Neutron Response Analysis of BeO OSL Personal Dosimeter

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Outline

- BeO OSL personal dosimeter system
- Thermal Neutron response analysis of BeO OSL personal dosimeter (Exp.---- MC Simulation)
- Fast Neutron response analysis of BeO personal dosimeter (Exp.------MC Simulation)
- Conclusion & Discussion

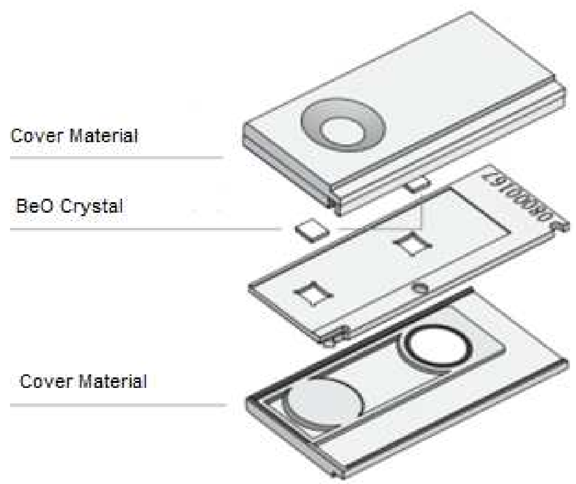
Motivation

- Our some users ask me“ What is the neutron measurement capability of BeO OSL personal dosimeter”. (major motivation).

- BeO OSL personal dosimeter system is very new , there is no study in the literature to reveal neutron measurement capability of BeO OSL personal dosimeter. (minor motivation).

BeO OSL personal dosimeter system

- Dosimeter has two detectors to measure Hp(0.07) and Hp(10)
- BeO detector 4.7 mmx4.7 mmx0.5 mm, 2.85 g/cm^3
- BeO detector is covered with 2.4 mm Teflon, 0.5 mm plastic are used for Hp(10) and Hp(0.07), respectively
- Effective Atomic Number of BeO detector, Zeff=7.13 (nearly tissue equi.)
- BeO detector has low energy dependency (unity)
- BeO detector has dose linearity up to 25 Sv
- BeO detector is less light sensitive compared with other OSL material

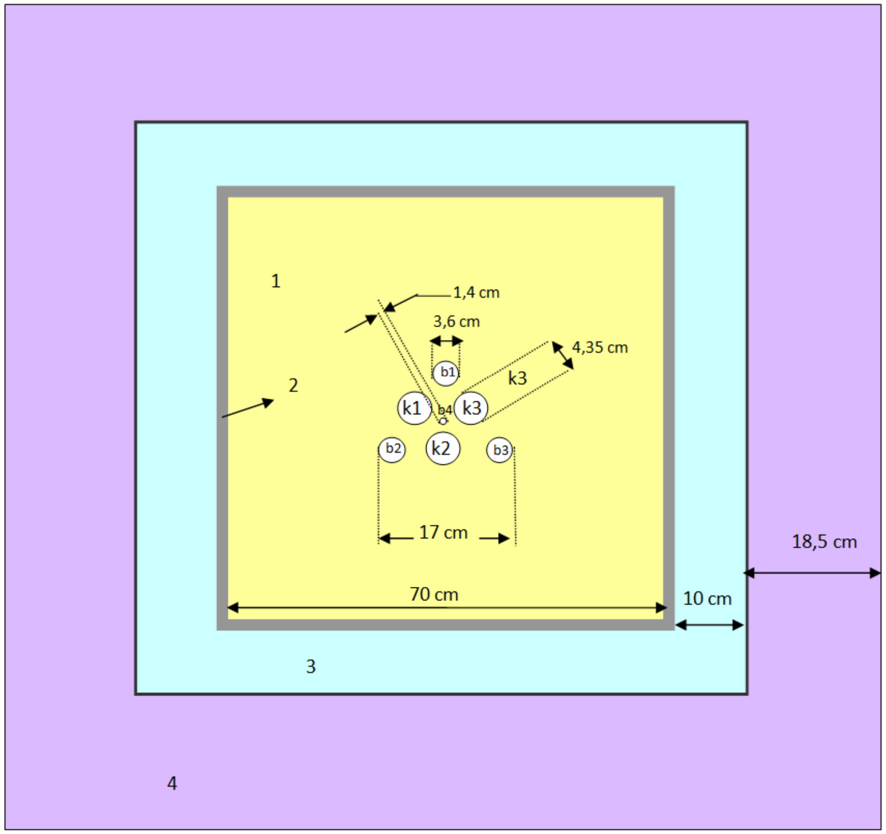


BeO OSL system

BeO detector is stimulated with blue light , using CW method

In the BeO based OSL system, reader calibrated for X-ray and gammas so quality factor is unit, evaluated dose results could be use as a absorbed dose value. (Gy) (Important point for this study)

