

CANDU[®]-Like Fields for Neutron Dosimetry

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Abstract

It is difficult to undertake neutron dosimetry at CANDU[®] power plants. This is because no dosimeter is uniformly sensitive over the entire range of neutron energies observed in these environments, which span from thermal up to several MeV. An ability to compensate dosimeter responses for the neutron spectra observed in the field would greatly improve dose measurements. Typically, however, bare ²⁵²Cf or (α ,n) neutron sources that are skewed towards higher energies are used for dosimeter calibration. In this work, therefore, MCNPX radiation transport simulations are performed to investigate how CANDU[®]-Like fields may be produced artificially at Canadian Nuclear Laboratories.

Keywords: CANDU[®], CANDU[®]-Like fields, neutron dosimetry, MCNPX

1. Introduction

The response of any radiation dosimeter is best optimized if it is calibrated in the radiation field that it is designed to assess. This is particularly true for neutron fields that NEWs (Nuclear Energy Workers) would typically encounter throughout CANDU[®] (CANadian Deuterium Uranium) power stations. These fields span thermal energies up to several MeV, and have significantly variable fluence distributions. Ideally, spectra found in such environments would be first characterized and then subsequently reproduced in a controlled, laboratory setting. Only then may they be routinely applied to the response calibration of neutron dosimeters. By characterizing these workplace fields, one would know beforehand the expected dose rate. This would then allow identification of any discrepancies against measured values. Using this information, correction factors can then be developed to appropriately modify the dose response functions [1].

Neutron emitting radionuclides are typically used to calibrate neutron dosimeters. This is largely due to their availability and convenience. Examples include spontaneous fission neutron sources, such as ²⁵²Cf, or a variety of other (α ,n) sources. Compared with CANDU[®] fields, though, spectra emanating from such sources are relatively hard and not representative of the complex, bimodal distributions that are commonplace [1]. These bimodal distributions usually have a thermal neutron peak at approximately 10^{-8} MeV, and a fast neutron peak near 1 MeV. The thermal neutron peak is populated by fission neutrons, produced in the reactor core, that have been slowed down (thermalized) by moderator materials and surrounding

shielding. The fast neutron peak is populated by those fission neutrons that have passed through the shielding unaffected. Found between the thermal and fast neutron peaks are neutrons that have been moderated to varying degrees.

The idea of creating artificial, CANDU[®]-Like fields has been previously studied at Canadian Nuclear Laboratories (CNL). At the time in which it was studied, the CNL Health Physics Neutron Generator (HPNG) facility contained a ²⁵²Cf source and a deuterium-deuterium (DD) neutron generator manufactured by Texas Nuclear. Various moderator assemblies were proposed to shape the primary spectra of these two sources into the desired distributions [1, 2, 3]. It was concluded that the DD neutron generator coupled with a cylindrical moderator assembly, the details of which are given in Section 2, would provide a practical path forward [1, 2, 3]. That study led to the subsequent fabrication of the moderator assembly and later on, to the successful characterization of CANDU[®]-Like fields [3]. Unfortunately, the DD neutron generator has been recently decommissioned resulting in the temporary loss of this unique, facility capability.

The objective of this paper is to reclaim that lost capability. Specifically, this paper will determine if a CANDU[®]-Like field may be generated by moderating the ²⁵²Cf source using the cylindrical moderator assembly previously developed. This latter source-moderator combination has not been previously examined, yet, it offers a unique opportunity to create a CANDU[®]-Like field with hardware currently available at CNL. Detailed here is a MCNPX (Monte Carlo N-Particle eXtended) [4] simulation of the expected neutron field. Results will be benchmarked against the moderated DD spectrum previously characterized and against measured CANDU[®] spectra.

2. MCNPX simulation

A schematic of the cylindrical moderator assembly (CMA) used in the Monte Carlo simulation, along with physical dimensions, is given in Figure 1 [1]. The CMA is constructed of natural uranium with a density of 18.9 g cm⁻³, natural iron with a density of 7.86 g cm⁻³, and polyethylene with a density of 0.93 g cm⁻³. The whole assembly is encased in a thin cadmium sheet with a density of 8.65 g cm⁻³. This CMA represents one of the two configurations utilized in the previous studies. The second configuration, which is not studied in this work, includes an additional cadmium sheet covering the end of the assembly. The purpose of this end cover is to remove thermal neutrons generated within the assembly itself [1].

