1106/MD18085/REP/01

Radioactive Materials Management

Deep Geologic Repository Conceptual Design

Annex 3

Shielding Calculations

December 2002

NOTICE to the Reader

"This document has been prepared by CTECH Radioactive Materials Management, a joint venture of Canatom NPM Inc. and RWE Nukem Ltd. ("Consultant"), to update the conceptual design and cost estimate for a deep geologic repository (DGR) for long term disposal of used nuclear fuel. The scope is more fully described in the body of the document. The Consultant has used its professional judgment and exercised due care, pursuant to a purchase order dated October 2001. (the "Agreement") with Ontario Power Generation Inc. acting on behalf of the Canadian nuclear fuel owners ("the Client"), and has followed generally accepted methodology and procedures in updating the design and estimate. It is therefore the Consultant's professional opinion that the design and estimate represent a viable concept consistent with the intended level of accuracy appropriate to a conceptual design, and that, subject to the assumptions and qualifications set out in this document, there is a high probability that actual costs related to the implementation of the proposed design concept will fall within the specified error margin.

This document is meant to be read as a whole, and sections or parts thereof should not be read or relied upon out of context. In addition, the report contains assumptions, data, and information from a number of sources and, unless expressly stated otherwise in the document, the Consultant did not verify those items independently. Notwithstanding this qualification, the Consultant is satisfied that the updated conceptual design and cost estimate was carried out in accordance with generally accepted practices in a professional manner .

This document is written solely for the benefit of the Client, for the purpose stated in the Agreement, and the Consultant's liabilities are limited to those set out in the Agreement."

Summary

Scoping shielding calculations were required in support of the Deep Geologic Repository design, to ensure that the routine dose rate to an individual worker during normal operations is not more than 2 mSv/year. To achieve this, the dose rate at the working face of operating areas within the facility shall not exceed 1.0 μSv/h. Supporting shielding calculations detailed in this report demonstrate that this dose rate limit will not be exceeded, with the exception of handling the existing road transportation casks. The dose rate to operators from these casks will be mitigated by incorporation of local shielding and management procedures.

Contents

1 Introduction 1
2 Shielding Assumptions 1
3 Computer Codes 2
4 UFC Design 3
4.1 INITIAL CALCULATIONS 3
4.2 LATER UFC DESIGN 3
5 UFC Jacket and Emplacement Room Spacer/Shield Plug 4
5.1 JACKET AND SHIELD PLUG 4
5.2 BENTONITE CALCULATIONS 5
6 UFC Transport Cask 5
7 UFPP Bulk Shielding 6
7.1 STORAGE MODULES 6
7.2 UFPP FUEL MODULE SHIELDED TRANSFER TUNNEL 6
7.3 IRRADIATED FUEL DRY STORAGE BASKET UFPP RECEIPT BUFFER STORE 7
7.4 UFPP UFC RECEIPT CELL 7
8 UFPP Storage Pool Depth 7
9 UFPP UFC Shielded Cart 8
9.1 SHIELDED CART BULK SHIELDING 8
9.2 TRANSFER TUNNEL ROOF THICKNESS 8
9.3 SHIELDED CART SHINE CALCULATIONS 8
9.3.1 Shine From Top of UFC Within Shielded Cart 9
9.3.2 Shine From Partially Raised UFC 9
9.3.3 Shine From Fuel Baskets 9
10 UFPP Sealing Cell 9
11 DGR Emplacement Room Operator Dose 10
12 DGR Emplacement Room Shielding/Spacer Plug Cask 10
13 Dose Burden 11
13.1 DURING USED FUEL HANDLING/EMPLACEMENT 11
13.1.1 Basket Line 11
13.1.2 Module Line 12
13.1.3 UFC Lines 13

13.2 DOSE UPTAKE BY CONSTRUCTION WORKERS 15

14 Conclusions 15

15 References 15

16 Tables 17

16.1.1 Table 1 UO2 Fuel Spectra for 280 MWh/kgU, 30 years cooling 17

16.1.2 Table 2 Material Compositions 18

16.1.3 Table 3 Fuel bundle Composition 19

17 Figures 20

1 Introduction

Scoping shielding calculations were required in support of the Deep Geologic Repository (DGR) design. To ensure that the routine dose rate to an individual worker during normal operations is not more than 2 mSv/year the dose rate at the working face of operating areas within the facility should not exceed 1.0 µSv/h. Shielding assessments were performed to ensure that the plant design would meet this dose rate limit. The assessments included:

- Used Fuel Container (UFC) design including bentonite clay jacket and emplacement room spacer plugs
- UFC Transport Cask
- Bulk Shielding of the Used Fuel Packaging Plant (UFPP) including the storage pool
- UFPP UFC Shielded Cart
- DGR Emplacement Room
- End-Plug Cask
- Dose Burden.

2 Shielding Assumptions

The reference fuel for shielding calculations has a burnup value of 280 MWh/kgU and an age of 30 years [1]. The source terms for this reference were taken from Volume 3 of [2] and are given in Table 1. The model has not taken into account the effect of any fuel end plate. The fuel composition was based on data for a 37-element fuel bundle and was taken from Table 2 of [2]. The material compositions assumed are given in Tables 2 and 3 and were taken from [3 to 9].

Used fuel bundles enter the facility in road transport casks containing either CANDU irradiated fuel dry storage baskets or modules. The fuel is removed from the cask and then from the basket or module, placed in UFC baskets. The loaded UFC baskets are lowered into a UFC that is located within a shielded cart. The UFC is sealed and inspected, removed from the cart and placed within a bentonite clay jacket. The UFC and jacket are placed within a transport cask and transferred underground to an emplacement room for subsequent emplacement.

3 Computer Codes

The Monte Carlo code MCBEND [4] was used to perform the majority of gamma dose rate calculations. Results were verified using the RANKERN computer code [5]. MCBEND models the transport of individual particles accurately by using a fine energy group representation of nuclear data, but with the same flexible geometry modelling package used by the RANKERN code.

In effect, MCBEND simulates what happens in practice, and performs a numerical experiment of the system being analysed. In order to achieve reasonable accuracy with practical amounts of computing time, the code is provided with acceleration techniques. These effectively cause more particles, or fractions of particles to be tracked in the direction of interest, compared to directions of less importance. A simple adjoint diffusion calculation is used to define which directions and energies are important.

The point kernel RANKERN code was used to verify a number of the gamma dose rate calculations, as it provides an efficient way of determining the contribution to dose rates from scattered and reflected radiation. RANKERN starts its calculation by randomly sampling a point within the defined source and then performing a line of sight calculation to the defined dose point. The contribution from radiation that is initially travelling in a different direction, but is then scattered back toward the dose point is accommodated by the use of build-up factors, that are based upon the path length between the current source point and the dose point. Further random points within the source are sampled, so as to integrate over the source volume, until a calculated result with a predefined statistical uncertainty is achieved. RANKERN also includes the facility to define reflection surfaces, so that a two-stage calculation may be performed.

RANKERN and MCBEND have been successfully applied to such problems as:

- Design of nuclear plant
- Interpretation and analysis of measurements on operating plant and in experimental facilities
- Calculation of personnel dose levels and radiation induced material changes.

Both RANKERN and MCBEND have been developed by the ANSWERS software service which acts as a centrally controlled repository for all the major computer codes and data libraries used in the areas of criticality, shielding and reactor physics in the UK. The ANSWERS Service employs a comprehensive set of software management QA procedures, covering the entire software life-cycle, in the development and validation of its software. The Quality Management System provided by these procedures has been certified against the International Standard ISO 9001.

In all but one case dose rates were calculated using ICRP51 conversion factors for gammas and ICRP60 for neutrons. The exception was the UFC design assessment where the dose rate was required in Grays per hour (see Section 4).

4 UFC Design

4.1 INITIAL CALCULATIONS

The UFC was modelled as a carbon steel (density 7.86 g/cc) cylinder of outer diameter 1116 mm and height 3712 mm. The height of the UFC was later changed to 3708 mm but this does not affect the results of these calculations. The internal dimensions were 924 mm and 3512 mm respectively. The internal dished ends were neglected for computational ease and pessimism. The steel cylinder was housed in a copper jacket with a gap of 1 mm between copper (density 8.96g/cc) and steel at the sides and 2 mm at the top. There was no gap at the base. The copper was modelled as 25 mm thick at the sides and the base and 31.8 mm thick at the top.

The basket was modelled as 55 carbon steel tubes, each with a wall thickness of 2.5 mm, on a 110 mm pitch and each with an outer diameter of 110 mm. The centremost tube was modelled as empty giving a total of 54 fuel bundles per basket. The six baskets (later changed to three baskets) were stacked with a nominal 10 mm gap between them. The model for the fuel bundles was based on details taken from Table 4.1 of [10]. The fuel bundles were modelled as cylinders of height 500 mm and diameter 102.5 mm.

The gamma and neutron dose rates were calculated using the MCBEND computer code. The dose rates were calculated at contact with the sides and base of the UFC. Dose rates were not calculated above the lid as the shielding there was greater than at the base.

The results were initially calculated in Sv/h. The following assumptions were made to convert Sv/h to Gy/h. For the gamma dose rate the result in Sv/h (calculated using ICRP51) is broadly equivalent to Gy/h, so no conversion was required. For the neutron dose rate, 1.0 Sv/h (calculated using ICRP21) is approximately equivalent to 0.1 Gy/h, so the results were reduced by a factor of ten. The results of the calculations are given in the Table below. It can be seen that the total dose rate is considerably less than the limiting value of 15 Gy/h. Secondary gammas were not calculated for these preliminary scoping calculations as they were expected to be trivial compared to the primary gamma dose rates.

Results

Dose Point	Gamma Dose Rate Gy/h	Neutron Dose Rate Gy/h	Total Dose Rate Gy/h
Side of UFC	0.038	2.3E-5	0.038
Base of UFC	0.034	1.6E-5	0.034

4.2 LATER UFC DESIGN

Following completion of the preliminary scoping calculations, described in the previous section, the UFC was redesigned. The new design is similar to the old, with the exception that the thickness of the ends of the steel container was increased from 100 mm to 159 mm in the new design and the thickness of the copper lid was decreased from 31.8 mm to 25 mm. It was not necessary to repeat the above calculations because the increase in steel thickness meant that the initial calculations were pessimistic. The reduction in copper lid thickness does not affect the results as the original calculation was based on the 25 mm thick copper base. The fuel will now be stored in three taller UFC baskets rather than six shorter ones, but this will not significantly affect dose rates in any way.

