Project Title: High-efficiency Thin-film-coated Semiconductor Neutron Detectors for Active Dosimetry Monitors

Covering Period: 15 July 2004 – 14 July 2009

Date of Report: 31 December 2009

Recipient: Kansas State University
2 Fairchild Hall
Manhattan, KS 66506

Award Number: DE-FG07-04ID14599

Subcontractors: None

Other Partners: None

Contact(s): William L. Dunn, PI
(785) 532-5628
dunn@mne.ksu.edu

Douglas McGregor, Co-PI
(785) 532-5284
mcgregor@mne.ksu.edu

Project Team: Nancy Elizondo

Project Objective: To develop a compact high-detection-efficiency neutron dosimeter capable of discerning between thermal and fast neutron doses in real time.

Background: A need exists for portable neutron dosimeters that have greater than 10% thermal neutron detection efficiency and can provide direct readout in the field. To be of widespread use, these dosimeters should have low power requirements. It would be ideal if these dosimeters could differentiate thermal from fast neutrons, as this would allow characterization (at least to some degree) of the source of those neutrons. Such neutron dosimeters could be used either as personnel dosimeters or in area survey instruments.

Status: The project has been completed and the objectives of the project were met. Prototype perforated semiconductor neutron detectors have been developed and characterized. Individual detectors have been packaged in small, rugged, battery-powered containers that can be worn as active dosimeters and read in real time. Two prototype neutron dosimeters were constructed and tested.

Initial Prototype. Because dosimeters worn on the body are affected by radiation backscattered from the body, an anthropomorphic torso phantom was used to characterize the dosimeter. Thermoluminescent dosimeters (TLD-600 and TLD-700) were used to estimate neutron dose at the surface of the phantom when irradiated by a neutron beam from the tangential beam tube of the TRIGA Mark II reactor at Kansas State University (KSU). Then, a prototype dosimeter was tested, with and without a cadmium backing. The prototype dosimeter provides LED

readout of total counts, which then were related to dose. Use of both bare and cadmium-backed detectors within the dosimeter package can provide some information about the neutron spectrum. Future dosimeters will provide LED readout directly in dose units.

The detector incorporated in the initial prototype dosimeter was a cylindrical silicon diode having an active diameter of 6 mm, which was etched with 9,681 cylindrical holes. Each hole was 30 μm in diameter and 76.6 μm deep. The holes were filled with ^6LiF and the top of the detector was coated with a layer of ^6LiF of thickness 39.5 μm . This design significantly increased the probability that an ejected reaction product will reach the semiconductor substrate and thus be detected.

The prototype dosimeter tested contained only a single bare detector, three AAA cell batteries, micro-electronics, and an LED readout in a package about 6×5×3 cm (see Fig. 1). Future devices will contain both bare and cadmium-backed detectors.

Figure 1 (right). Photograph of the initial neutron dosimeter package. The LED displays the total number of counts recorded since the device was last reset; the display is reset by pushing the reset button visible on the front face.

A Biodex Medical anthropomorphic torso phantom was used for characterization of the neutron dosimeter. The phantom consists of a body-shaped cylinder with lung, liver, and spine inserts. Lung inserts are filled with Styrofoam beads and the phantom is filled with water in order to simulate appropriate local densities. Nevertheless, the composition of the phantom is somewhat different from that of the human body in that there is little or no nitrogen in the phantom and the phantom has no ribs. Nevertheless, our phantom, which was called “Harry,” simulates much of the anatomical structure of the upper torsos of an average male or female.

Thermoluminescent dosimeters (TLDs) were used to estimate the neutron dose on the surface of Harry due to a mixed-field beam from the TRIGA reactor. The gamma-ray portion of this dose must be separated from the total dose in order to estimate the neutron portion. The TLD-600 dosimeters contained 95.6% ^6Li and the TLD-700 dosimeters contained 0.007% ^6Li and are thus unresponsive to neutrons. Therefore in a mixed n - γ field, the TLD-600 response is given by

$$R_{600}^{\gamma+n} = R_{600}^{\gamma} + R_{600}^n, \quad (1)$$

while the TLD-700 response is simply

$$R_{700}^{\gamma+n} = R_{700}^{\gamma}. \quad (2)$$

With k defined as

$$k = \frac{R_{600}^{\gamma}}{R_{700}^{\gamma}}, \quad (3)$$

the neutron response in a mixed field can be obtained from

$$R_{600}^n = R_{600}^{n+\gamma} - kR_{700}^{\gamma}, \quad (4)$$

if the value of k is known.