In this simulation, the ²⁵²Cf source is positioned at the center of the HPNG facility, at a height of 1 m above the floor, and 0.01 m from the natural uranium core of the CMA. The room itself is 10.86 m in length, 8.40 m at its widest point, and 4.27 m from the floor to the ceiling. The walls are modeled as concrete with a density of 2.35 g cm⁻³. These dimensions are based on a previously published schematic [1]. A diagram of the HPNG facility, showing the placement of the CMA, is given in Figure 2. The room is filled with U.S. standard air at density of 1.225 kg m⁻³.

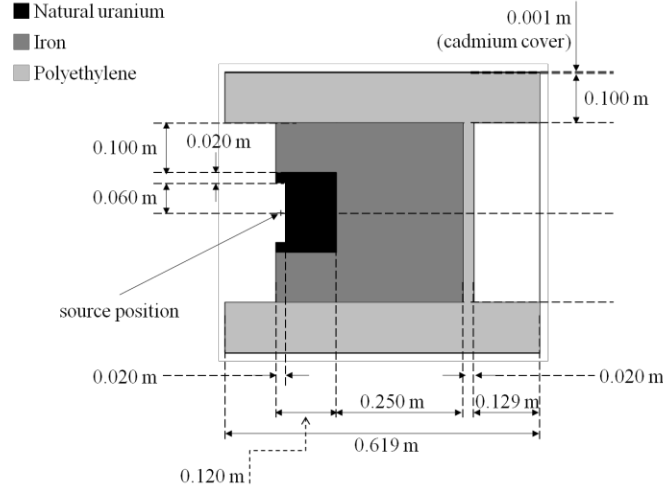


Figure 1. Cross sectional view of the cylindrical moderator assembly [1]. The dimensions are symmetric about the center axis.

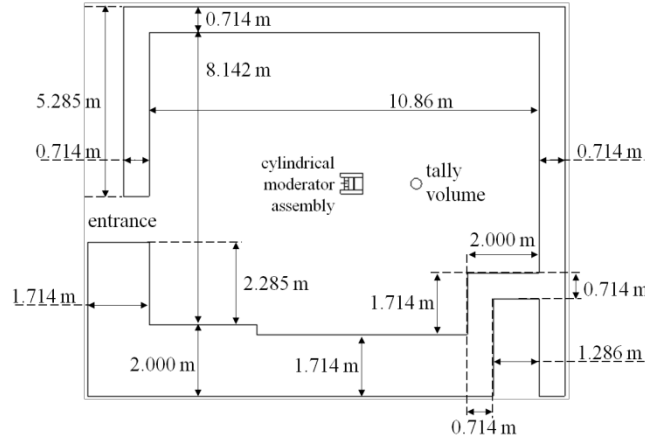


Figure 2. Cross section view of the Health Physics Neutron Generator facility as seen from above. Included are the placements of the cylindrical moderator assembly and tally volume.

The ^{252}Cf neutron emission spectra, given in Figure 3, is modelled as an isotropic point source using the Watt continuous fission energy distribution described by Equation (1). Here, $p(E)$ represents the number of neutrons per unit energy interval from E to $E + dE$, C a normalization constant such that $\int_0^\infty p(E)dE = 1$, $a = 1.025 \text{ MeV}$ and $b = 2.926 \text{ MeV}^{-1}$ [1, 5]. Figure 3 illustrates the ^{252}Cf neutron fluence rate distribution calculated at 1 m from this point source representation.

