The Californium Irradiation Setup at the University of Minnesota

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This document describes the radiation levels that are expected at various locations near the Californium irradiation facility.

Geometry

The experimental setup is shown in figure 1 below.

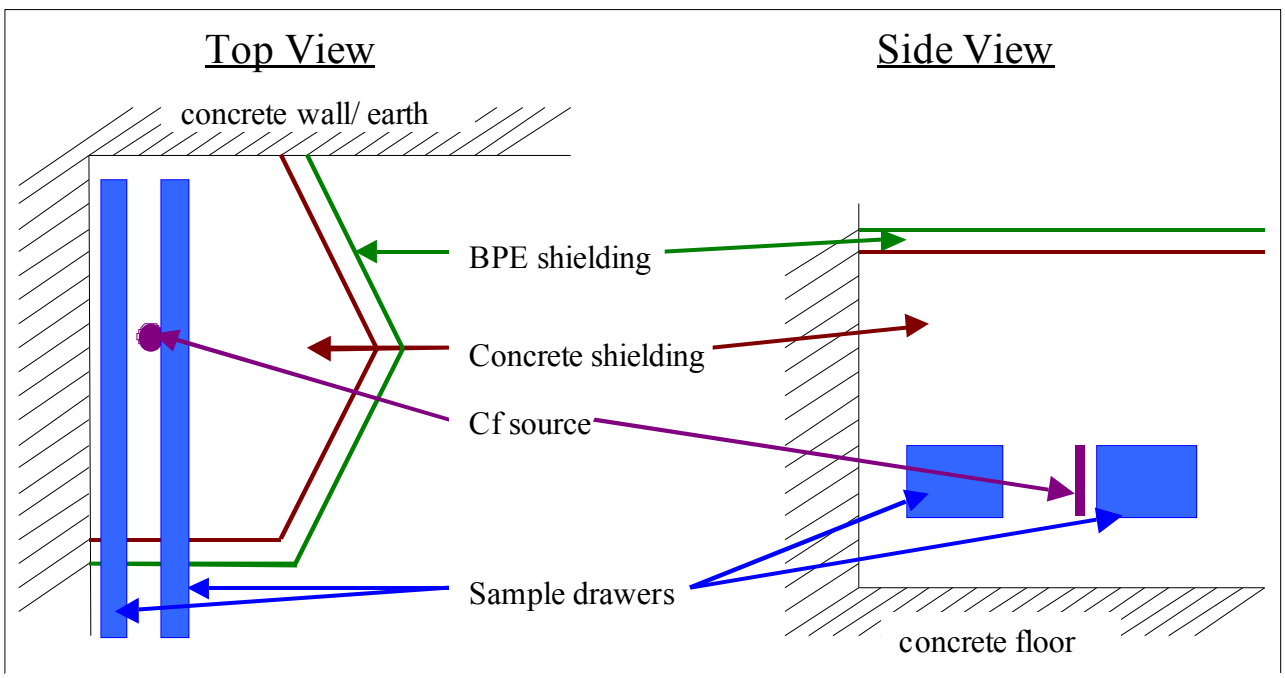

Figure 1: Schematic of the irradiation setup. Concrete shielding blocks, extending 1.5m from the source horizontally and 0.5m vertically surround the source. Two experimental areas are located in drawers that made out of borated polyethylene (BPE). The concrete is covered with a 20cm thick layer of BPE on all sides.

Concrete and borated polyethylene (BPE) shield the user from harmful neutron radiation. At least 1.5m of concrete and 0.2m of BPE are between the source and a user standing on the floor. Only directly above the source, on top of the irradiation setup, is there less shielding and a higher neutron flux.

The calculation of the neutron flux in the two drawers as well as the dose rate was done with simplified geometries to reduce computing time.

Neutron Spectrum

The neutron kinetic energy spectrum produced by the spontaneous fission process of Californium is shown in Figure 2.