To estimate the value of k , both TLD-600 and TLD-700 dosimeters were irradiated using a Panoramic Irradiator, which contained a ^{137}Cs source with known source strength. The initial charge (before irradiation) of each TLD was recorded. TLDs were placed at five positions (20, 30, 40, 50 and 60 cm from the source) and were irradiated for 40 minutes. The initial charge of each TLD was subtracted from the final charge. The dose rate at each position could be estimated by

$$D = \frac{S_0}{4\pi r^2} \Re, \quad (5)$$

where S_0 is the source strength (s^{-1}), r is the source-to-dosimeter distance in cm, and $\Re$ is the response function in Sv cm^2 . The average value of k determined from the readings at the five different distances was $k = 0.849 \pm 0.030$.

An experiment was performed using the tangential beam port of the TRIGA Mark II reactor at Kansas State University (KSU). A shutter was used to open and close the beam and a beam catcher was placed behind the phantom. A TLD pair (TLD-600 and TLD-700) was placed in the middle of the front of the phantom, Harry, which was positioned 1 m from the shutter. The TLD pair was irradiated at a reactor power of 100 kW for 1.5 hours. The same experimental procedure was repeated with a cadmium sheet between the TLD pair and Harry. The TLD responses were read out using a Harshaw 2000-C model reader at a heating ramp rate of 10 C s^{-1} from 135 C to 265 C in a nitrogen atmosphere.

The initial charge for each TLD was recorded and subtracted from the final charge after irradiation. The bare and Cd-filtered responses due to neutrons were calculated, using Eq. (4) and the experimentally determined value of k . The bare TLD-600 neutron dose rate per kW of reactor power was determined to be

$$D_n^b = 15.428 \pm 0.309 \text{ mSv h}^{-1} \text{ kW}^{-1}, \quad (6a)$$

and the cadmium-filtered dose rate per kW was determined to be

$$D_n^f = 13.909 \pm 0.286 \text{ mSv h}^{-1} \text{ kW}^{-1}. \quad (6b)$$

The uncertainties were estimated assuming 2% uncertainty in the TLD responses and the specified uncertainty in the value of k .

The neutron dosimeter counts were observed after placing the dosimeter on Harry with the display facing the beam (see Fig 2). The dosimeter was irradiated at a reactor power of 1 kW for times of 5, 10, and 15 minutes. The same experimental procedure was repeated after filtering the dosimeter with a cadmium sheet between the dosimeter and Harry. Dosimeter counts N were recorded for each configuration. The count-to-dose conversion factors $C = D_n / N$ and their standard deviations were determined and are given in Table 1. The inverses of these quantities

indicate that the dosimeter records of order 10 counts per μSv . These experimental results compare favorably with simulation results (using the MCNP5 code), which indicate (see Jahan et al., 2007) that a boron-loaded Si detector whose efficiency is about 15% also will record of order 10 counts per μSv .

Figure 2 (above). The dosimeter mounted on the torso phantom.

Count-to-dose conversion factors determined as above can be used, in principle, to convert dosimeter readout in counts into dose units. However, the actual dose will depend on the energy spectrum of the neutrons. Thus, we define a spectrum factor as

$$S = \frac{N_b - N_c}{N_b}, \quad (7)$$

where N_b is the total count of the bare detector and N_c is the total count of the cadmium-filtered detector. Simulations show, for instance, that S is larger for a ^{235}U reactor Watt spectrum (~ 0.43) than for a room-temperature Maxwellian spectrum (~ 0.23).

A paper was given at the International Forum on Future Directions in Atomic and Condensed Matter Research and Applications, which was held in Melbourne, Australia from 22-23 September 2008, and resulted in publication number 8 listed in the Publications section.

Table 1. Counts from the dosimeter mounted on Harry for three different irradiation times with the reactor operating at 1 kW, equivalent doses, and the dose-count conversion factors.

Irrad. Time (min)	N (counts)	D_n (μSv)	$\sigma(D_n)$ (μSv)	$C = D_n / N$ ($\mu\text{Sv}/\text{count}$)	$\sigma(C)$ ($\mu\text{Sv}/\text{count}$)
Bare					
5	12,533	1,285.68	25.78	0.1026	0.0023
10	25,114	2,571.37	51.55	0.1024	0.0022
15	37,484	3,857.05	77.33	0.1029	0.0021
Cadmium-filtered					
5	11,968	1,159.09	23.79	0.0968	0.0022
10	23,886	2,318.19	47.59	0.0971	0.0021
15	36,153	3,477.28	71.38	0.0962	0.0020

Second Prototype

Publications:

The following eight publications resulted from the research.