Thermal neutron irradiation system (TNIS)



2-D drawing of TNIS

Region-wise material densities

Region Number	Material Type	Material Density(g/cm^3)
1	Paraffin	1.03
2	Cadmium	8.65
3	Boric Acid	1.42
4	Lead	11.3

Thermal neutron irradiation system (TAEK-SANAEM)

Experimental Procedure

a.) TNIS has 3 Am-Be cylinder sources with activity 16 Ci at k1,k2 and k3 with a dimension 1.6 cm diameter, 3cm height.

b.) Am-Be source provides 2.2E+6 n/sec-Ci (ISO standard 8925-1)

c.)3 BeO OSL dosimeters were irradiated as a function of time at b1, b2 and b3 irradiation holes.

d.) Dosimeters were read using OSL reader. (Accredited at RADKOR)

e) In this study, Determined all dose values were reported for Hp(10)

Monte Carlo Simulation

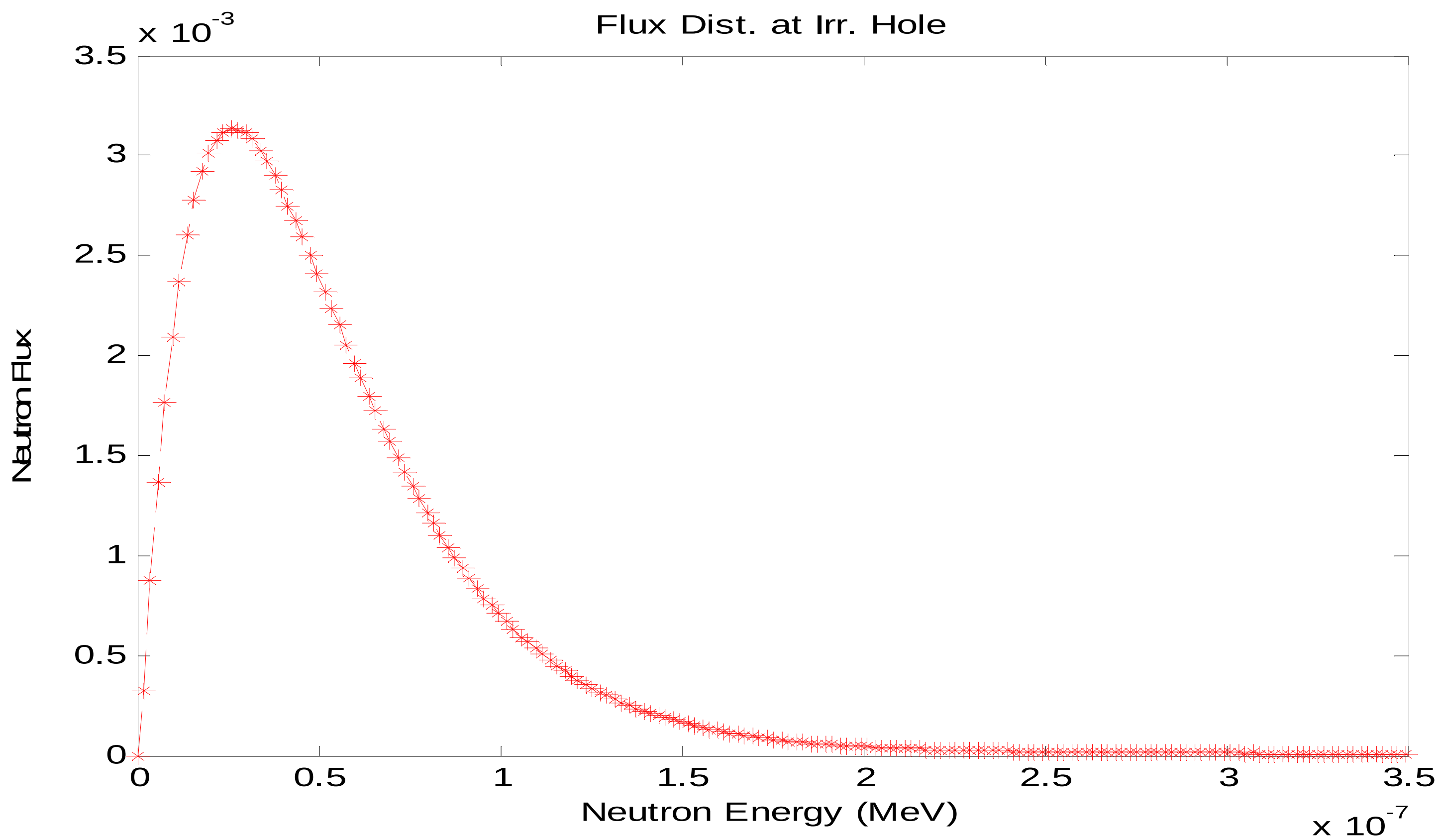
- The TNIS geometry was modeled using MCNP5-Vised with real dimension
- Am-Be source was defined as cylinder volumetric source, energy spectrum was given according to ISO 8529-1
- MC simulations were performed with photon and neutron mode to reveal contribution of gammas
- Number of history was selected in such a way that tallies' relative error remain under 1%.
- F6 tally was used to estimate absorbed doses for neutrons and photons, separately.
- ENDF-VI material lib. was used in MC simulations
- MC run time is roughly 6 hours, using 24 parallel processing cores.