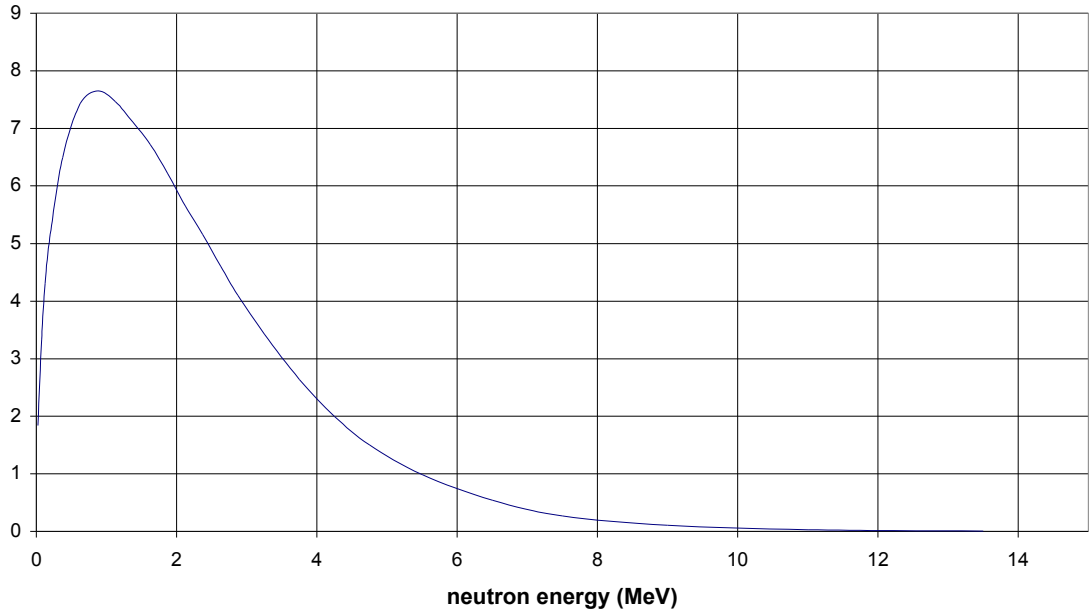

Figure 2: Neutron kinetic energy spectrum produced in the spontaneous fission of Californium nuclei.

Californium produces 2.314×10^{9} neutrons/mg/sec, which corresponds to 3.28×10^{10} neutrons/sec for our 14mg source. At a distance of 1m this neutron flux corresponds to an unshielded dose rate of 300mSv/hour.

Dose Rate

The dose in Sievert (Sv) was calculated from the neutron flux based on the conversion factor given in the PDG book[1]. The conversion between particle flux and dose rate is not exact, and different authors give different factors. However, the dose rate is supposed to measure damage to humans, and that varies from person to person.

Consequently, the dose rates given here should be viewed as estimates with an uncertainty of a factor of two. That is sufficient for shielding calculations since the purpose of shielding is to minimize radiation exposure to humans working on the experiment. The goal is to stay well below all recommended limits.

The conversion is from neutron flux, i.e. the number of neutrons passing through a unit surface in unit time. This implies that the dose rate is independent of the surface area considered in the problem. However, if the surface area is considerably smaller than a human body, then the full-body dose limit is not applicable, and only the organ dose limit should be considered.

The following values were used to find the conversion factor:

[1] C. Caso *et al.* Eur. J. Phys **C3**, 164 (1998). Figure 26.4 in that book contains a reference to Sannikov, A V and Savitskaya, E N, *Ambient dose and ambient dose equivalent conversion factors for high energy neutrons*, CERN TIS RP 93-14 (1993)

Neutron energy (MeV)	Conversion factor (pSv/cm^2)
0.001	6.2
0.01	9
0.1	70
1	320
4	400
10	450
20	600
100	300
250	280
1000	600

Table 1: Fluence-to-dose equivalent conversion factor for various neutron energies.