5 UFC Jacket and Emplacement Room Spacer/Shield Plug

5.1 JACKET AND SHIELD PLUG

The UFC and its contents were modelled as described previously in Section 4. Initial calculations were carried out by modelling the proposed UFC jacket as a cylinder of buffer material 250 mm thick around the copper body of the UFC. The calculational model is shown in Figure 1. The composition of the buffer material is given in Table 2 and was taken from [3]. Dose rates were calculated at contact with the side of the 250 mm of buffer and the dose rates are summarised in the table below.

The UFC spacer/shield plug, positioned at the end of a UFC located within an emplacement room, was modelled as a buffer cylinder placed adjacent to the UFC copper lid. Dose rates were calculated at contact with the exposed surface of the buffer plug. 780 mm of buffer material was required to reduce the contact dose rate too less than 1.0 μSv/h. Neutron and secondary gamma contributions to the total dose rate were negligible.

To help ensure that the emplaced UFC temperature constraints were not exceeded, a UFC jacket end thickness of 256.5 mm was proposed followed by a buffer shield plug thickness of 750 mm. From the above shielding calculation results, it is evident that the proposed overall jacket/plug thickness will provide adequate shielding.

Jacket Results

	Gamma μSv/h	Neutron μSv/h	Secondary Gamma μSv/h	Total μSv/h
Radial dose rate after 250mm bentonite	1.35E3	3.71E0	1.82E0	1.35E3

5.2 BENTONITE CALCULATIONS

The initial UFC jacket calculations reported in the Section 5.1 were performed using buffer material for the jacket i.e. 50% bentonite clay and 50% silica sand. 100% bentonite is now proposed instead. To quantify the effect of changing materials the calculations for the 250 mm thick buffer jacket were repeated using 250 mm of bentonite. It is intended to use Wyoming bentonite, but as the composition for this was not available, the composition for sodium bentonite was used instead, as a sensitivity study. The compositions are given in Table 2 and the results are summarised in the table below. The surface gamma and total dose rates for bentonite are less than half the equivalent buffer dose rates. The increase in the small neutron dose rate does not present any problems. It is concluded that any calculations performed using buffer material for the jacket rather than bentonite are likely to be pessimistic, assuming that the composition of Wyoming bentonite is similar to that of sodium bentonite.

Bentonite/ Buffer Results

Jacket Material	Bentonite	Buffer
Gamma surface dose rate μSv/h	4.91E2	1.35E3
Neutron surface dose rate μSv/h	5.09E0	3.71E0
Total surface dose rate μSv/h	4.96E2	1.35E3

6 UFC Transport Cask

Once a UFC has been fitted with a bentonite jacketed, the whole assembly is placed within a transport cask for transfer underground to the repository. To control operator dose the transport cask's surface dose rate is limited to less than 1.0 μSv/h. Calculations were performed to determine the amount of shielding, additional to the 250 mm UFC jacket, required to achieve this dose rate. The layers of additional shielding required were: 50 mm air, 50 mm steel, 50 mm polythene and 100 mm steel.

The radial surface dose rates for this amount of shielding are given in the table below. These dose rates were based on the use of a buffer jacket. A 100% bentonite jacket will, from the evidence given in Section 5, reduce the total dose rate further.

Transport Cask Results

Calculation	Gamma µSv/h	Neutron µSv/h	Secondary Gamma µSv/h	Total µSv/h
Radial dose rate after cask shielding	0.61	0.11	Neg.	0.72

7 UFPP Bulk Shielding

7.1 STORAGE MODULES

Calculations were required to derive a bulk shielding thickness for remote operation cell walls within the UFPP i.e. the basket receipt and cutting cells, the module receipt and fuel module/basket handling cells, such that the dose rate to an operator working outside the cells was less than 1.0 µSv/h. The worst case for the majority of cells was assumed to be two storage modules at a distance of 500 mm from the inside of the cell wall. This stand off distance will be maintained by physically limiting the travel of the cells overhead cranes. The calculational model was set up as described in the following paragraph.

For pessimism two modules were modelled side by side at a distance of 500 mm from the shield wall, assumed to be constructed from ordinary concrete with a density of 2.32 g/cc. The fuel bundles were modelled as cylinders of height 500 mm and diameter 102.5 mm. The bundle composition is shown in Table 3 and was taken from Table 2 of [2]. One module contains 48 tubes that are 1000 mm long and each of which hold two fuel bundles. The amount of concrete required to reduce the operating face dose rate to 1.0 µSv/h was then calculated. The composition used for the concrete is shown in Table 2. The calculational model is shown in Figures 2 and 3.

Dose rates were calculated at contact with the operating face of the shield wall. The amount of concrete shielding required to reduce the dose rate to less than 1.0 µSv/h was calculated to be 1100 mm. Neutron and secondary gamma contributions to the total dose rate, for this thickness of concrete, were negligible.

7.2 UFPP FUEL MODULE SHIELDED TRANSFER TUNNEL

The fuel module shielded transfer tunnel is intended to shield a single module while it is transferred from the fuel module receipt bay to the fuel module/basket handling cell. Calculations were performed to determine the amount of steel and polythene shielding required for the tunnel wall. Concrete was not used due to the limited space available. The tunnel wall was modelled at a distance of 450 mm from a single module as shown in Figure 4. Results showed that the shielding required to reduce the cold side dose rate to less than 1.0 μSv/h was 300 mm of steel followed by 100 mm of polythene and another 50 mm of steel. The dose rates for the transfer tunnel wall with this shielding are shown in the table below.

UFPP Fuel Module Shielded Transfer Tunnel Dose Rates

Primary Gamma Dose Rate (μSv/h)	Neutron Dose Rate (μSv/h)	Secondary Gamma Dose Rate (μSv/h)	Total Dose Rate (μSv/h)
0.56	0.28	0.05	0.89

7.3 IRRADIATED FUEL DRY STORAGE BASKET UFPP RECEIPT BUFFER STORE

The basket receipt cell buffer store can hold a maximum of 32 irradiated fuel dry storage baskets stored in a double stacked array of 16. Calculations were required to determine whether the proposed 1100 mm thick concrete walls provided enough shielding to give a cold side dose rate of less than 1.0 μSv/h. The calculational model is shown in Figure 5. The primary gamma dose rate on the cold side of the wall was calculated to be 0.63 μSv/h and the neutron dose rate was negligible. It is concluded that the wall will provide adequate shielding.

7.4 UFPP UFC RECEIPT CELL

The UFC receipt cell has two separate UFC holding bays; one for passed UFCs and the other for failed UFCs. The bays are some 14 m apart to prevent contamination. Each bay can hold up to ten UFCs that are stored in a vertical orientation. Calculations were required to determine the thickness of the concrete cell wall that would give a cold side dose rate of less than 1.0 μSv/h. The calculational model is shown in Figure 6. The thickness of concrete required to reduce the surface dose rate to less than 1.0 μSv/h was 700 mm.

8 UFPP Storage Pool Depth

The storage modules were modelled as outlined in Section 7, but with their overall dimensions increased to 1.0 m x 1.3 m x 0.65 m as indicated in Figure 44 of [11]. It was assumed that the centres of the uppermost bundles lay some 140 mm below the top of the module. The storage pool was modelled as a 6x6x6 array of modules covered by water as shown in Figure 7. The model also included a raised module in transit. The depth of water was required to reduce the pool surface dose rate to less than 1.0 μSv/h.

Dose rates were calculated at contact with the surface of the pool water. The gamma dose rate from the raised module was found to dominate and 2300 mm of water was required above the raised module to reduce the pool surface dose rate to less than 1.0 μSv/h. This equates to 3250 mm of water above the main array. If the module were to be raised any higher then the pool surface dose rate would exceed the limit. Neutron and secondary gamma contributions to the total dose rate were negligible.

9 UFPP UFC Shielded Cart

9.1 SHIELDED CART BULK SHIELDING

The calculational model for the UFC was based on [6 and 7]. The UFC was modelled within the shielded cart and the amount of radial steel/polythene shielding was adjusted until the radial surface dose rate was calculated to be less than 1.0 μSv/h. The calculational model of the shielded cart is shown in Figure 8.

The shielding required to reduce the surface dose rate to less than 1.0 μSv/h was 160 mm steel, 120 mm polythene, 50 mm steel, moving radially outwards as shown in Figure 8.

9.2 TRANSFER TUNNEL ROOF THICKNESS

The thickness of the transfer tunnel roof above the shielded cart inner vessel is required to provide sufficient shielding to give a surface dose rate of less than 1.0 μSv/h.

The calculational model is shown in Figure 9. It was assumed that the copper lid had not yet been put on the UFC but that the steel lid was in place. The top of the fuel was assumed to lie 564 mm below the top of the transfer cart i.e. 589 mm below the concrete transfer tunnel ceiling.

The thickness of concrete required to reduce the surface dose rate to 1.0 μSv/h above the transfer tunnel ceiling was 600 mm.

9.3 SHIELDED CART SHINE CALCULATIONS

These calculations are concerned with radiation “shining” through the 25 mm gap between the top of the cart and the transfer tunnel ceiling and then being reflected towards an operator standing in the tunnel. The gap is provided to allow the shielded cart to travel along the tunnel without coming into contact with the transfer tunnel ceiling.

9.3.1 Shine From Top of UFC Within Shielded Cart

The first set of calculations considered the effect of shine from the top of the UFC within a shielded cart, as the shielded cart is moved into position beneath an access port too a process cell above the transfer tunnel. As the cart has no lid there is the possibility of radiation from the top of the UFC reflecting off the concrete ceiling down towards an operator standing close to the cart. The calculational model is shown in Figure 9.

Calculated dose rates adjacent to the cart and level with the gap between the top of the cart and the ceiling were 11.0 μSv/h for gammas and 1.0 μSv/h for neutrons. Predicted dose rates at operator head height (at any distance from the cart) were negligible.

9.3.2 Shine From Partially Raised UFC

The second set of calculations were concerned with radiation shine through the gap between the top of the cart and the ceiling whilst the UFC is partially raised up to the cell above (for example as it would be at the UFC inner vessel purging cell). The calculational model is shown in Figure 10. The model considered the effect of the radiation being scattered from the ceiling and walls of the transfer tunnel.

Calculated dose rates adjacent to the cart and level with the gap between the top of the cart and ceiling were high, 1.83E3 μSv/h for gammas and 12.5 μSv/h for neutrons. The gamma dose rate (due to radiation reflected from the transfer tunnel ceiling and walls) at operator head height was less than 0.4 μSv/h and the neutron dose rate was negligible.

9.3.3 Shine From Fuel Baskets

Finally the third set of calculations considered the same radiation shine as described in Section 9.3.2, but from baskets that were lowered into the UFC from the fuel module/basket handling cell. The calculational model is shown in Figure 11. The calculated gamma dose rate at operator head height was less than 2.0 μSv/h and again the neutron dose rate was negligible. As this dose rate is above the normal limit of 1.0 μSv/h, access in the transfer tunnel may require restricting whilst fuel is lowered into a UFC.

