Annals of Nuclear Energy 114 (2018) 586–591

Contents lists available at ScienceDirect

# Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Technical note

# Improving the efficiency of shielding container design calculations

Doddy Kastanya

*Safety and Licensing Department, Candesco Division of Kinectrics Inc., 26 Wellington Street East, Toronto, ON M5E 1S2, Canada*

ARTICLE INFO

*Article history:*
Received 28 August 2017
Received in revised form 24 November 2017
Accepted 7 December 2017
Available online 3 February 2018

*Keywords:*
MCNP
Shielding container
Transportation of radioactive materials
CANDU refurbishment

ABSTRACT

The reactor refurbishment project for a CANDU (Canada Deuterium Uranium) reactor is an example of a major reactor-maintenance-related project where significant amount of non-fuel radioactive materials are generated. The transportation of these radioactive materials to an off-site location for further processing must be done in a safe way following applicable International Atomic Energy Agency (IAEA) regulations. The shielding container used for transporting these radioactive materials must be designed such that the dose rate requirements in the vicinity of the container are met. In the process of designing such container, the designer needs to take into account various possible configurations of radioactive materials that will be loaded into the container. Even in a stylized fashion, the numbers of possible combinations of radioactive materials placement inside the container are simply astronomical and the evaluations of all possible combinations, given the current state of computational power, are simply prohibitive. In this paper, an alternative methodology is proposed where the principle of super-position is explored to aid the design process of the container. The overall dose rates in the area surrounding the radioactive materials are determined using the dose rates from the constituent radionuclides and the source strength of each type of the radioactive components being loaded into the container. Using this alternative approach, a significantly higher number of potential configurations can be evaluated in a relatively short period of time, which directly improves the robustness of the final container design.



## 1. Introduction

A CANDU (Canada Deuterium Uranium) reactor is designed to have around a lifetime of over 50 years, with a major refurbishment project around its midlife to replace some of the key components. Several CANDU refurbishment projects have been executed in the recent years in Canada namely at the Point Lepreau Generating Station in the province of New Brunswick (GNB, 2008) and at the Darlington Nuclear Generating Station in the province of Ontario (Leslie, 2016). From such refurbishment project, a significant amount of low-level and intermediate-level radioactive materials are generated. The transportation of these radioactive materials to an off-site location for further processing must be done in a safe way following applicable International Atomic Energy Agency (IAEA) regulation (IAEA, 2012). The shielding container used for transporting these radioactive materials must be designed such that the dose rate requirements in the vicinity of the container as specified in the regulation are met.

In the design process of such shielding containers, the designer might not know exactly the composition of radioactive material which will be loaded into the container. It is true that the limiting configuration could be considered; however, the final design might prove to be too conservative for most applications. Moreover, the associated costs for manufacturing such container will be on the high side as well. Therefore, it would be useful for the designer to know the range of dose rates which could be generated from typical loading configurations. Unfortunately, the number of different typical configurations to be considered is very high which makes it very impractical to evaluate every single one of these configurations.

In order to circumvent this challenge, the idea of super-position is explored to aid the design process of the shielding container. The overall dose rates in the area surrounding the radioactive materials are determined using the dose rates from the constituent radionuclides and the source strength of each type of the radioactive components being loaded into the container. Using this alternative approach, a significantly higher number of potential configurations can be evaluated which directly improves the robustness of the final container design.

The objective of this paper is to introduce an alternative approach to provide quick estimates of expected dose rates in the vicinity of a particular shielding container, during the design process of such container. Based on these dose rates estimates, the designer can determine the appropriate thickness for the shielding container which will meet the regulatory requirements. The details of the methodology will be presented in the next