Figure 3 shows a graph of this dependence. The function increases by two orders of magnitude at a value of about 0.1MeV. Neutrons with a lower energy don't produce much damage and the conversion factor stays below 10 for smaller neutron energies.
The total dose for a particular geometry can therefore be found by recording the neutron flux spectrum and multiplying it (bin-by-bin) with the conversion function.

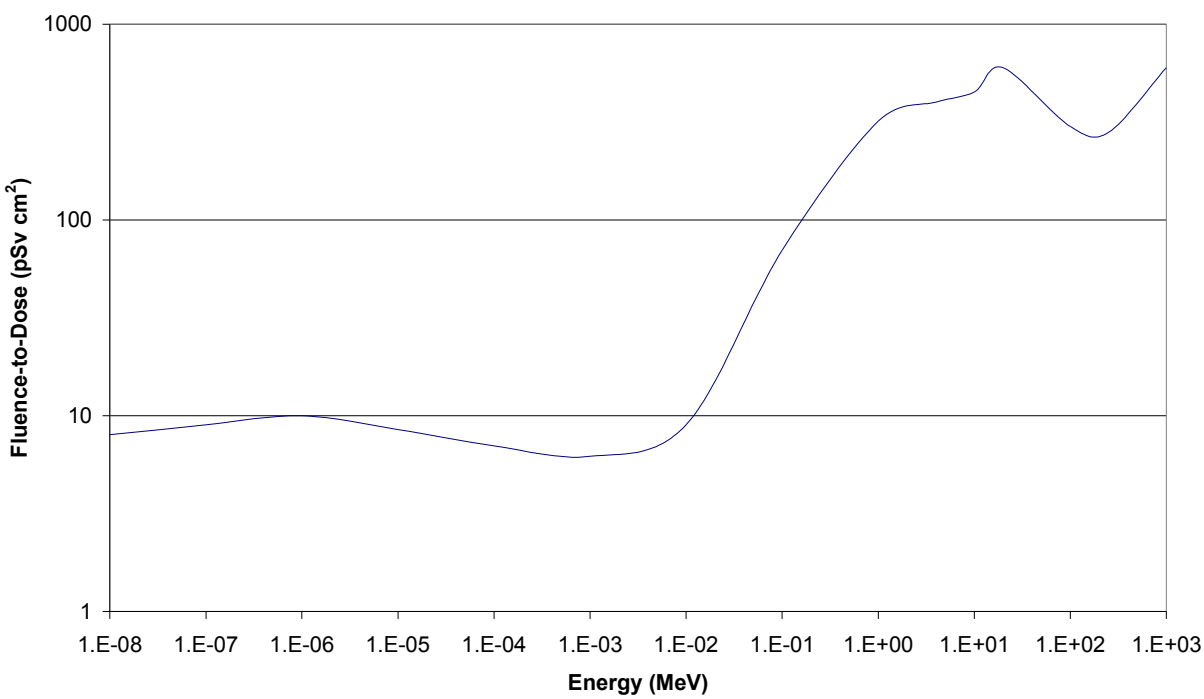

Figure 3: Neutron fluence to dose equivalent conversion factor for various neutron energies.

The maximum allowable full-body dose rate is 50mSv per year according to Nuclear Regulatory Commission (NRC) regulation 10 CFR part 20. Individual organs may receive a dose of up to 500mSv per year. A lethal dose corresponds to about 60000mSv.
The desired dose rate in this setup is 0.02mSv per hour (2mrem/hour). This rate is not to be exceeded. It is low enough to minimize health risks.

Shielded Dose

The dose rate outside of the setup was calculated with the MCNP particle transport code. This software has been widely used to compute neutron flux from nuclear reactors, particle accelerators, and fission sources. The results presented here were obtained with MCNP version 4C.
Since the size of the setup is comparable to a human body, this dose should be compared to the full-body dose limit given above.
The shielded dose rate outside of the irradiation setup was calculated assuming a spherical geometry with no drawers. The geometry consisted of a sphere of concrete with a density of 2.3grams/cm^3. The composition of the concrete is shown in Table 2.