$$p(E) = Ce^{-E/a} \sinh(bE)^{1/2} \quad (1)$$

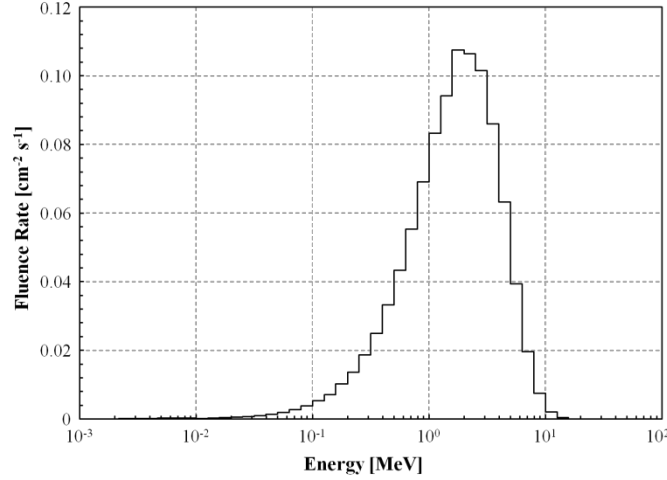


Figure 3. Watt continuous fission energy distribution for ^{252}Cf spontaneous fission neutron source at 1 m from the source.

The volume averaged, neutron fluence rate distribution in air is tallied inside a 30 cm diameter, spherical volume. The tally volume is centered behind the CMA at a distance of 2 m from the source and 1 m above the floor (see tally volume in Figure 2). The neutron fluence distribution is determined in units of cm^{-2} normalized per source neutron [4]. Moreover, the fluence distribution is calculated for neutron energies ranging from 7.940×10^{-10} MeV up to 1.994×10^1 MeV in 52 increments, equally spaced on a logarithmic scale. These energy bins are specifically chosen to correspond with measured Bonner Sphere spectra, thus, allowing for a direct comparison to be made [2, 3, 6]. For all simulations, default neutron physics models are implemented [4]. Furthermore, neutron cross sections based on the ENDF70 (Evaluated Nuclear Data File) nuclear data libraries, at room temperature, are utilized when available [4]. Thermal neutron scattering cross sections for polyethylene, available using the $S(\alpha, \beta)$ card, are also implemented [4]. Finally, a rectangular mesh-based variance reduction technique is used to speed up convergence and reduce simulation time.

3. Results

Section 3.1 details the moderated neutron fluence rate distribution calculated at the tally volume. An analysis of the fractional contributions to the total fluence rate, for selected energy ranges, is also included. In Section 3.2, the corresponding ambient dose equivalent rate distribution is determined.

3.1 Neutron fluence rate distribution

The ^{252}Cf source underwent an absolute measurement on February 18, 2010 [7]. On that date, the neutron emission rate was determined to be $2.3 \times 10^8 \text{ s}^{-1}$ as per the source certificate [7]. Based on a decay half-life of 2.645 years, the neutron emission rate on February 18, 2015 is calculated to be $6.2 \times 10^7 \text{ s}^{-1}$. Figure 4, then, illustrates the calculated neutron fluence rate distribution at 2 m from the source and 1 m above the floor, adjusted to February 18, 2015, in

units of $\text{cm}^{-2} \text{s}^{-1}$ as a function of neutron energy in MeV. Included in Table 1 are the fractional contributions to the total neutron fluence rate for selected energy ranges.

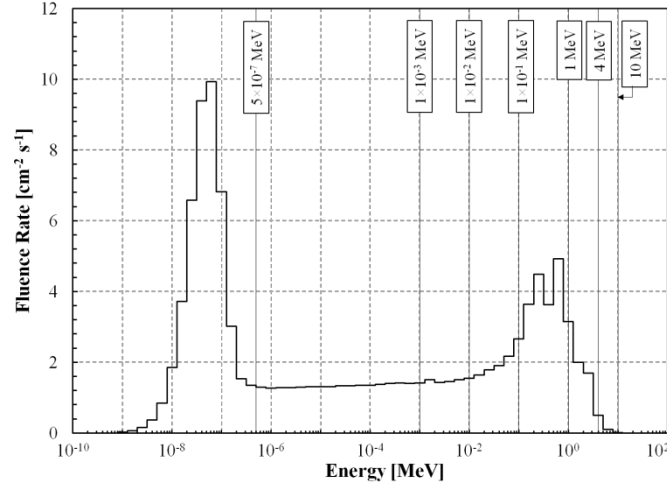


Figure 4. Moderated ^{252}Cf neutron fluence rate distribution at 2 m from the source and 1 m above the HPNG facility floor.