10 UFPP Sealing Cell

The UFPP sealing cell walls and ceiling are required to provide sufficient shielding to give an external surface dose rate of less than 1.0 μSv/h for the case when a UFC is partially raised into the cell.

The calculational model is shown in Figure 12. It was assumed that the steel and copper lids were removed from the UFC and that the top of the fuel and steel was flush with the cell floor as shown in Figure 12. All cell walls were assumed to be equidistant from the UFC centreline.

The dose point for the ceiling was situated directly above the open UFC. The wall dose points were taken vertically along one wall on the UFC centreline opposite the UFC floor opening. The ceiling thickness required to achieve a surface gamma dose rate of less than 1.0 μSv/h was 850 mm and the wall thickness was 750 mm. Neutron and secondary gamma dose rates were negligible.

11 DGR Emplacement Room Operator Dose

After the jacketed UFC (with shielding/spacer blocks in place at the end) is placed within the dense backfill/buffer material emplacement structure, the operator must insert the lower shielding blocks beneath the jacketed UFC. During this operation he will receive a dose due to radiation scattered from under the UFC. A barrier will be used to ensure that the operator does not approach the emplacement structure beyond a distance of 1.25 metres. Calculations were required to determine the dose rate to the operator during this procedure. Details of the calculations and assumptions are outlined below.

The calculational model was based on [8] and is shown in Figures 13 and 14. The end of the 1.00 m of shielding/spacer buffer blocks (actually 256.5 mm of bentonite followed by 750 mm of buffer material) was assumed to lie flush with the end of the dense backfill/buffer material emplacement structure as shown in Figure 13. For computational ease the hexagonal cross-section was modelled cylindrically as shown in Figure 14, this is a slightly pessimistic assumption. The minimum vertical distance between the jacket and the high performance concrete floor was 742 mm. The jacket material was modelled as buffer material and not bentonite that has superior shielding properties.

Primary gamma and neutron dose rates were calculated using the MCBEND computer code. The RANKERN computer code was also used to confirm the MCBEND results for reflected gamma radiation. The dose points were situated 1.25 metres from the end of the shielding/spacer blocks as shown in Figure 13. The peak primary gamma dose rate at that distance was 0.7 μSv/h and the neutron and secondary gamma dose rates were negligible.

12 DGR Emplacement Room Shielding/Spacer Plug Cask

After the jacketed UFC is placed in the emplacement room, the UFC transport cask is removed and a shielding/spacer plug cask is positioned next to the emplacement room shield wall gamma gate. Calculations were required to ensure that the shielding/spacer plug cask provided sufficient shielding to give an external dose rate of less than 1.0 μSv/h. The calculational model is shown in Figure 15. The emplacement room shield wall was assumed to have the same construction as the transport cask i.e. 100 mm steel, 50 mm polythene and 50 mm steel. During the transfer of the shielding/spacer plugs from the cask, both the cask and shield wall gamma gates are open. Therefore the subsequent space they took up was modelled as an air gap. The position of the shielding/spacer plugs within the cask varies, so for simplicity (and pessimism) the cask was modelled as being empty. Calculations determined that a 20 mm thick steel cask provided sufficient shielding to reduce the surface dose rate on the outside of the cask to less than 1.0 μSv/h.

13 Dose Burden

13.1 DURING USED FUEL HANDLING/EMPLACEMENT

The basis of design for the facility is such that the routine dose to an individual worker during normal operations is less than 2 mSv/year. This has been achieved by ensuring that throughout the facility, with the exception of handling the existing road transportation casks, an individual worker is not exposed to a dose rate greater than 1.0 μSv/h. Assuming a nominal time estimate for individual worker exposure of 2000 hours per year this gives a total dose of no more than 2 mSv/year. This section follows the used fuel route through the facility to demonstrate that, with the exception of handling the road transportation casks, the dose rate to any operator does not exceed 1.0 μSv/h.

The fuel arrives at the receipt area of the plant in either modules (approximately 90% of fuel volume) or irradiated fuel dry storage baskets (10%). As the modules form the bulk of the fuel, there are two lines dedicated to processing this container type and one line to baskets.

The modules are carried in road transportation casks; it is expected that the baskets will be transported in a similar type of cask. Historical operational data indicates that the contact dose rate of the road transport cask is 0.23 mSv/h which falls to 0.08 mSv/h at a distance of one metre. These dose rates are obviously not consistent with the design of the new facility and therefore present the potential for increased operator dose in this area.

The logistics study shown in Appendix A of the Design Report indicates a road transportation cask handling time of up to two hours from receipt of the cask on its transporter up to the point where it is handled remotely. In total 629 road transportation casks (plus basket casks which are assumed to be similar) are received per year and are handled over two separate shifts. Assuming that the operators handle the cask at a distance of one metre the total operator dose for handling the road transportation casks would be 50 mSv/year.

Should these flasks be used within the facility a number of mitigating measures should be employed such as: minimising occupancy times, increasing operator distance from the flask, use of localised shielding and diversification of operators.

13.1.1 Basket Line

The storage basket within its transport cask arrives at the transporter receiving/shipping area where the impact limiter is removed and the cask is placed on a cart. It is then taken to the cask cart decontamination cell where the cask is vented and the lid bolts removed. The cask passes through to the basket transport cask receipt cell where the cart scissor lift raises the cask to the basket receipt cell gamma gate.

All operations within the basket receipt cell are done remotely and the cell has 1100 mm thick walls that are sufficient to give a cold side surface dose rate of less than 1.0 μSv/h as shown in Section 7.1. Hence operators outside the cell will receive a dose rate of less than 1.0 μSv/h. Once within the cell, a double-lidding system is used to remove the cask lid while keeping it free of contamination, the basket is then removed and placed in the buffer store. The shielding provided by the basket receipt cell buffer store walls is described in Section 7.3. The cask lid is replaced, the empty cask is lowered back to the basket transport cask receipt cell where the bolts are replaced; the cask and cart are monitored and decontaminated if necessary.

The storage basket is removed from the buffer store and passes through a shield door to the basket cutting cell. This is also a remote operations cell with a wall thickness of 1100 mm. In this cell the top of the storage basket is removed and placed into a skip for subsequent disposal. Used fuel bundles are removed remotely from the now open storage basket and placed individually within a waiting UFC basket. The empty storage basket is placed in a skip for subsequent disposal, while full UFC basket is transferred through a shield door to the Fuel Module/Basket Handling Cell (FMBHC).

The FMBHC is a remote operations cell with 1100 mm thick walls. Here the UFC basket containing used fuel bundles is held pending transfer to a UFC. Empty UFC baskets are lifted into the FMBHC from the new UFC basket delivery tunnel. The basket delivery facility is an area that will be considered in the detailed design stage and is not expected to present any shielding weakness. Design solutions will be required to prevent contamination leaving the FMBHC.

At this point the waste skip in the basket cutting cell will contain the remains of the storage basket. Once full, the waste skip is lowered via a gamma gate to the used cask basket size reduction/drumming/export area where it is processed so that it is suitable for export.

13.1.2 Module Line

The module within its transport cask arrives at the transporter receiving/shipping area where the impact limiter is removed and the cask is placed on a cart. It is taken to the cask cart decontamination cell where the cask is vented and the lid bolts removed. The cask passes through to the module transport cask receipt cell where the cart scissor lift raises the cask to the basket receipt cell gamma gate.

All operations within the module receipt cell are done remotely and the cell has 1100 mm thick walls. Once within the cell a double-lidding system is used to remove the cask lid while keeping it free of contamination and the module is removed. Following the loading of an empty module (see below) the cask lid is replaced and the cask is lowered back to the module transport cask receipt cell where the bolts are replaced. The cask and cart are monitored and decontaminated if necessary.

The module has two possible routes from the module receipt cell. If temporary storage is required it will go to the inlet pool, having been cleaned if necessary, prior to placement within the storage pool.

The dose rate at the surface of the module buffer storage pool is less than 1.0 µSv/h (see Section 8) so that an operator working above the pool is protected. The transfer of the module from the module receipt cell to the storage pool will be considered at the detailed design stage. Design solutions will be required to reduce the potentially high transient dose rate to the operator at this point.

If it is to be processed immediately, the module passes along the fuel module shielded transfer tunnel (see Section 7.2) to the FMBHC.

The FMBHC is a remote operations cell with 1100 mm thick walls as already discussed in the previous section. Within this cell the fuel is removed from the module ready for placement in a UFC basket. The empty module is returned to the fuel module receipt cell. The module is decontaminated, dried and then replaced in a cask ready for return to the customer.

At this point in the process the fuel from either a basket or a module is in the FMBHC and has been placed in one of three UFC baskets ready to be loaded into the UFC. Two identical lines leave the FMBHC, one at each end of the cell.

13.1.3 UFC Lines

The UFC shielded cart stands in the shielded cart transfer tunnel under the UFC basket export port, where fuel baskets are lowered into it. Supporting calculations have shown that dose rates to an operator standing in the transfer tunnel, during this operation, would be just less than 2.0 µSv/h (Section 9.3.3). Therefore, man access at this point may require restriction while fuel is being placed into the cart. Once the baskets are in place the steel lid is put on the UFC inner vessel.

The cart shielding has been designed to give a surface dose rate of less than 1.0 µSv/h so that it is possible for an operator to approach a cart to facilitate remedial work if necessary. As the cart has no lid there is the possibility of radiation from the top of the UFC reflecting down from the concrete ceiling towards the operator. However, calculations (see Section 9.3.1) demonstrated that dose rates at operator level were negligible.

The ceiling of the shielded cart transfer tunnel is 600 mm thick. Supporting calculations in Section 9.2 have shown that this reduces the dose rate above the tunnel too less than 1.0 µSv/h. This enables operators to enter cells above the transfer tunnel to carry out maintenance operations whilst the shielded cart is positioned below.

The transfer cart takes the UFC along the transfer tunnel to the transfer ports for a series of cells. At each port the UFC is lifted so that the top of the UFC enters the cell above. The dose rate at operator height in the tunnel below is less than 1.0 µSv/h, provided that the top of the fuel within the UFC does not pass higher than the floor of the cell (See Section 9.3.2).

At the sealing cell the UFC is purged and the copper lid fitted. The lid is welded at the UFC lid welding cell with the weld checked at the UFC inspection cell. The thickness of the walls and ceilings for these cells was shown by calculation (See Section 10) to give external dose rates of less than 1.0 μSv/h.