Element	**Atomic fraction (%)**
Oxygen	69.3
Silicon	21.7
Calcium	4.0
Sodium	1.0
Iron	1.3
Aluminum	2.7

Table 2: Element composition of the concrete used in the MCNP calculation.

A 0.2m thick shell of borated polyethylene surrounded the concrete. Its composition is shown in Table 3. A density of 0.99grams/cm^3 was used.

Element	**Weight fraction (%)**
Hydrogen	12.7
Boron	3.0
Carbon	71.0
Oxygen	13.3

Table 3: Element composition of borated polyethylene. In contrast to Table 2, the composition is given in terms of the weight fraction.

The flux on the surface was calculated for several various distances from the source. The result is shown in Figure 4. With just the concrete sphere, the dose at a distance of 1.5m is 0.1mSv per hour. Assuming a workweek of 40 hours, this corresponds to a dose rate of 4mSv per week for a user of the facility, or a maximum of about 12 weeks per year at the facility.
The additional layer of BPE lowers the dose to 0.3μSv, a reduction by a factor of about 1000. At this level, even a constant exposure (24 hours per day, 365 days per year) would only add up to about 3mSv, which is well below the NRC limit and similar to the natural background dose.

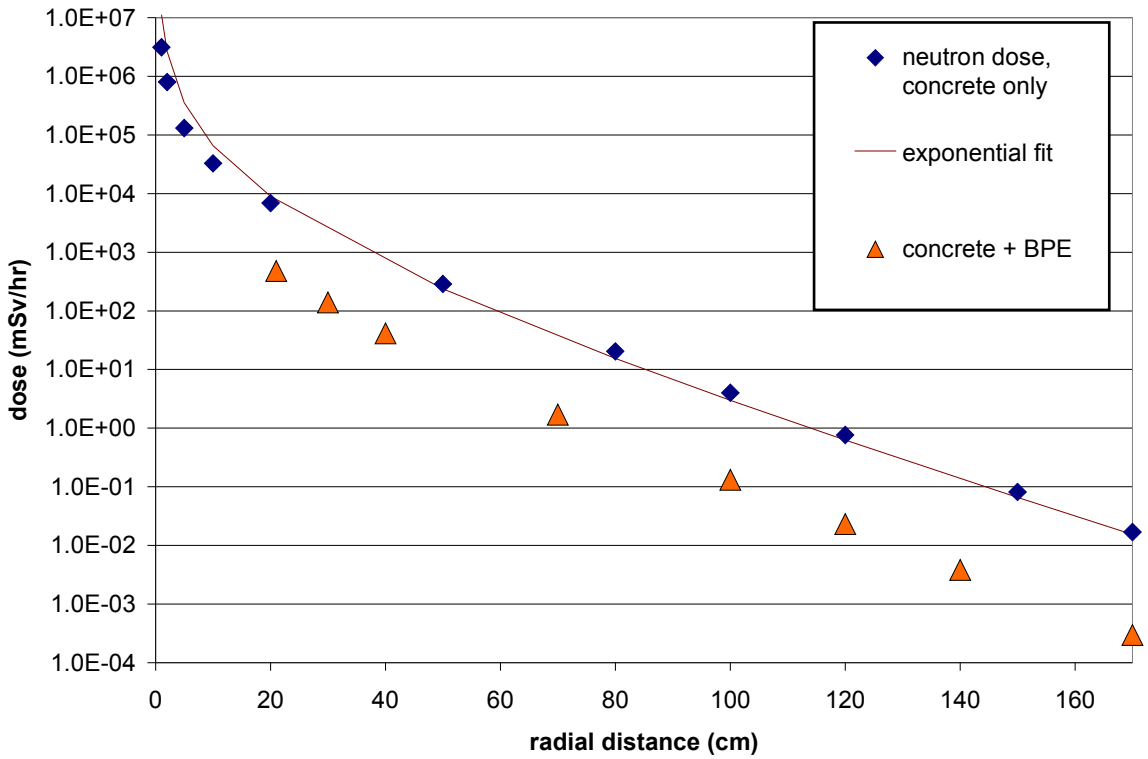

Figure 4: Converted dose rate at the outside of a sphere for various radii, with and without a 0.2m thick shell of BPE.

Dose due to a crack in the concrete

It could in principle be imagined that the concrete shielding breaks or cracks and neutrons from the Cf reach the user unattenuated. This situation may occur after an earthquake or a similar destructive disaster. In this case, flux is significantly higher and the direct neutron flux can be calculated according to simple geometry arguments: Since the unshielded dose rate is 300mSv/hour at a distance of 1m, the dose rate at 1.7m is: 300mSv/hour $\times$ $100^2/170^2$ = 100mSv/hour.

The crack geometry used in the simulation is shown in Figure 5. The sphere from the previous problem now has a thin crack down the center with the source at the center of the crack. The crack is 10cm high and has a variable thickness, between 0.5cm and 2cm. It spans the entire diameter of the sphere. The mirror geometry of the problems gives the advantage of reducing the total computing time.

As mentioned above, this dose rate does not depend on the size of the crack. Moreover, since the crack is small compared to a human body, the organ limit of 500mSv/year applies here. The dependence of the dose rate on the radial distance is shown in Figure 6.