Table 1. Fractional neutron fluence rate distribution of moderated ^{252}Cf spectrum at 2 m from the source and 1 m above the HPNG facility floor.

Energy Range [MeV]	Neutron Fluence Rate [$\text{cm}^{-2} \text{s}^{-1}$]	Percent of Total Fluence Rate [%]
$< 5 \times 10^{-7}$	45.64	41.47
$5 \times 10^{-7} - 1 \times 10^{-3}$	22.70	20.62
$1 \times 10^{-3} - 1 \times 10^{-2}$	7.42	6.74
$1 \times 10^{-2} - 1 \times 10^{-1}$	10.16	9.24
$1 \times 10^{-1} - 1$	19.83	18.02
$1 - 4$	4.19	3.81
$4 - 10$	0.12	0.11
> 10	0.00	0.00
Total	110.06	100.00

The thermal neutron energy range less than 5×10^{-7} MeV accounts for 41% of the total neutron fluence rate. The next energy group, from 5×10^{-7} to 1×10^{-3} MeV, contributes 21% of the total fluence rate. This contribution is nearly equaled by neutrons found in the 0.1 to 1 MeV energy range, which forms 18% of the total fluence rate. The majority of the remaining contribution, at 16%, is given by neutrons with energies between 1×10^{-3} MeV and 0.1 MeV. Neutrons with energies greater than 1 MeV contribute the remaining 4%. Overall, the total neutron fluence rate at the tally point is determined to be $110 \text{ cm}^{-2} \text{s}^{-1}$, measured at 2 m from the source and 1 m above the HPNG facility floor.

3.2 Neutron ambient dose equivalent rate distribution

ICRP (International Commission on Radiological Protection) publication 74 fluence-to-ambient dose equivalent conversion factors are used to estimate the ambient dose equivalent rate expected from the moderated spectra [8]. A cubic spline interpolation technique, determined at the midpoint of each of the 52 energy bins, was used to estimate the appropriate conversion factors which are illustrated in Figure 5. These are then folded through with the neutron fluence rate distribution discussed in Section 3.1.

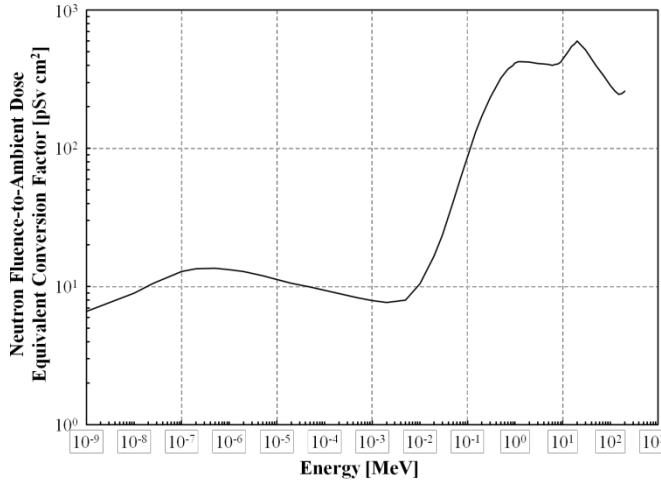


Figure 5. ICRP-74 fluence-to-ambient dose equivalent conversion factors [8].

Fast neutrons, with energies 1 MeV and higher, contribute significantly more to the neutron ambient dose equivalent rate than less energetic neutrons. For instance, for neutrons with energies less than 10^{-2} MeV, the conversion factors are on the order of 10 pSv cm². For neutrons greater than 1 MeV, in contrast, they are on the order of 100 pSv cm². This can have an effect on the manner in which radiation dose is delivered as is highlighted in Figure 6 and Table 2.