After inspection the UFC is lifted remotely into the UFC receipt cell. If it has passed inspection it is stored in the “passed” UFC holding bay, otherwise it goes to the “failed” UFC holding bay. Separate grapples are used to handle passed and failed UFCs to prevent cross contamination. The cell has the facility to decontaminate and dry UFCs if necessary. All operations within the UFC receipt cell are done remotely and the cell has sufficiently thick walls to give a cold side surface dose rate of less than 1.0 μSv/h as shown in the supporting calculation (see Section 7.4).

Failed UFCs are returned via the transfer tunnel to the failed UFC lid removal cell where the lid is removed. Then the UFC moves back under the FMBHC port and the fuel lifted back into the cell to be placed later into a new UFC. The failed UFC goes to the shielded cart swabbing/decontamination and UFC loading area where it is cleaned and removed.

Passed UFCs are lowered into the UFC jacketing dispatch cell. Man access is permitted for this cell to put the UFC jackets in place on the support frame, but not when a UFC is in the cell. The UFC is lowered into the open support frame that is then closed around the UFC. The jacket end cap is put in place and the support frame rotated to a horizontal position. At this point the upper portion of the support frame is raised to allow the jacketed UFC to be transferred on a roller bed into the UFC transport cask via a gamma gate. The shielding provided by the transport cask has been designed to give a surface dose rate of less than 1.0 μSv/h (see Section 6). Therefore, man access is acceptable for any operation while the UFC is within the cask.

The transport cask is transferred to the waste shaft and lowered underground for onward movement to an emplacement room. The cask is then connected to a gamma gate at the emplacement room shielded wall. Following transfer from the cask to behind the shield wall, the jacketed UFC is placed into its final location within the emplacement room. The emplacement room shield door and gamma gate will be designed to provide the same shielding as the transport cask so that the dose rate throughout the operation will be less than 1.0 μSv/h.

The transport cask is removed and the shielding/spacer plug cask containing the buffer plugs is put in its place. The shielding provided by the shielding/spacer plug cask has been designed to give a surface dose rate of less than 1.0 μSv/h (see Section 12). The shielding/spacer plugs are added to the end of the UFC. The thickness of these shielding/spacer plugs is greater than that required by the supporting shielding calculations (see Section 5.1) for UFC spacing reasons.

Finally the remaining lower sealing material blocks are inserted beneath the jacketed UFC. Supporting calculations demonstrate that operator dose rate is below 1.0 μSv/h if the operation is carried out 1.25 m from the face of the emplacement structure (see Section 11).

13.2 DOSE UPTAKE BY CONSTRUCTION WORKERS

It is the current intention to excavate each new section of the emplacement facility while emplacement is taking place in a previously constructed section. The emplacement facility is designed such that workers will only receive a negligible dose. This is achieved in several ways:

- At all times workers are shielded from the active emplacement facility by 30 m of granite
- The underground facility is designed to be clean so there is no risk to workers from active contamination. This is achieved by having separate ventilation systems for both active and construction operations.

Procedures will be in place to ensure that personnel do not inadvertently enter the wrong area. Should accidental entry occur the maximum dose rate (by design) to the worker would be less than 1.0 μSv/h.

14 Conclusions

The facility has been designed such that the routine dose to an individual worker during normal operations is no more than 2 mSv/year. This is achieved by ensuring that throughout the facility an individual worker is not exposed to a dose rate greater than 1.0 μSv/h. The exception to this is the handling of existing road transportation casks. The dose rate to operators from these casks will be mitigated by incorporation of local shielding and management procedures, should a better design not be incorporated. The supporting shielding calculations detailed in this report demonstrate that this dose rate limit is not exceeded, except under non-routine operations and when handling the existing road transport casks.

15 References

1 Technical Specification for Updating the Conceptual Design and Cost Estimate for a Deep Geological Repository for Used Nuclear Fuel. 06819-UFM-03789-00010-R00.

2 Characteristics and Radionuclide Inventories of Used Fuel from OPG Nuclear Generating Stations. 06819-REP-01200-10029-R00-Vol 1.

3 Technical Specification for a Screening-Level Study to Select Preferred Used Fuel Disposal Container Geometries and Capacities. 06819-TS-01110-10000-R00.

4 MCBEND. A Monte-Carlo Program for General Radiation Transport Solutions. User Guide to Version 9E. ANSWERS/MCBEND(94)15.

5 RANKERN. A Point-Kernel Program for Gamma-Ray Transport Solutions. User Guide for Version 14. ANSWERS/RANKERN(95)03.

6 Figure 7 of Main Design Update Report – Assembly of Used Fuel Baskets and Container.

7 Figure 2 of Main Design Update Report – 108 Bundle Used Fuel Basket.

8 Figure1 of Main Design Update Report – DGR Emplacement Room showing General Arrangement of Emplaced Blocks and UFCs.

9 P. Baumgartner, 2000. Elemental Composition of Disposal Vault Sealing Materials. Report 06819-REP-01300-10015-R00, AECL, Toronto, Ontario.

10 Deep Geological Repository Facility and Packaging Requirements. 06819-PR-01110-10000-R01

11 Engineering for a Disposal Facility Using the In-Room Emplacement Method. AECL-11595, COG-96-223.

16 Tables

16.1.1 Table 1 UO2 Fuel Spectra for 280 MWh/kgU, 30 years cooling

Combined Photon Spectra (fission products and actinides)		Combined Neutron Spectra (Alpha-n and spontaneous fission)	
Mean Energy (MeV)	Photons/s/kgU	Boundaries (MeV)	Neutrons/s/kgU
5.500	9.50E+01	17.300-14.200	0.0
4.750	1.05E+02	14.200-12.200	1.022E+00
4.250	1.82E+02	12.200-10.000	8.923E+00
3.750	3.14E+02	10.000-8.610	1.377E+01
3.250	5.42E+02	8.610-7.410	4.372E+01
2.800	7.34E+05	7.410-6.070	1.322E+02
2.400	7.48E+03	6.070-4.970	2.638E+02
2.000	1.01E+07	4.970-3.680	7.372E+02
1.580	2.74E+08	3.680-3.010	8.553E+02
1.120	7.92E+09	3.010-2.730	5.007E+02
0.650	6.36E+11	2.730-2.470	5.507E+02
0.300	2.60E+10	2.470-2.370	2.306E+02
0.170	2.53E+10	2.370-2.350	4.579E+01
0.120	2.81E+10	2.350-2.230	2.667E+02
0.085	3.91E+10	2.230-1.920	7.694E+02
0.055	8.58E+10	1.920-1.650	6.976E+02
0.030	1.64E+11	1.650-1.350	8.467E+02
0.010	3.62E+11	1.350-1.000	1.066E+03
		1.000-0.821	5.232E+02
		0.821-0.743	2.469E+02
		0.743-0.608	4.372E+02
		0.608-0.498	3.492E+02
		0.498-0.369	4.027E+02
		0.369-0.297	2.092E+02
		0.297-0.183	9.715E-01
		0.183-0.111	6.141E-01
		0.111-0.067	9.450E-02

16.1.2 Table 2 Material Compositions

	Sodium Bentonite 2.703 g/cc	Buffer 1.954 g/cc	Dense Backfill 2.226 g/cc	High Performance Concrete 2.429 g/cc	Ordinary Concrete 2.32 g/cc
Element	Weight %	Weight %	Weight %	Weight %	Weight %
Si	29.5	32.6	29.6	37.6	15.5
Al	8.4	3.6	7.5	3.3	1.4
Fe	3.0	1.3	2.4	0.9	1.0
Na	1.6	0.7	2.2	1.4	
Ca	1.3	0.6	1.4	2.3	25.7
Mg	1.3	0.6	0.7	0.2	
K	0.5	0.2	3.1	1.7	
C	0.4	0.2	0.4	0.3	6.0
Ba	0.4	0.2			
S	0.3	0.1		0.1	2.0
Ti			0.2		
P			0.1		
Mn					
H	0.8	2.0	0.8	0.5	4.0
O	52.5	58.0	51.5	51.8	49.8

Data for Sodium Bentonite taken from [9].

Data for buffer, dense backfill and high performance concrete taken from [3].

Data for ordinary concrete taken from [4].

16.1.3 Table 3 Fuel bundle Composition

Fuel Bundle 5.85 g/cc	
Component	Volume Proportion
Uranium dioxide (10.6 g/cc)	0.501
Zircalloy (6.55 g/cc)	0.082
Void	0.417

Data taken from [2]

17 Figures

Figure 1 Calculational Model of UFC and Jacket (Not to Scale)