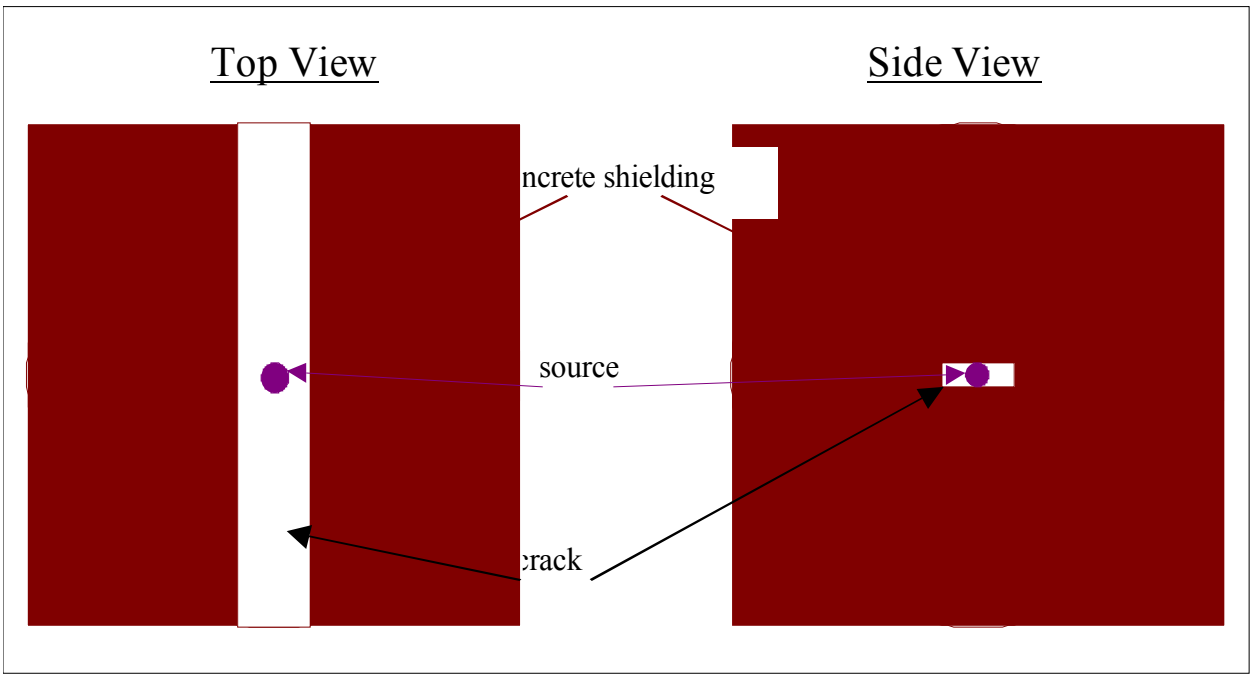

Figure 5: Schematic view of a crack in the shielding.

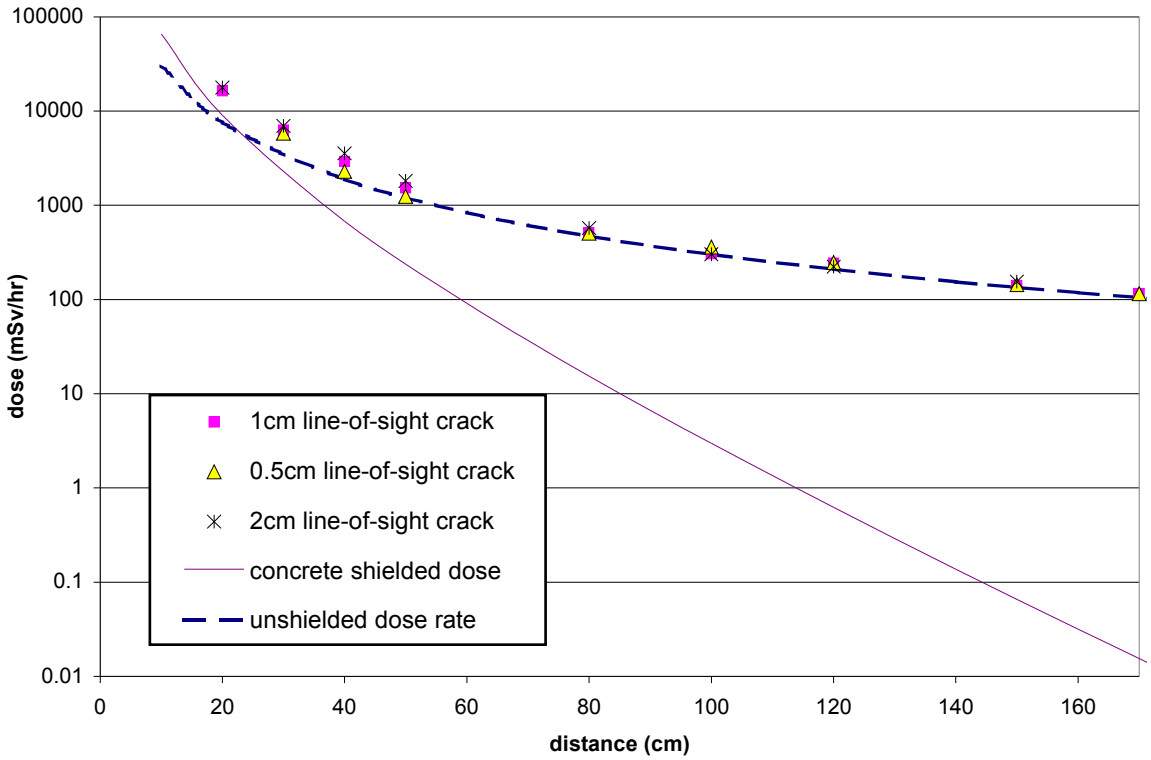

Figure 6: Dose rate at the end of a crack that gives a direct line-of-sight to the source. The solid line shows the dose rate with concrete shielding, it is identical to the line in Figure 4.