Thermal neutron spectrum

Neutron spectrum were determined at irradiation channel surface using MCNP5 F2 surface flux tally

The estimated neutron spectrum is shown in figure

When the figure is examined, most of neutron energy below 0.05 eV



Determination of absorbed dose rate

- Absorbed dose rates in BeO crystal were estimated using MCNP5 F6 tally (MeV/g-part.)

- F6 tally results were modified in order to absorbed dose rate (Gy/sec) using this definition given in below

- Basically, direct neutron, 59 keV prompt gammas, gammas due neutron capture and 4.438 MeV gammas originated from excited Carbon take into account to determine absorbed dose in BeO crystal.

Coefficients in used to calculate absorbed dose rate

The absorbed dose rate were calculated using eq. ,given in below

Determination of absorbed dose rate

- How to obtain these conversion coefficients?

For example:

For prompt gamma dose rate (59keV)

$$Gamma\ dose\ rate = F6_{result}\ x\ 251.5\ mGy/sec$$

=1.6E-13(Mev--Joul)x 1000 (g--kg)x 3x16(Source Str.)x 0.36 (decay branching ratio) x 0.59E-2 (escape probability from active source region)x1000 (mGy--Gy)

For neutron dose rate

$$Neutron\ dose\ rate = F6_{result}\ x\ 16.09\ mGy/sec$$

=1.6E-13(Mev--Joul)x 1000 (kg--g)x 3x16 ((Source Str.))x2.2E+6 (Ci—neutron/sec-Ci) x1000 (mGy--Gy)

Thermal neutron response of BeO (Results)