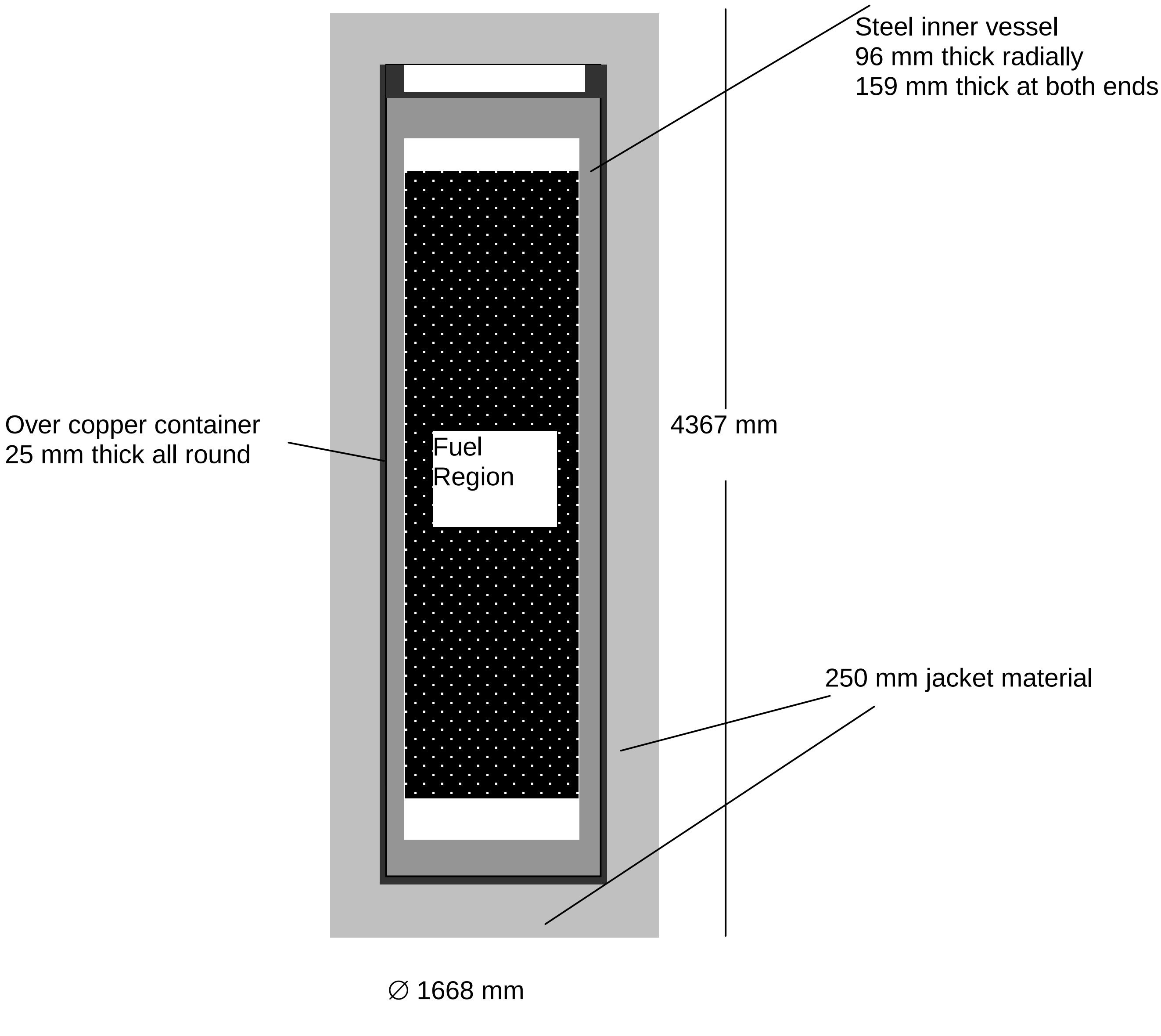


Figure 2 Calculational Model of Storage Modules (Not to Scale)

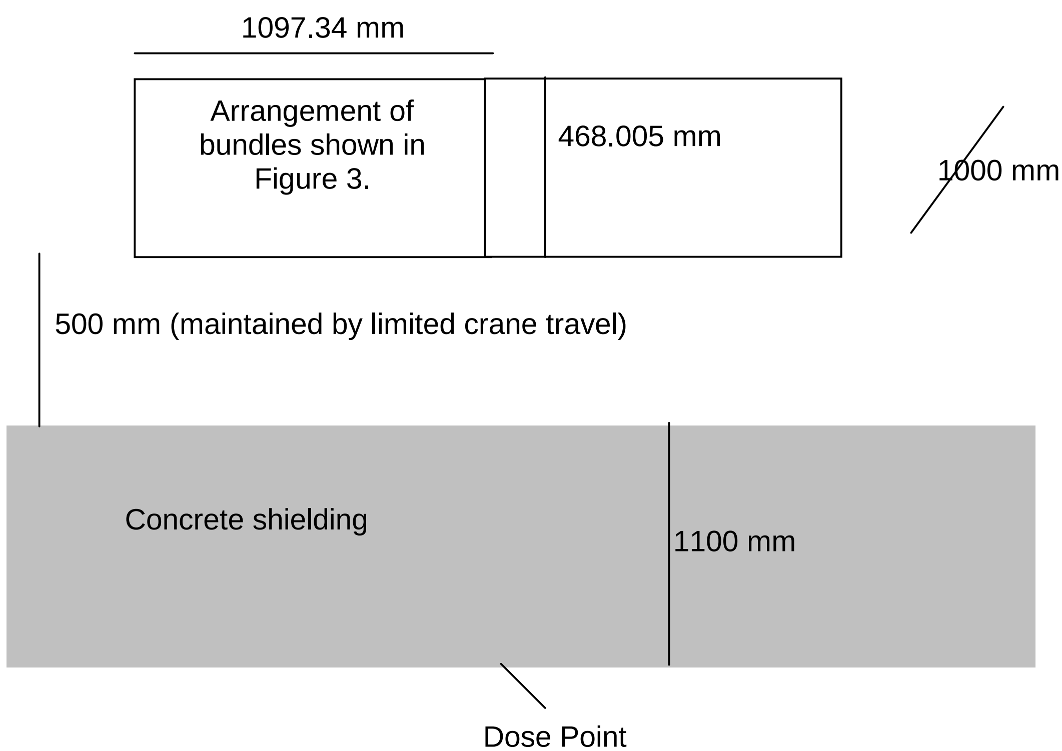


Figure 3 Arrangement of Fuel Bundles within Module (Not to Scale)

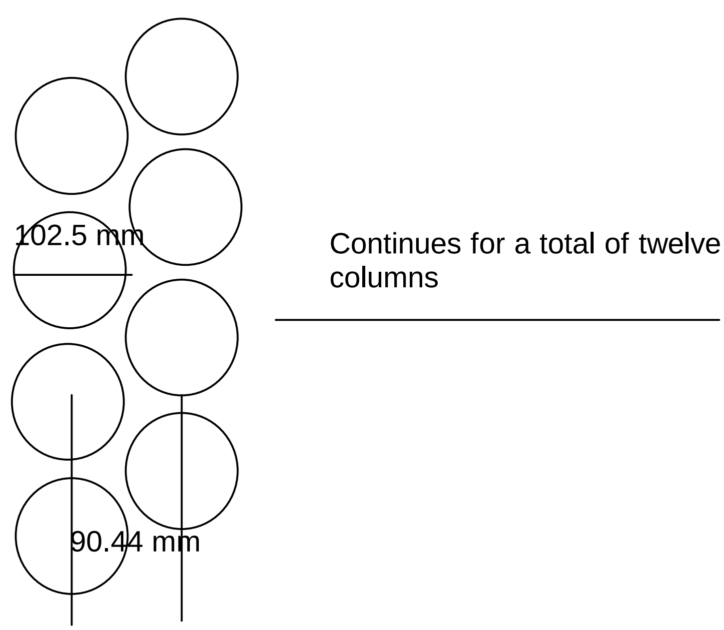


Figure 4 Calculational Model of Fuel Module Shielded Transfer Tunnel (Not to Scale)

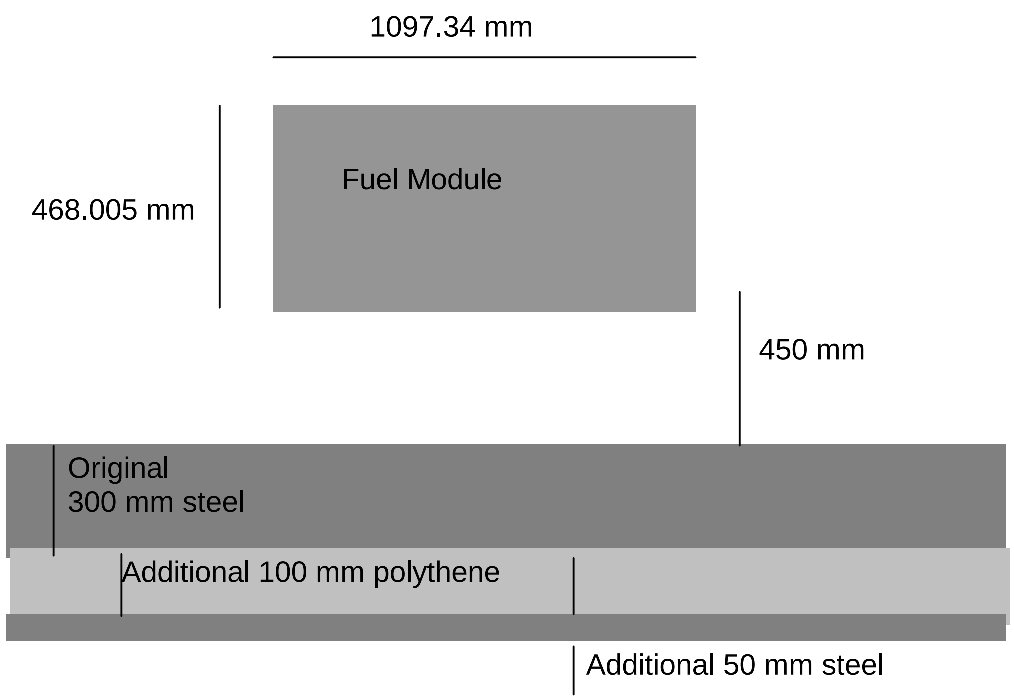
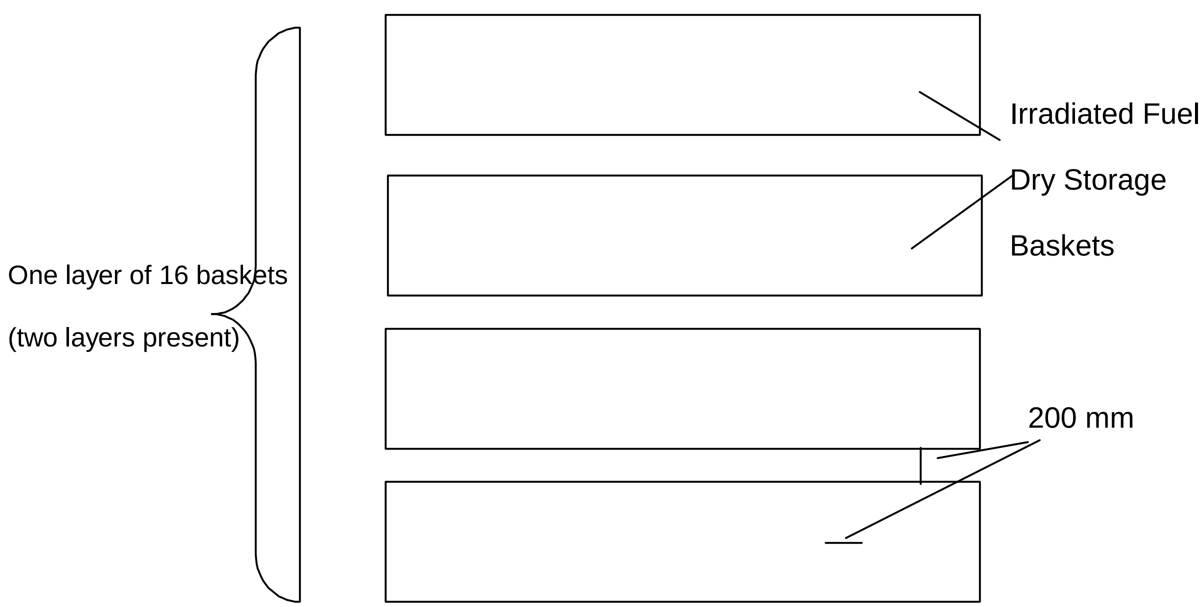


Figure 5 Calculational Model of Basket Receipt Cell Buffer Store (Not to Scale)