The graph shows that at a distance of 1.7m, the direct flux from the source dominates over contributions due to neutron scattering in concrete. The simulation also reproduces the independence on the width of the crack.

Dose due to an off-center crack

If the crack is moved off-center, then neutrons have to pass through concrete first, and no direct neutrons from the source can reach the outside. The geometry for this case is shown in below.

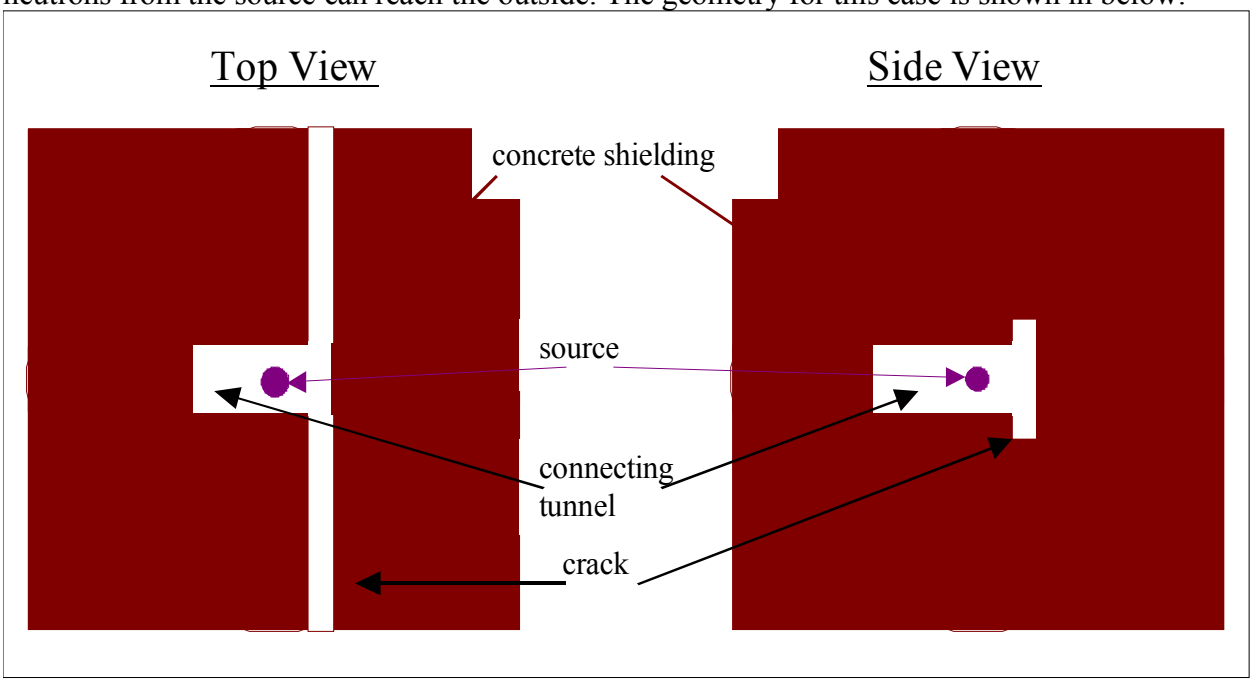

Figure 7: Schematic view of a crack in the shielding that is off-center.

The distance between the source and the closest edge of the crack is always 4cm. The crack is 10cm high and its width varies between 0.5cm and 2cm.
For distances less than about 0.8m, the dose at the crack is comparable to the dose for just the concrete, indicating that the concrete simply attenuated most of the neutrons at the end of the crack. For larger distances, the dose at the end of the crack is considerably larger than the dose for just the concrete. This excess in neutron flux is produced when neutrons scatter at the edge of the crack and travel almost parallel to the crack edge.
Figure 8 shows a comparison of the dose rate at the end of the crack to the shielded dose rate. The effect of increased dose for large distances can be seen in the figure. The dashed line shows a fit to the solid line plus a $1/r^2$ contribution. This fit assumes that the excess in neutron flux is produced by the above mechanism. Neutrons enter the crack at all directions, and then propagate unattenuated inside the crack. This propagation is described by a $1/r^2$ behavior. To be correct, the distance along the crack should be considered rather than the radial distance from the source. However, for distances much larger than the 4cm distance between the source and the crack, this is a good approximation.

