’s law:

\[ I = I_0 e^{-\mu r} \]

where \( I_0 \) is the incident radiation intensity, \( I \) is emerged radiation intensity that passes through the shielding material of thickness \( r \) and \( \mu \) is the linear attenuation coefficient. Based on the calculation, the minimum required concrete wall thickness for 227.5 Ci of Cobalt-60 \(^{60}\text{Co}\) source activity is 1-meter thickness. The door thickness is 250mm with 200mm lead thickness and 50mm boronated HDPE.

2.3. Hot cell design and limitation

The containment size of the hot cell is 2.4 meter x 1.9 meter x 2.825 meter (height). The size is suitable for research activities of the designed hot cell. The space available in the reactor hall is about 6-meter square area. The overhead crane available in the RTP hall is 3 tonnes. By considering safety factor of lifting: 2.0, the maximum weight of each interlocking concrete shall not be more than 1.5 tonne. In addition, lifting plug at each interlock concrete shall be able to withstand the concrete weight during lifting.

During fabrication of the interlock concrete, the lifting plug shall be embedded between reinforced steel bars (rebar) in the concrete. The rebar position shall not be in line with shielding thickness as it will allow radiation streaming through the gap between rebar and concrete. Thus, deformed type of reinforcement steel bar is advised.

In order to limit the numbers of interlock concrete, the concrete is designed as big as possible. Thus, 1-meter concrete thickness is selected in designing the interlock concrete. Two layers of interlocking concrete wall are not advised as it will lead to a bigger number of interlock concrete. Curved brick-built wall design is selected as interlocking concrete design as it is much easier to produce concrete formwork using steel compared to chevron brick-built wall design. Overlapping layers of concrete is selected in designing the hot cell concrete roof. At the concrete roof, there is a top loading chute for source loading from transfer cask into the hot cell. This chute is designed based on the existing transfer cask in RTP. Figure 3 shows the conceptual hot cell design with interlocking concrete walls.

2.4. Monte Carlo N-Particle Transport Code (MCNP)

Detail radiation simulation is performed by using the MCNP5 Monte Carlo code. All shielding inputs are taken into account in the simulation as well as using ENDF/B-VII libraries. These inputs are used for all interactions in simulation. The 227.5 Ci \(^{60}\text{Co}\) has 2 high energy gammas, with a total of 1.6835x10^{13} gammas per second. The gamma intensity of 20wt% fuel type with burnup around 50% is 1.56x10^{14} gammas per second. This corresponds to the low energy gamma of 0.5MeV and below [14]. The doses are calculated by means of cell and mesh tally, with flux-to-dose conversion factors, using the data sets of ICRP-21. The defined model and source are shown in Figure 4. The dose-rates inside and outside of the concrete wall are calculated using two mesh tally cases. The first one is axial
direction (XZ view) and the second one is radial direction (XY view). The MCNP mesh tally results are normalized to the gamma intensity or neutron intensity from the source.

**Figure 3.** The conceptual hot cell design with interlocking concrete wall.

**Figure 4.** The MCNP model (a) The defined 227.5Ci of Co60 gamma used as source-term, (b) The defined fission product’s gamma used as source-term from RTP fuel; and the mesh tally specification.

### 3. Result and discussion

The overall size of hot cell is 4.4 meter x 3.9 meter x 3.825 meter (height). There are 89 pieces of curve brick concrete with 25 different types of shape, designed for the hot cell wall. The wall thickness is 1 meter, with curve interlocking concrete design as shown in Figure 5. In addition, there are 41 pieces of 24 different types square shape concretes, designed for the hot cell roof. The
maximum weight of one piece of concrete is 1.25 tonnes. Thus, each concrete lifting plug is designed for lifting load of up to 2.25 tonnes weight.

![Figure 5. Curve brick interlock concrete.](image1.png)  
![Figure 6. Concrete roof.](image2.png)

The results of the radiation simulations are shown in Figure 7 to Figure 10, the 1-meter concrete shield shows that it is capable of reducing the dose-rate from the inside the wall to the outside surface. The dose-rate at the outer surface of the concrete wall is reduced approximately more than 10 million times when compared to the inner wall surface, the reduced values are 12.1 μSv/hr for gamma source and 0.65 μSv/hr for RTP fuel fission product source. Similar reduction of dose rate is observed at the ZnBr₂ windows, both for gamma and RTP fuel fission products. The highest dose rate value in front of ZnBr₂ window for gamma and RTP fuel fission product are 10.1 μSv/hr and 2.04 μSv/hr, respectively. However, the result shows that a relatively high radiation leaked through the door. The highest dose rate, located at lead door are 1570 μSv/hr and 489 μSv/hr for 227.5 **60**Co source and RTP fuel fission product, respectively.

![Figure 7. XY axis view, dose-rate map results for 227.5Ci Co60 gamma source.](image3.png)  
![Figure 8. XZ axis view, dose-rate map results for 227.5Ci Co60 gamma source.](image4.png)
Scattered radiation are observed at the left and right side and also at the upper and bottom side of the door. The low energy gamma, dominant from fission products seems to be stopped by the door. However, the scattered gammas, even the low energy gammas could also escape through the left, right, top or bottom side. Therefore, additional shielding or modification is needed in the future especially at the gap between interlock concrete and the lead door in order to further reduce the dose rate.

4. Conclusion

Through the evaluation, the design of interlocking concrete wall with 1-meter thickness are able to act as a shielding for 227.5Ci $^{60}$Co and fission products of 50% burnup RTP 20wt% fuel. However, a set of measurement data is still needed to verify the percentage of discrepancy for the simulation in future. This study proves that the hot cell with interlocking alpha-gamma shielding design, installed in controlled area or RTP Hall is fit to handle research activities.

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