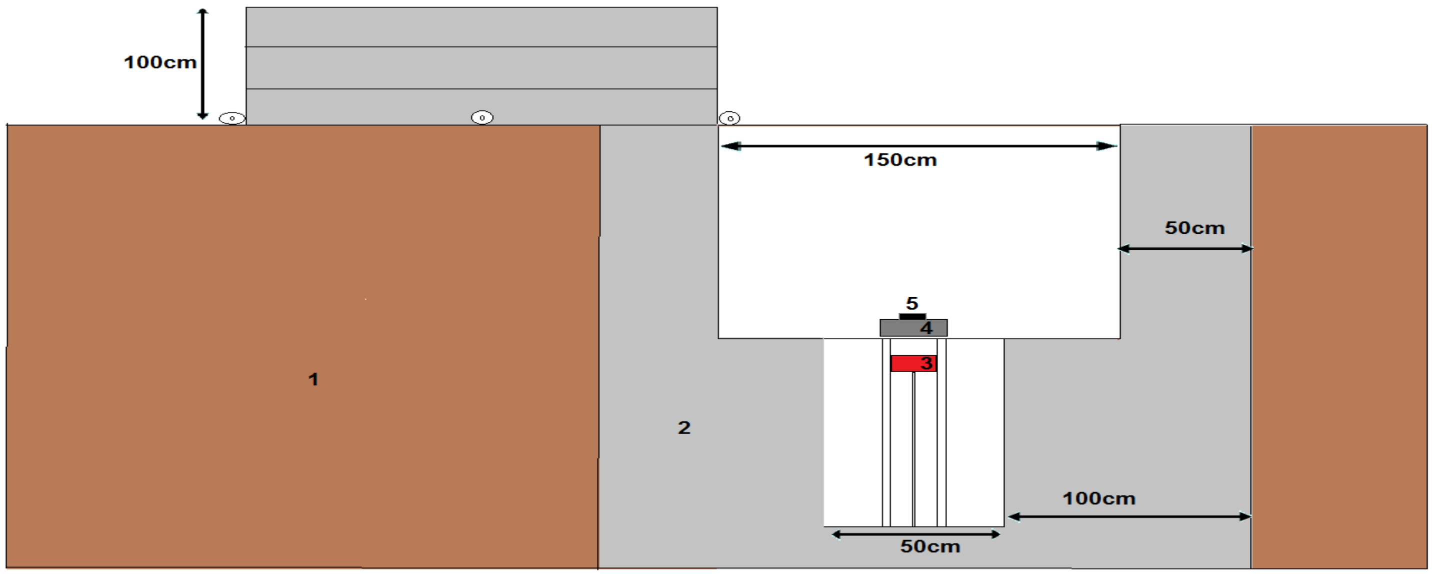
Experimental and MC results are given as function of time

Irradiation Time (sec)	Meas. BeO (mGy)	Gamma MC (mGy)	Neutron MC (mGy)	Total MC (mGy)
36	0.11	0.096	0.008	0.104
180	0.56	0.481	0.04	0.521
360	0.99	0.96	0.08	1.04
1800	4.95	4.81	0.402	5.21
3600	9.54	9.62	0.803	10.4
7200	20.47	19.25	1.607	20.86

Thermal neutron response of BeO (Results)

Experimental results are compared with MC simulation results.

Fast neutron irradiation system FNIS (TAEK-SNAEM)



Region Number	Material Type	Material Density(g/cm^3)
1	Background (Soil)	1.03
2	Concrete	2.35
3	Am-Be Source	---
4	Lead	11.3
5	OSL dosimeter	----

Fast neutron irradiation system (TAEK-SANAEM)

Experimental procedure given as

a.) FNIS has 1 Am-Be cylinder sources with activity 20 Ci with a dimension 1.6 cm diameter, 3cm height.

b.) Am-Be source provides 2.2E+6 n/sec-Ci (ISO standard 8925-1)

c.)3 BeO OSL dosimeters were irradiated as a function of time

d.) Dosimeters were read using OSL reader. (Accredited in RADKOR)

e) Determined dose values were reported for Hp(10) BeO detector

MC simulation

- The FNIS geometry is modeled using MCNP5-Vised with real dimension
- Am-Be source is defined cylinder volumetric source,

  energy spectrum is given according to ISO 8529-1
- MC simulations were performed with photon neutron mode to reveal of gammas
- Number of history was selected in such way that tallies' relative error remain under 1%.
- F6 tally was used to estimate absorbed dose, for neutron and photons separately.
- ENDF-VI material lib. were used in MC simulation
- MC Run time is roughly 360 minutes using 24 parallel processing cores.