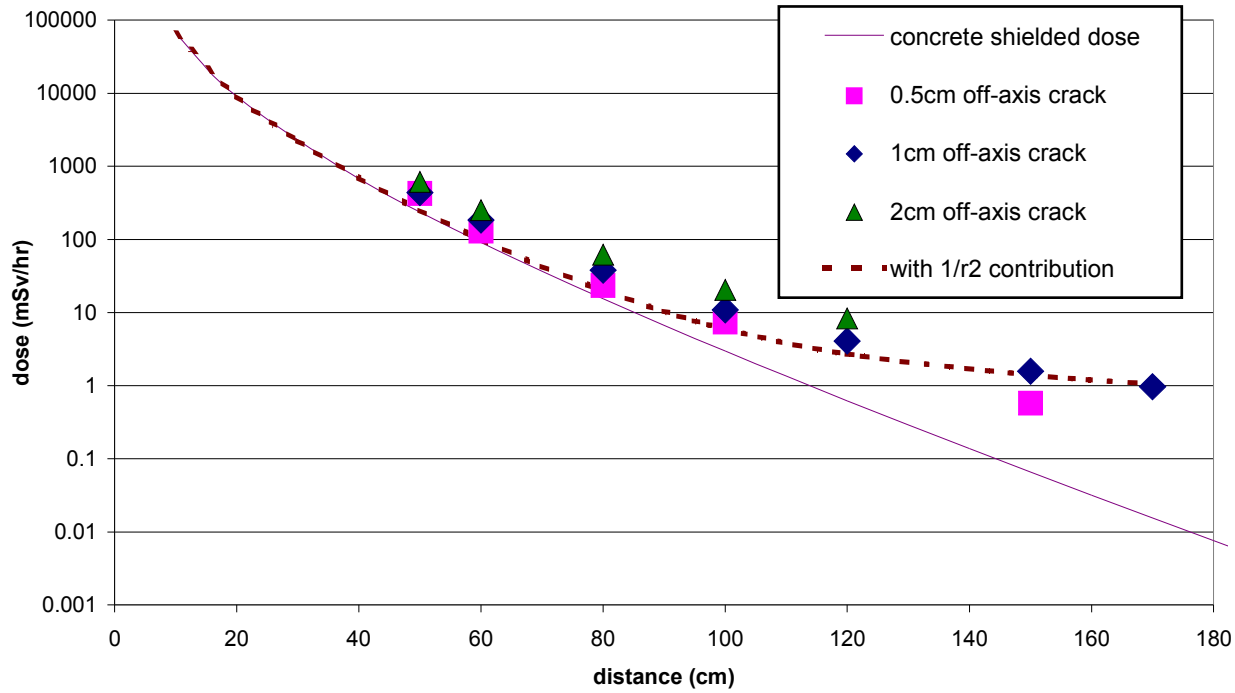

Figure 8: Dose rate produced at the end of a crack that is 10cm high and has a variable width. The distance of closest approach between the source and the crack is 4cm. The solid line shows the dose for concrete shielding only. The dashed line shows a fit to the solid line plus a distance^{-2} contribution.

Adding BPE around the crack reduces the number of neutrons entering the crack and consequently also the neutron flux at the end of the crack. With 2.5cm thick walls, the neutron dose rate decreases by roughly an order of magnitude.
The dose rate at the end of the crack starts to be significantly larger than the dose rate coming from the concrete at a distance of about 1m. Below that value, the number of neutrons propagating unattenuated in the crack is smaller than the number of neutrons entering the crack from the concrete.