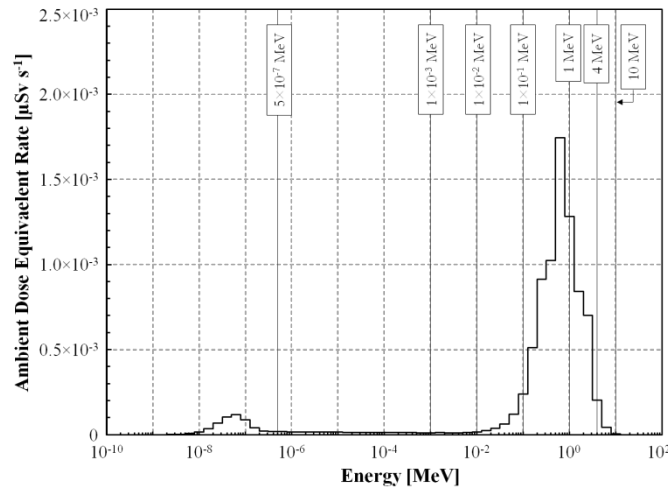


Figure 6. Moderated ²⁵²Cf neutron ambient dose equivalent rate distribution at 2 m from the source and 1 m above the HPNG facility floor.

For instance, even though thermal neutrons ($<5 \times 10^{-7}$ MeV) form 41% of the total neutron fluence rate, they are only responsible for 6% of the total ambient dose equivalent rate expected. Likewise, neutrons from 1×10^{-3} to 1×10^{-2} MeV are responsible for less than 3% of the total dose rate, despite forming 21% of the total fluence rate. In contrast, neutrons in the fast region, from 0.1 to 1 MeV, form 18% of the total fluence rate but contribute nearly 64% of the total ambient dose equivalent rate. Similarly, fast neutrons greater than 1 MeV contribute 4% of the total fluence rate yet are responsible for nearly 20% of the total ambient dose equivalent rate. The total ambient dose equivalent rate, measured at 2 m from the source and 1 m above the HPNG facility floor, is determined to be $8.58 \times 10^{-3} \mu\text{Sv s}^{-1}$.

Table 2. Fractional neutron ambient dose equivalent rate distribution of moderated ^{252}Cf spectrum at 2m from the source, 1 m above the floor.

Energy Range [MeV]	Neutron Ambient Dose Equivalent Rate [$\mu\text{Sv s}^{-1}$]	Percent of Total Ambient Dose Equivalent Rate [%]
$<5 \times 10^{-7}$	5.19×10^{-4}	6.05
$5 \times 10^{-7} - 1 \times 10^{-3}$	2.40×10^{-4}	2.80
$1 \times 10^{-3} - 1 \times 10^{-2}$	6.39×10^{-5}	0.75
$1 \times 10^{-2} - 1 \times 10^{-1}$	4.83×10^{-4}	5.63
$1 \times 10^{-1} - 1$	5.48×10^{-3}	63.83
1 – 4	1.75×10^{-3}	20.37
4 – 10	4.91×10^{-5}	0.57
>10	2.13×10^{-7}	2.48×10^{-3}
Total	8.58×10^{-3}	100.00

4. Discussion

In the original feasibility study carried out, the moderated DD neutron field was characterized by Bonner Sphere measurement. Table 3 lists the measured, fractional contributions to the total fluence rate and ambient dose equivalent rate determined in that work. The spectrum generated was ultimately shown to be CANDU[®]-Like. That conclusion was achieved after direct comparison against measured spectra from several CANDU[®] power stations [2, 3]. By extension then, in comparing results from this work against that moderated DD neutron field, an assessment can be made of its likeness to CANDU[®] fields.

The moderated DD spectrum in that work had a measured total fluence rate of $350 \pm 70 \text{ cm}^{-2} \text{ s}^{-1}$ and a total ambient dose equivalent rate of $119 \pm 24 \mu\text{Sv h}^{-1}$ [2, 3]. These values were measured at $100 \pm 1 \text{ cm}$ from the source target in the generator, and normalized to an external monitor count rate of 5381 cps [2, 3]. Monte Carlo simulations of these same quantities were found to be $351 \pm 35 \text{ cm}^{-2} \text{ s}^{-1}$ and $123 \pm 12 \mu\text{Sv h}^{-1}$, respectively. These Monte Carlo estimates assumed an effective neutron emission rate from the DD generator of $8.7 \times 10^7 \text{ s}^{-1}$ [2]. There is excellent agreement between MCNP calculations and the measured spectra for this DD generated field. The fluence and ambient dose equivalent rates measured in this work were assessed at 2 m from the source, and found to $110 \text{ cm}^{-2} \text{ s}^{-1}$ and $31 \mu\text{Sv hr}^{-1}$, respectively. The neutron emission rate for the ^{252}Cf source was assessed as $6.2 \times 10^7 \text{ s}^{-1}$ which is