*E-mail address:* Doddy.Kastanya@candesco.com

https://doi.org/10.1016/j.anucene.2017.12.020
0306-4549/

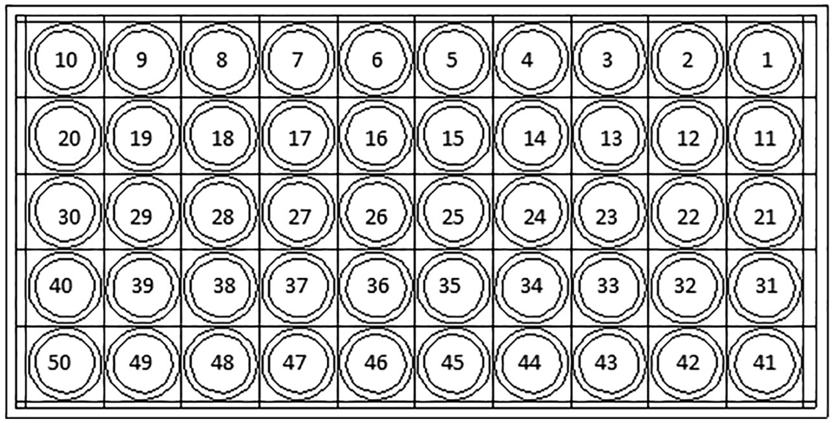
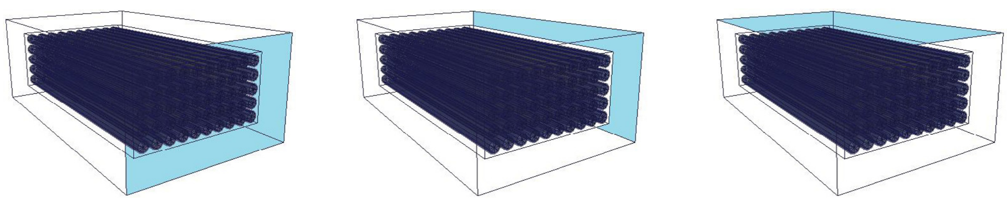



**Fig. 1.** Arrangement of the cylindrical tubes (viewed from the short-end of the configuration).



(a) Short-end (b) Side (c) Top

**Fig. 2.** Different faces of the configuration.

section. However, it is important to note that, without loss of generality, some simplifications are introduced in relation to the number of radionuclides considered and the source strength of numerous radioactive components. The extensions of this methodology to more radionuclides and increasing variation in source strength should be straight forward.

## 2. Methodology

The calculations to determine the dose rates in various points around the shielding container are done using the MCNP 5 code Version 1.60 (LANL, 2003). For the purpose of demonstrating the methodology for getting quick and reasonable estimates of dose rates, the radioactive materials considered in this study come as hollow cylindrical tubes. This particular geometry is chosen to represent specific wastes from CANDU refurbishment project such as feeder tubes, calandria tubes, or other heat transport system (HTS) pipes. The specific configuration considered in this study consists of 50 cylindrical tubes arranged in a 5 × 10 array (see Fig. 1). The gamma source is defined uniformly as a very thin coating in the inside of each tube. For the purpose of current study, only four gamma-emitting nuclides are considered namely $^{60}$Co, $^{95}$Nb, $^{95}$Zr, and $^{124}$Sb. For each nuclide, 50 MCNP runs are executed where each one of them represents a case where the source is located in just one tube (*i.e.*, only in tube 1, tube 2, etc. – see Fig. 1 for tube numbering). The density of the source for each nuclide is set to 1 Bq/cm$^2$. This approach allows the final calculated dose rate to be scaled up based on the source strength of each nuclide. Using this approach, additional nuclides can be easily incorporated into the model since for the current configuration of 50 tubes, one additional nuclide translates into 50 additional MCNP runs to generate the nuclide-based dose rates at points of interest.

The dose rates are calculated at the center of three faces of the configurations. The three faces considered are short-end, side, and top (see Fig. 2). For each face, three different locations have been picked to measure the dose rates namely 1 cm (to represent "on contact" dose rate), 1 m, and 2 m away from the imaginary box enclosing the configuration.[1]

**Table 1**
Hypothetical surface activities of wastes.

| Nuclide | Type 1 (Bq/cm$^2$) | Type 2 (Bq/cm$^2$) | Type 3 (Bq/cm$^2$) | Type 4 (Bq/cm$^2$) |
|---|---|---|---|---|
| 60Co | 70,000 | 52,500 | 28,000 | 7000 |
| 95Nb | 10,100 | 7575 | 4040 | 1010 |
| 95Zr | 470 | 352.50 | 188 | 47 |
| 124Sb | 58 | 43.50 | 23.20 | 5.80 |

For the current setup, the overall dose rate at a point of interest can be determined by using the following formula:

$$DR = \sum_{i=1}^{4}\sum_{n=1}^{50} S_{i,n} d_{i,n} \tag{1}$$

where $i$ denotes the nuclides, $n$ denotes the node location, $S_{i,n}$ denotes the source strength for nuclide $i$ at node number $n$, and $d_{i,n}$ denotes the dose rate per unit source at the point of interest from executing the case where the source originates at node number $n$ and the nuclide utilized is $i$. In order to confirm the accuracy this approach, several known configurations are tested where the dose rates from executing the MCNP problem with all sources considered are compared to the ones obtained using the proposed approach (*i.e.*, super-position). Knowing that the present problem is fairly linear in nature, the comparison between results obtained from the super-position approach and MCNP is done solely for

[1] The imaginary box is setup such that the surfaces of the outer most tubes (in the "top" and "side" direction) and the end of the tubes (in the "short-end" direction) are less than 0.25 cm inside the box.