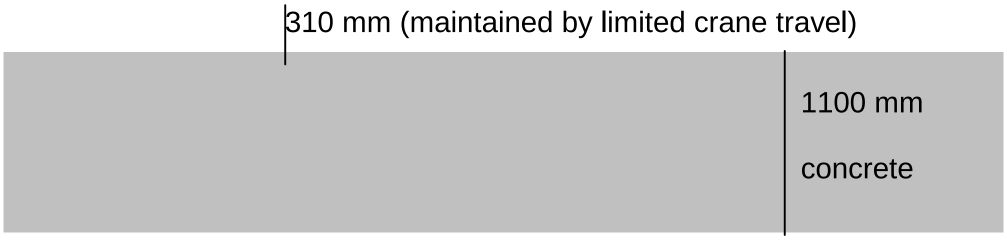


Figure 6 Calculational Model of UFC Receipt Cell (Not to Scale)

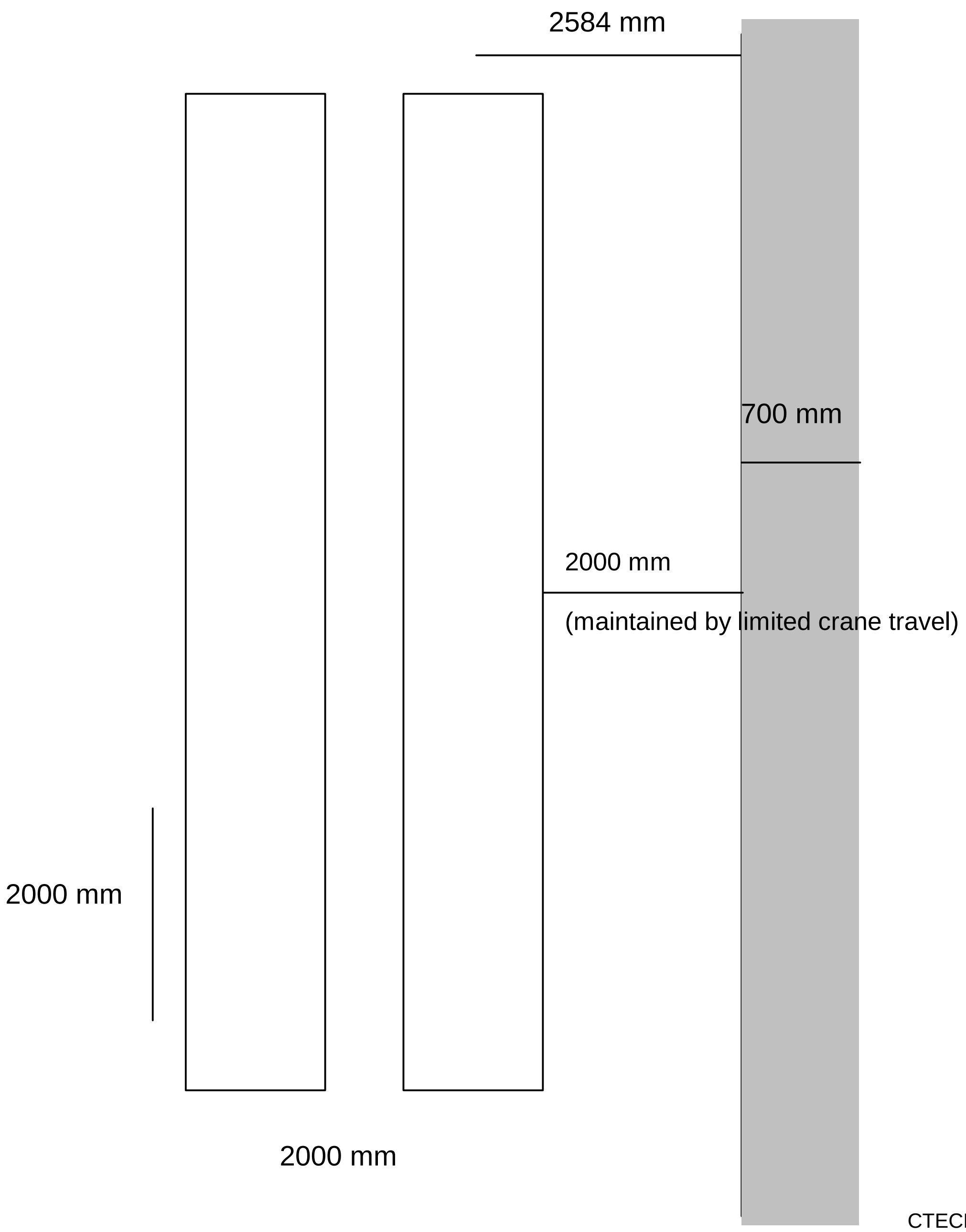


Figure 7 Calculational Model of Storage Pool (Not to Scale)

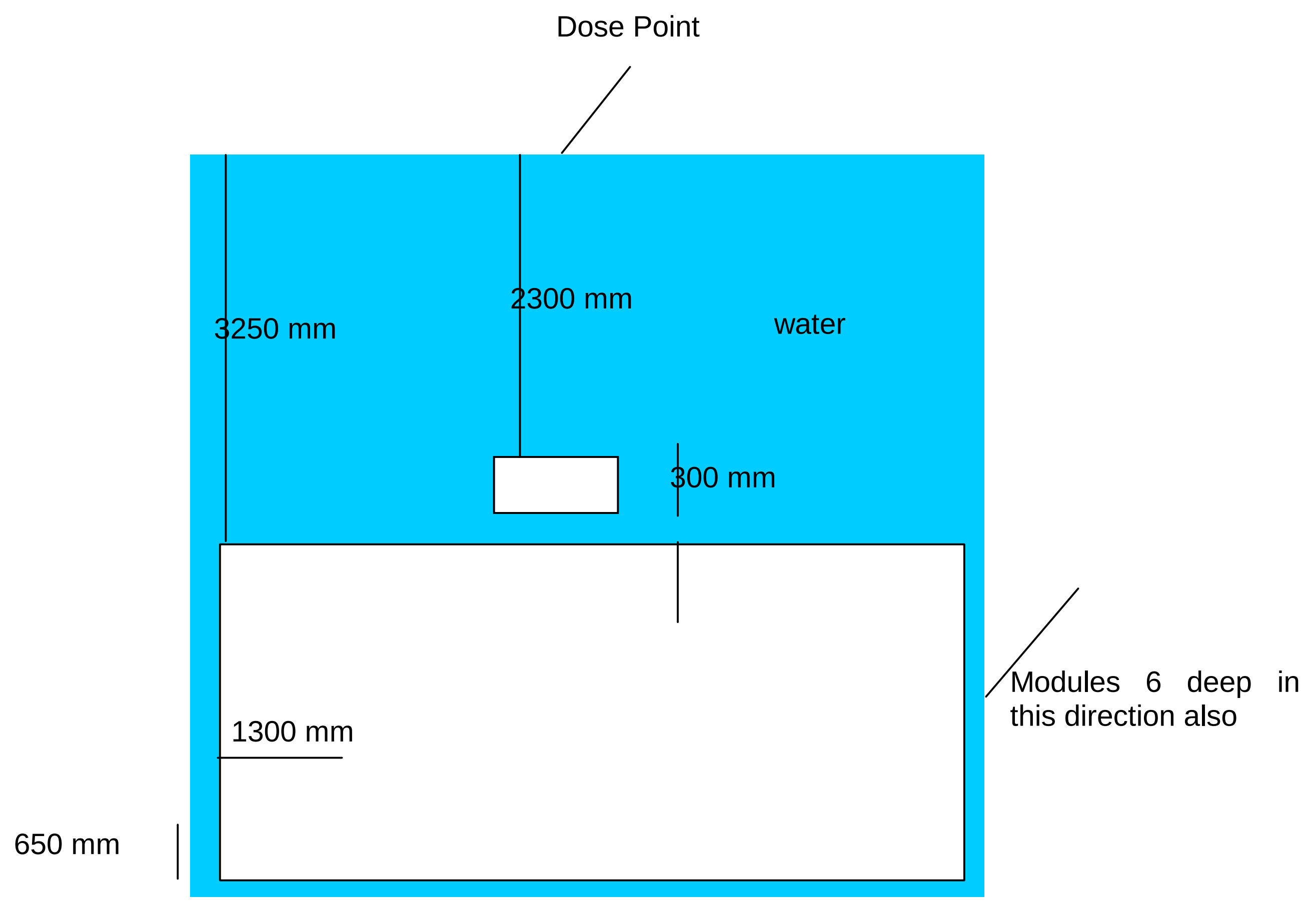


Figure 8 Section through Calculational Model of Shielded Cart (Not to Scale)