comparable to the DD generator. At 1 m from the source, the total fluence and ambient dose equivalent rates could be roughly approximated as being 4 times larger, and therefore approach equivalent levels produced by the DD generator. These values, however, are ultimately not our primary consideration. Rather, when calibrating dosimeter response functions, the priority is to accurately represent the shape of the fluence rate distribution itself.

For brevity and clarity in the discussion to follow, the moderated spectrum which used the DD neutron generator as the primary neutron source will be referred to as field-dd. The spectrum generated in this work, which uses ^{252}Cf as the primary neutron source, will be referred to as field-cf.

Table 3. Fractional fluence and ambient dose equivalent dose rate distributions of moderated DD neutron spectrum measured at 1 m from the source target [2, 3].

Energy Range [MeV]	Percent of Total Fluence Rate [%]	Percent of Total Ambient Dose Equivalent Rate [%]
$<5 \times 10^{-7}$	34	4.2
$5 \times 10^{-7} - 1 \times 10^{-3}$	30.5	3.4
$1 \times 10^{-3} - 1 \times 10^{-2}$		
$1 \times 10^{-2} - 1 \times 10^{-1}$	7.9	3.6
$1 \times 10^{-1} - 1$	20.3	56.2
1 – 4	7.1	31.8
4 – 10	0.2	0.7
>10	0.00	0.00

Upon examination of Table 1 and Table 3, an argument can be made that fluence rate distribution calculated in this work is similar to that generated using the DD generator. For instance, thermal neutrons with energies less than 5×10^{-7} MeV form nearly 34% of the total fluence rate in field-dd; for field-cf, that contribution is slightly higher at 41%. Moving towards higher energies, neutrons from 5×10^{-7} to 1×10^{-2} MeV form 31% of the total fluence rate in field-dd. The equivalent contribution in field-cf approaches 27%. From 0.01 to 0.1 MeV, field-dd and field-cf neutrons form 8% and 9% of their respective total fluence rates. Fast neutron contributions, from 0.1 to 1 MeV, were found to be 20% for field-dd and 18% for field-cf. Finally, neutrons with energies greater than 1 MeV contribute nearly 7% to the total neutron fluence rate in field-dd, whereas the equivalent contribution in field-cf approaches 4%.

The ambient dose equivalent rate distribution provided a similar narrative. However, it is clear that a slightly harder field-dd spectrum compared to field-cf spectrum is observed. For instance, field-dd neutrons with energies ranging from 0.1 to 1 MeV contribute 56% of the total ambient dose equivalent rate observed in that study. In comparison, the equivalent field-cf contribution is nearly 10% larger, at 64% (Table 2). At higher energies between 1 and 4 MeV, however, field-dd neutrons are responsible for 31% of the total dose rate. Field-cf neutrons in this energy range, in contrast, only contribute 20%.

These results also demonstrate that minor variations in the fluence rate distribution, particularly at high energies, can impact the amount of dose delivered. This is important because CANDU[®] fields are known to vary somewhat between facilities and also within individual power stations [1, 2, 3, 6]. Given that no neutron dosimeter is uniformly sensitive to the entire neutron energy range observed, these variations can result in uncertainties in measured dose. This highlights the need for calibrating dosimeters using CANDU[®]-Like fields that are reasonable, artificial representatives of CANDU[®] fields.

To highlight this, at the Darlington power plant reactivity deck, neutrons with energies ranging from 0.1 to 1 MeV contribute 20% to the total fluence rate measured. This same contribution is responsible for nearly 42% of the total ambient dose equivalent rate received [6]. At the Darlington primary heat transport pump, in contrast, contributions to the total fluence rate is much larger at 46%. Correspondingly, then, contribution to the total ambient dose equivalent rate approached 82% [6]. The fluence rate and ambient dose equivalent rate distributions calculated for field-cf fall within the range of these measured values.