- [1] D.S. McGregor, S.L. Bellinger, D. Bruno, W.J. McNeil, E. Patterson, J.K. Shultis, C.J. Solomon, T. Unruh, "Perforated Semiconductor Neutron Detectors for Battery Operated Portable Modules," *Proc. SPIE*, **6706** (2007) pp. 0N1-0N12.
- [2] C.J. Solomon, J.K. Shultis, W.J. McNeil, T.C. Unruh, B.B. Rice, and D.S. McGregor, "A Hybrid Method for Coupled Neutron-Ion Transport Calculations for ^{10}B and ^6LiF Coated and Perforated Detector Efficiencies," *Nucl. Instrum. and Meth. A*, **580** (2007) pp. 326-330.
- [3] J.K. Shultis and D.S. McGregor (2006), "Efficiencies of Coated and Perforated Semiconductor Neutron Detectors," *IEEE Trans. Nuclear Science*, **NS-53**, pp. 1659-1665.
- [4] Q.M. Jahan and W.L. Dunn (2006), "Modeling of highly efficient portable neutron dosimeters," American Nuclear Society Student Conference, 30 March – 1 April, 2006, Rensselaer Polytechnic Institute, Troy, NY.
- [5] W.L. Dunn, Q.M. Jahan, D.S. McGregor, W.J. McNeil, E.L. Patterson, B.B. Rice, J.K. Shultis, and C.J. Solomon (2007), "Design and performance of a portable neutron dosimeter," *Proc. 2nd Workshop on European Collaboration for Higher Education and Research in Nuclear*

Engineering and Radiological Protection, 13-15 March, 2006, Valencia, Spain, pp. 85-92.

[6] Q. Jahan, E. Patterson, B. Rice, W.L. Dunn, and D.S. McGregor (2007), "Neutron dosimeters employing high-efficiency perforated semiconductor detectors," *Nucl. Instr. and Meth. in Phys. Res. B.* **263**, 183-185.

[7] McGregor, D.S., S. Bellinger, D. Bruno, W.L. Dunn, W.J. McNeil, E. Patterson, B.B. Rice, J.K. Shultis, and T. Unruh (2009), "Perforated diode neutron detector modules fabricated from high-purity silicon," *Rad. Phys. Chem.* **78**, 874-881.

[8] Henderson, C.M., Q.M. Jahan, W.L. Dunn, J.K. Shultis, and D.S. McGregor (2010), Characterization of prototype perforated semiconductor neutron detectors, *Rad. Phys. Chem.* **79**, 144-150.

Patents: Two patent applications have been submitted. One concerns the sinusoidal trench design, which minimizes angular dependence of the detector efficiency. The other concerns a method for filling the trenches.

Milestone Status Table:

ID Number	Task/Milestone Description	Planned Completion	Actual Completion	Comments
1	Order semiconductor wafers	7/04	7/04	
2	Design first-generation masks	8/04	8/04	
3	Begin to develop film application process	9/04	9/04	
4	Construct first perforated devices	2/05	2/05	
5	Conduct detector simulations	4/05	4/06	
6	Develop initial coating techniques	6/05	6/05	
7	Develop prototype detectors	7/05	6/05	
8	Order anthropomorphic phantom	9/05	11/05	
9	Begin to develop read-out circuitry	9/05	5/05	
10	Design masks for improved designs	10/05	7/05	
11	Select preferred deposition method	11/05	12/06	
12	Characterize efficiency	12/05	11/06	
13	Complete initial package construction	4/06	4/06	
14	Construct improved detectors	8/06	12/08	
15	Test bare and Cd-covered detectors	12/06	3/07	
16	Test detectors in monoenergetic beam	3/07	1/08	
17	Integrate bare and Cd-covered detectors	4/07	5/09	
18	Calibrate integrated package	6/07	7/09	

Budget Data:

			Approved Spending Plan			Actual Spent to Date		
	From	To	DOE Amount	Cost Share	Total	DOE Amount	Cost Share	Total
Year 1	7/15/04	7/14/05	\$96,417	0	\$96,417	\$43,945 ^{1,2}	0	\$43,945
Year 2	7/15/05	7/14/06	\$107,774	0	\$107,774	\$87,786 ²	0	\$87,786
Year 3	7/15/06	7/14/07	\$102,502	0	\$102,502	\$91,362 ²	0	\$91,362
Year 4	7/15/07	7/14/08	\$97,065	0	\$97,065	\$26,144	0	\$26,144
Year 5	7/15/08	7/14/09	\$57,456		\$57,456	\$57,456	0	\$57,456
Totals				0		\$306,693	0	\$306,693

¹ Expenditures in Year 1 were low because senior personnel charged less time than budgeted and some purchases and expenses were delayed until diode development processes were refined.

² Amounts changed slightly from earlier reports due to change in accounting for administrative costs.