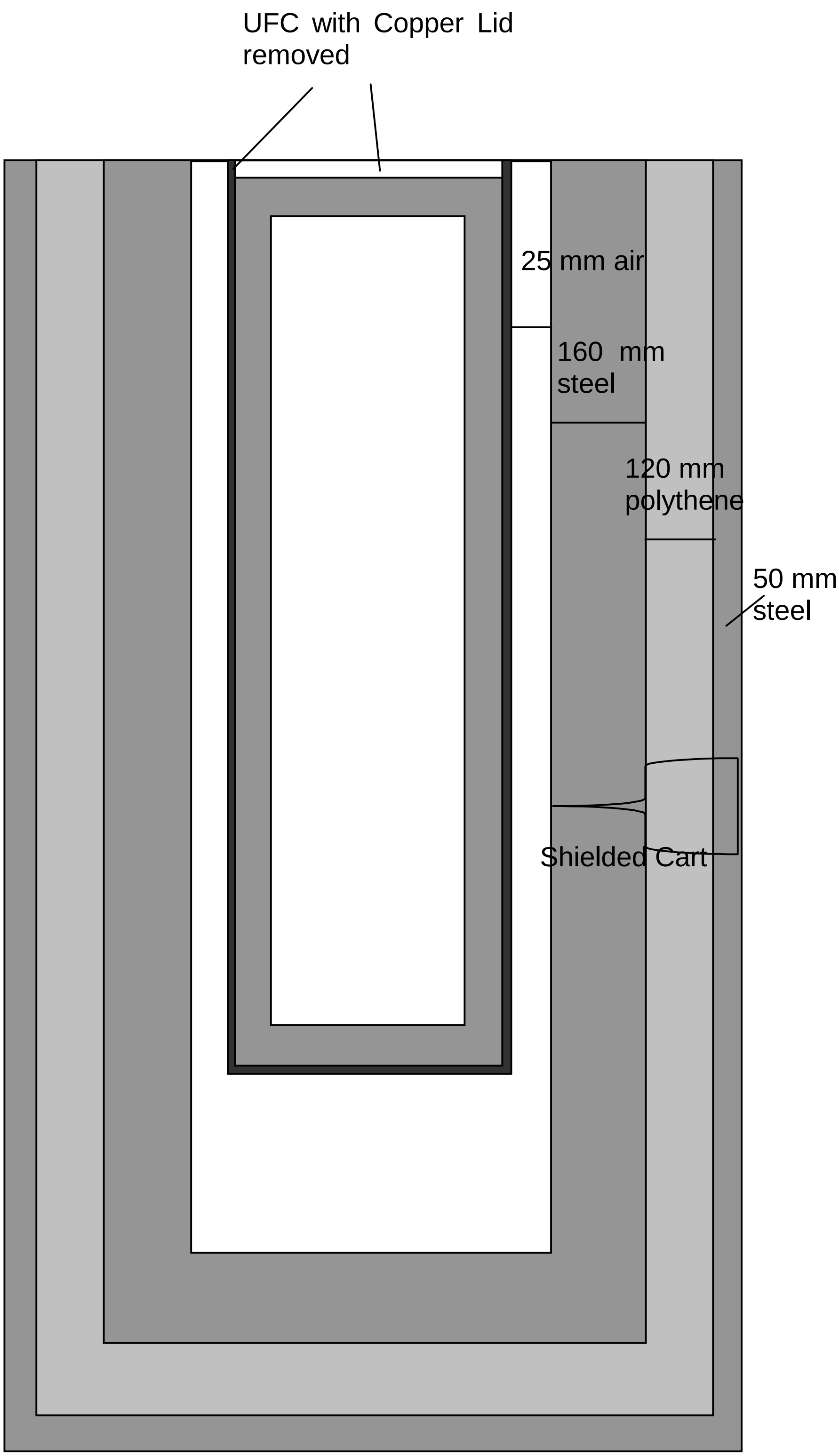


Figure 9 Section through Calculational Model of Shine from UFC in Shielded Cart and Transfer Tunnel Roof (Not to Scale)

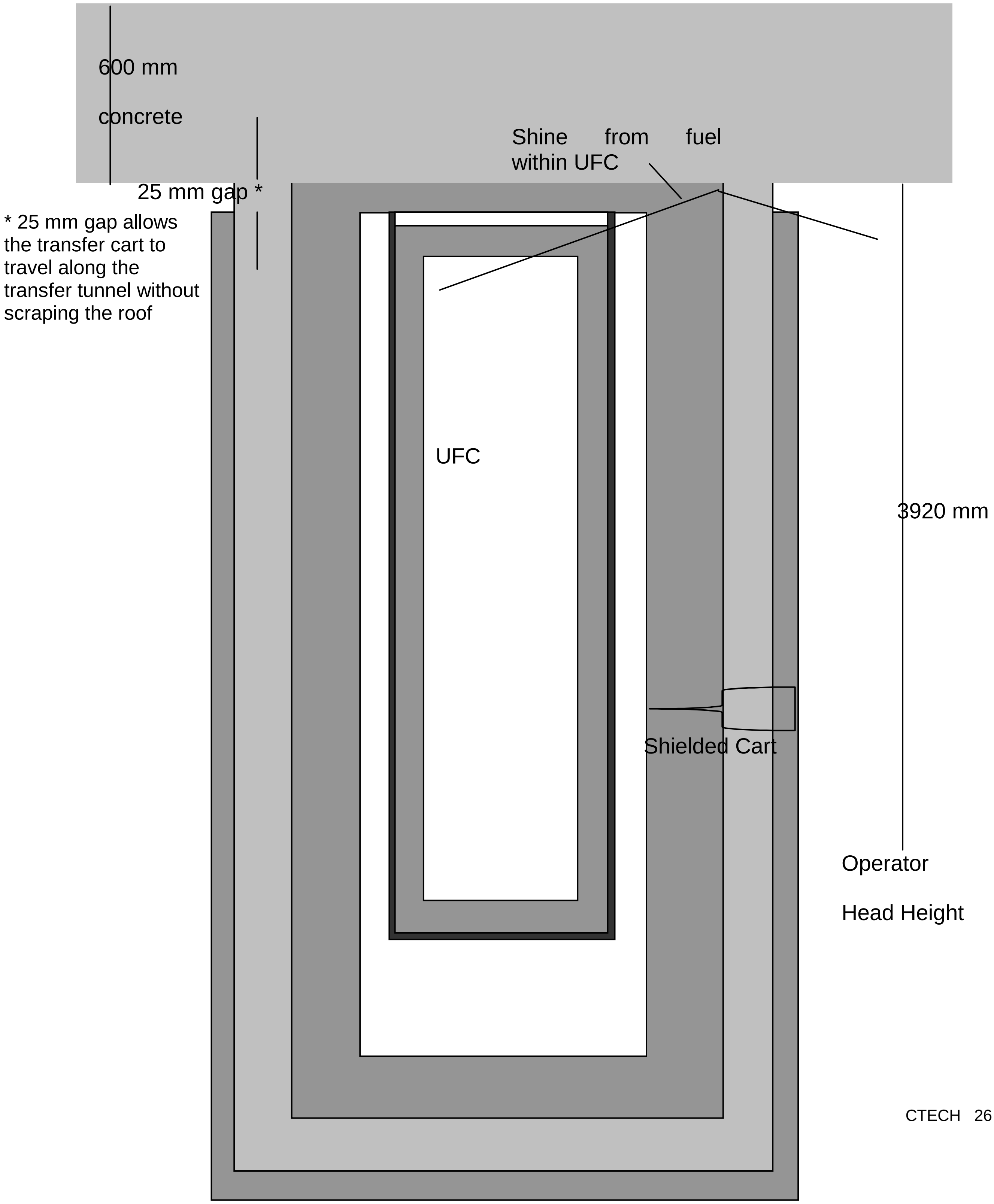


Figure 10 Section through Calculational Model of UFC Raised to Cell (Not to Scale)

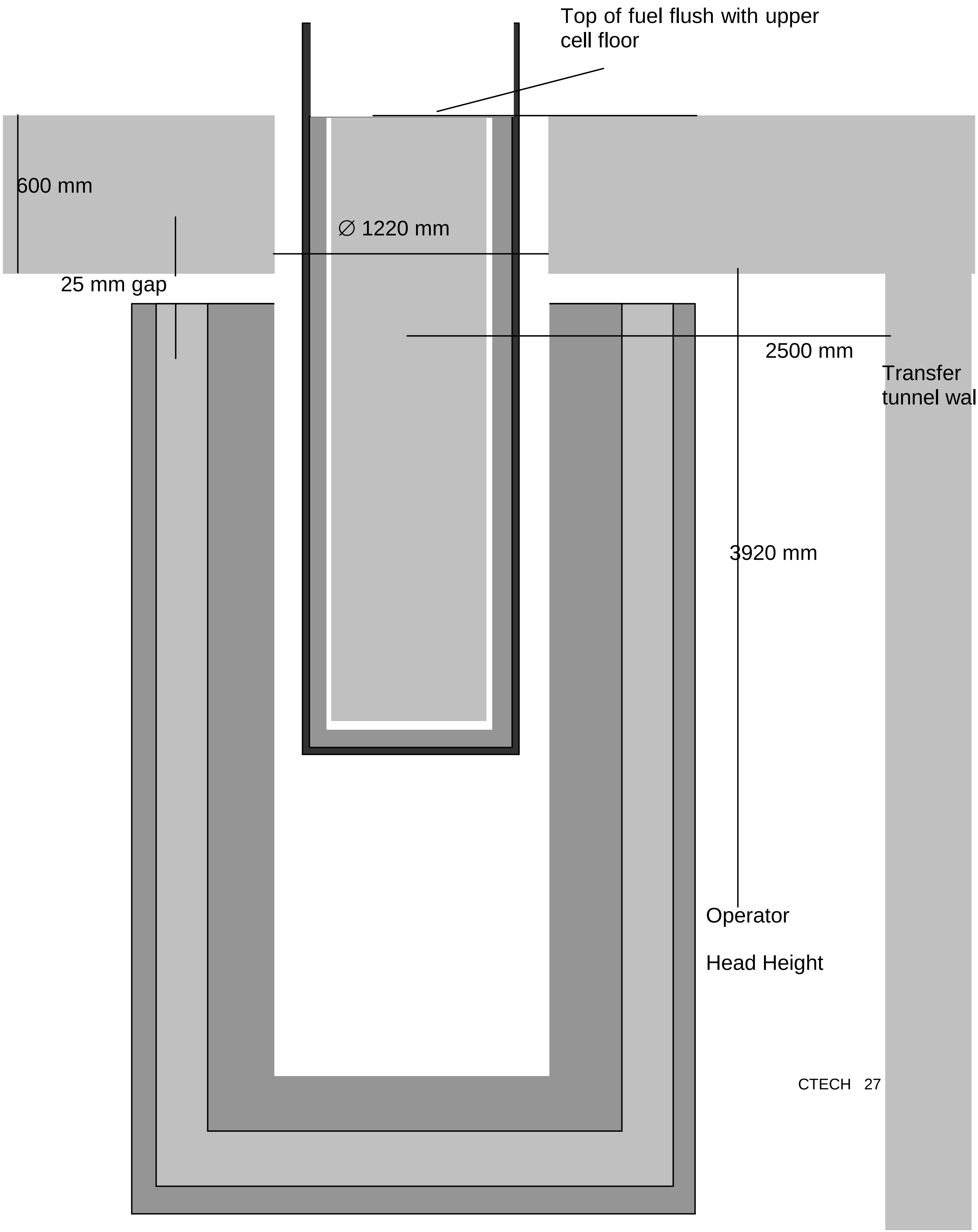


Figure 11 Section through Calculational Model of Baskets Lowered into UFC (Not to Scale)