Coefficients in used to calculate absorbed dose rate

The coefficients were obtained same definition,

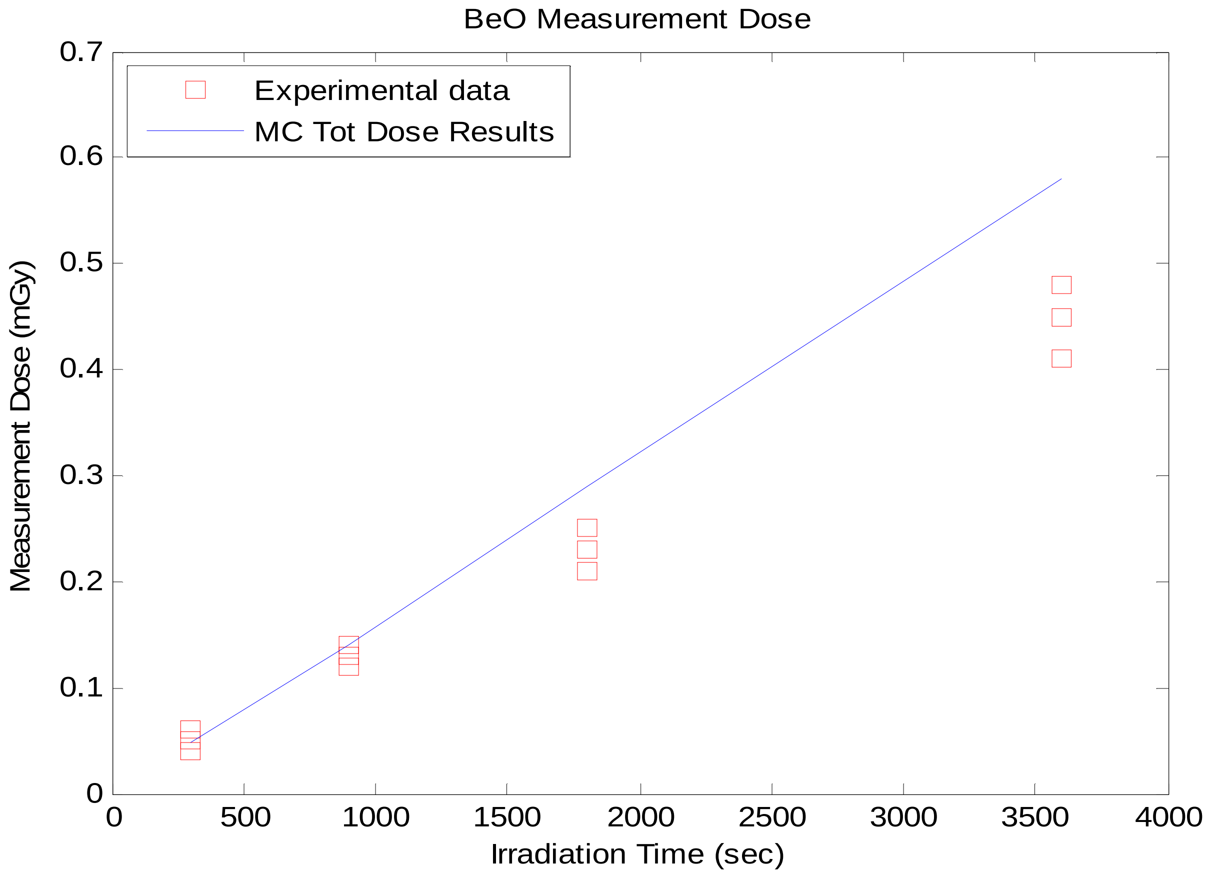
Fast neutron response of BeO (Results)

Experimental and MC results are given as function of time

Irradiation Time (sec)	Meas. (BeO) (mGy)	Neutron MC (mGy)	Gamma MC (mGy)	Total MC (mGy)
300	0.05	0.04	0.008	0.048
900	0.13	0.12	0.024	0.14
1800	0.23	0.24	0.048	0.29
3600	0.45	0.48	0.096	0.58

Fast neutron response analysis of BeO OSL personal

Experimental results are compared with MC simulation results.



Neutron Cross Sections of BeO OSL dosimeter

To reveal neutron response of BeO detector, spectrum averaged microscopic cross section were generated using MCNP5, EndfVI material library was used.
(n,α) reaction dominate in fast region
(n, γ) reaction dominate in thermal region.

Microscopic cross sections (barn)	Thermal Spectrum Averaged	Fast Spectrum Averaged
σ_{tot}	8.84E-5	4.23E-5
σ_{cap}	7.90E-8	6.73E-7
$\sigma_{n,\gamma}$	7.90E-8	≈0
$\sigma_{n,\alpha}$	≈0	6.73E-7

Conclusion

- BeO OSL dosimeter could be measure neutron dose but neutron sensitivity of BeO is too low (very small M. cross section) when compared with x-ray and gammas.
- Fast neutron sensitivity better than thermal neutron, due to reaction type.
- 2-detector BeO OSL dosimeter could be measure neutron dose, could not distinguish from x-ray, gamma dose
- Least four BeO detector (2-gamma, 1-thermal, 1- fast) and appropriate filter material have to be used to measure true neutron dose value. (future work)

Acknowledgement

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- Thank you for your attention