Continuing on, only 15% of the total neutron fluence rate at the Darlington reactivity deck was found in the thermal energy regime, while at the primary heat transport pump, only 4% was [6]. This translates into 2%, and less than 1%, of the total ambient dose equivalent rate measured at these locations [6]. Comparatively, for field-cf, 42% of the total neutron fluence rate is found in the thermal neutron energy regime. This translates into nearly 6% of the total ambient dose equivalent rate calculated. This contribution is relatively large when compared to the two Darlington sites, however, this same contribution remains less than that measured at the Pickering power plant moderator dump valve room. At this latter location, nearly 48% of the total neutron fluence rate was assessed as thermal [6]. While this is more representative of field-cf, it should also be mentioned that the measured spectrum is relatively soft. That is, up to 21% of the total ambient dose equivalent rate is due to thermal neutrons [6]. Finally, the results highlighted in this work also illustrate the minimal significance that thermal neutrons generally have on the dose delivered.

The neutron fluence rate distribution calculated in this work for field-cf is similar to the previously characterized field-dd, and is also comparable to measured CANDU[®] distributions. Furthermore, the expected ambient dose equivalent rate distributions determined for field-cf also agree. For this reason, it is concluded that field-cf can be equally well considered as CANDU[®]-Like as field-dd and is a very good representation of a CANDU[®]-Like field.

5. Conclusion

A Monte Carlo based feasibility study successfully demonstrated that a CANDU[®]-Like field could be artificially generated at Canadian Nuclear Laboratories' Health Physics Neutron Generator facility. This was achieved by moderating the primary spectrum of a ²⁵²Cf neutron source using a specially designed cylindrical moderator assembly. The moderator assembly was previously constructed to moderate neutrons emanating from a deuterium-deuterium neutron generator into a CANDU[®]-Like field. Though successful, the neutron generator has since been decommissioned justifying this alternative methodology. The resulting neutron fluence rate distribution assessed in this work had the desired bimodal characteristics of measured CANDU[®] fields. This artificially created CANDU[®]-Like field, then, could be used

to calibrate neutron dosimeters worn by Nuclear Energy Workers at CANDU[®] power stations. With the expected ambient dose equivalent rate known, variations in measured dose values can then be quantified. Correction factors can then be implemented to improve dosimeter responses and, ultimately, radioprotection practices.

6. Acknowledgements

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7. References

- [1] J. C. Nunes, W. G. Cross and A. J. Waker, "Feasibility of creating 'CANDU-LIKE' workplace neutron fields in an existing irradiation facility," *Radiation Protection Dosimetry*, vol. 72, no. 1, pp. 11-20, 1997.
- [2] J. C. Nunes and R. T. Faught, "Characterization of laboratory-produced CANDU-like workplace neutron fields," *Radiation Protection Dosimetry*, vol. 95, no. 3, pp. 231-238, 2001.
- [3] J. C. Nunes and R. T. Faught, "COG-00-121 - Characterization of CANDU-LIKE Workplace Neutron Fields Produced in a Laboratory at CRL," CANDU Owners Group, 2001.
- [4] LANL, "MCNPX 2.7.0 Extension Technical Report LA-UR-11-02295," Los Alamos National Laboratory, 2011.
- [5] J. Atanackovic, "TN-10-3003 Monte Carlo Characterization of a New Cf-252 Neutron Source at AECL," CANDU Owners Group, 2011.
- [6] J. C. Nunes, A. J. Waker and A. Arneja, "Neutron spectrometry and dosimetry in specific locations at two CANDU power plants," *Radiation Protection Dosimetry*, vol. 63, no. 2, pp. 87-104, 1996.
- [7] J. Atanackovic, "Characterization of neutron fields coming from bare and heavy water moderated Cf-252 spontaneous fission source using Bonner sphere spectrometer and set of ISO shadow cones," *Applied Radiation and Isotopes*, vol. 99, pp. 122-132, 2015. .
- [8] ICRP, "ICRP74: Conversion coefficients for use in radiological Protection against external radiation," International Commission on Radiological Protection, 1996.