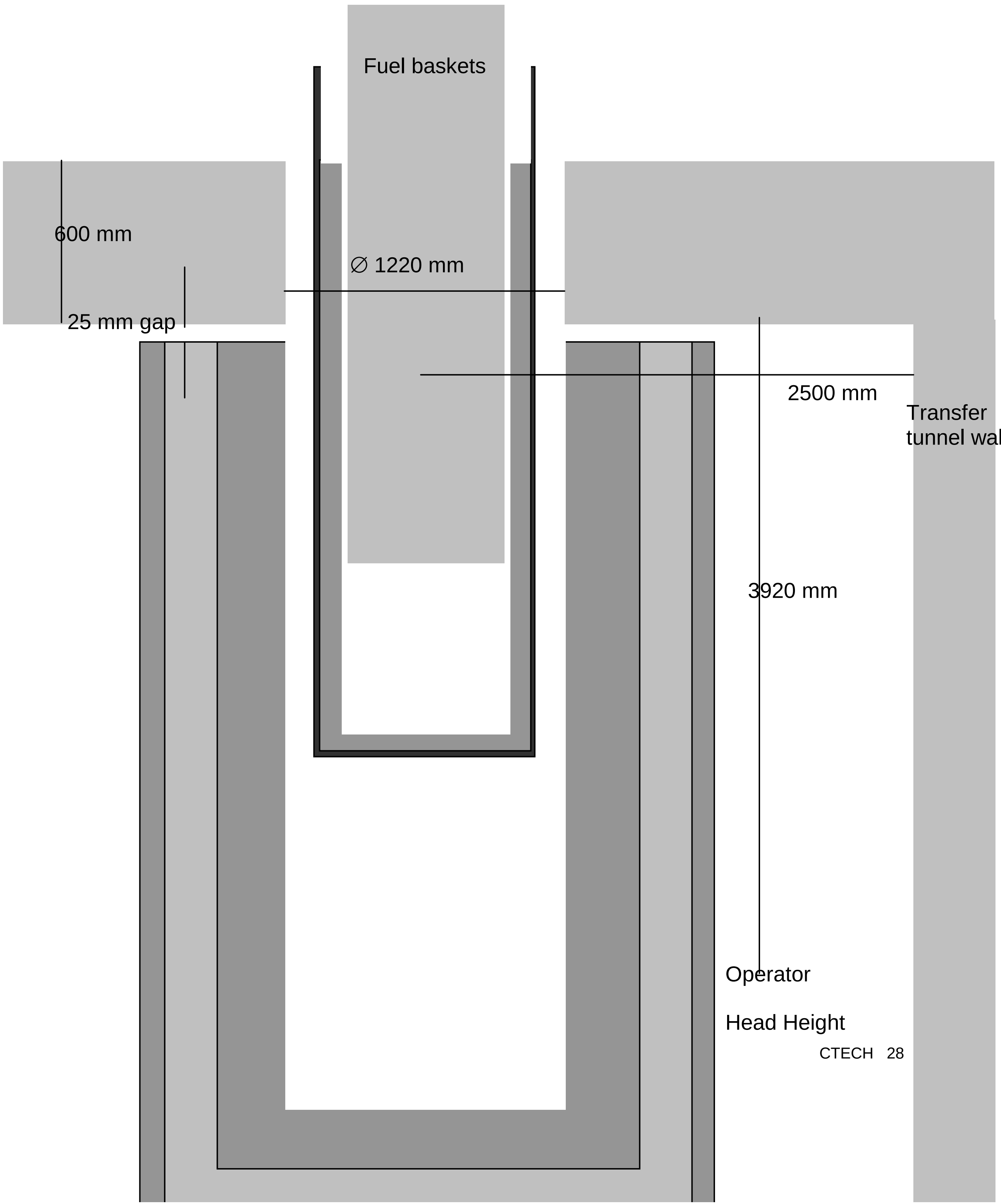


Figure 12 Section through Calculational Model of Sealing Cell (Not to Scale)

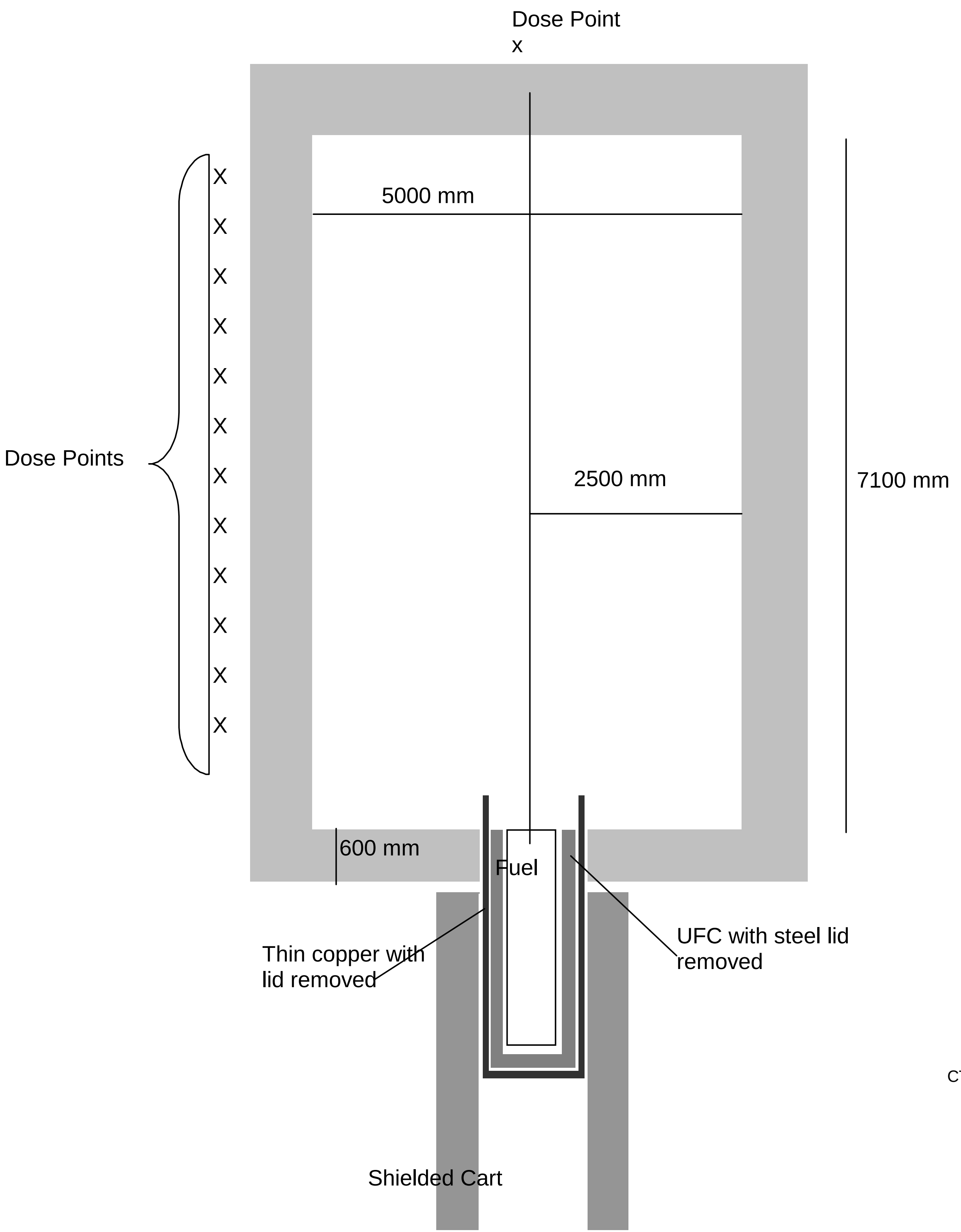


Figure 13 Axial Section through Calculational Model of Emplacement Room (Not to Scale)

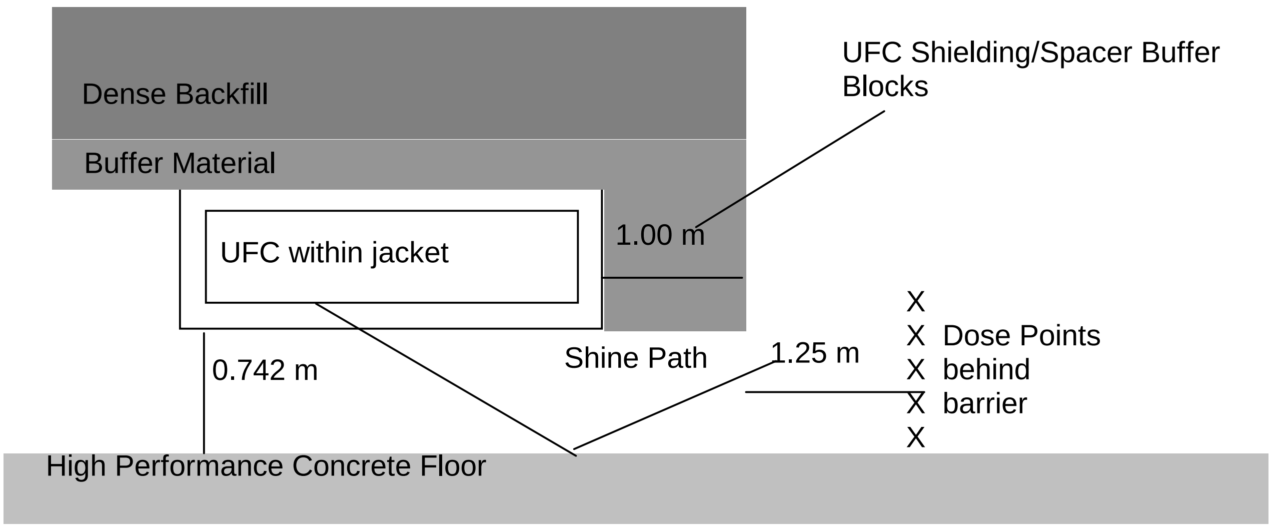


Figure 14 Radial Section through Calculational Model of Emplacement Room (Not to Scale)

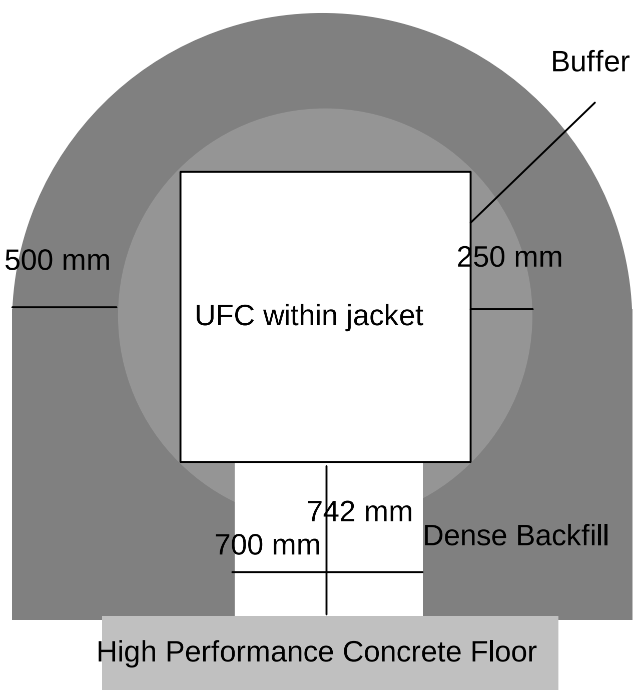


Figure 15 Section through Calculational Model for Shielding/Spacer Plug Cask (Not to Scale)