Dose Rate Analysis to Support Radiological Characterization of Used CANDU Fuel

NWMO-TR-2022-03 April 2022

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ABSTRACT

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Report No.: **NWMO-TR-2022-03**
Author(s): Imelda Ariani
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Abstract

An accurate estimate of dose rates associated with different used fuel configurations is required to support the radiological characterization of used fuel during handling of the fuel as part of the long-term management of Canada's used nuclear fuel. In the current analysis, the effective dose rate calculations have been updated to reflect the latest radionuclide inventory estimates (Heckman and Edward 2020) and the latest container design and to include a more detailed representation of the within-bundle source term spatial distribution (by applying a ring-wise source term distribution instead of a uniform distribution).

Two geometry configurations were analysed: i) unshielded single used fuel bundle and ii) used fuel bundles inside a Used Fuel Container (UFC). Dose rate receptor locations up to 100 m from the source have been considered. Two burnup values were examined: 220 and 290 MWh/kgU. Decay times up to 1E+07 years were considered for the single used fuel bundle and single UFC configurations. It should be noted that the current Engineered Barrier System design is based on the heat generation of used 37-element CANDU fuel bundles with 10 years of decay time. Nevertheless, the zero-decay time case was included for information.

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LIST OF ABBREVIATIONS

AP	Antero-Posterior
APM	Adaptive Phase Management
CANDU	CANada Deuterium Uranium
DGR	Deep Geological Repository
ENDF	Evaluated Nuclear Data File
ICRP	International Commission on Radiological Protection
MCNP	Monte Carlo N-Particle
NWMO	Nuclear Waste Management Organization
UFC	Used Fuel Container

1. INTRODUCTION

1.1 Background

The Nuclear Waste Management Organization (NWMO) is responsible for implementing Adaptive Phased Management (APM), the federally approved plan for the safe long-term management of Canada's used nuclear fuel. Under the APM, used nuclear fuel will ultimately be placed within a Deep Geological Repository (DGR) in a suitable host rock formation. Key to the DGR design concept is the use of multiple natural and engineered barriers to isolate the nuclear waste from the environment. The Used Fuel Container (UFC) is a major engineered barrier that must be strong enough to withstand geological pressures, including the hydrostatic load of glaciation events, and chemically resistant to long-term corrosion. To support the radiological hazard assessment of used CANDU fuel bundles, dose rate estimates associated with an unshielded used fuel bundle and a UFC filled with used fuel bundles have been calculated.

1.2 Purpose and Scope of the Document

The present work scope documented in this report provides the effective dose rate calculations based on the latest estimate of used fuel radionuclide inventory (Heckman and Edward 2020) and the UFC design.

1.3 Limits and Applicability

Results documented in this report are based on specific fuel characteristics and container design. The conclusions drawn from the study are subject to change if the fuel or container design changes.

2. METHODOLOGY

2.1 General Methodology

The general methodology for estimating the effective dose rates outside of a used fuel bundle or a UFC involves estimating the radiation source emissions from used fuel (see Section 2.2) and simulating the particle transport from the source location to the receptor locations of interest (see Section 2.3).

A screening study examining a variety of CANDU fuel bundle parameters including the 28-element, modified 37-element, and regular 37-element fuel bundle designs and the long and standard-length fuel bundles (Heckman and Edward 2020) has been performed. Based on the study, the standard-length, regular 37-element fuel bundle has been selected as the NWMO reference used fuel bundle. Consequently, all calculations performed in this report apply the reference used fuel bundle characteristics. Two burnup values were examined: 220 MWh/kgU, which represents the highest median burnup of any decade for used fuel from all Canadian CANDU reactors and 290 MWh/kgU, which represents the highest 95th percentile burnup of any decade from all Canadian CANDU reactors (Heckman and Edward 2020). Up to twelve decay times ranging from zero to ten million years were analysed.

Additional details on the calculation models are provided in Appendix A. The modeling assumptions are listed in Section 2.4.

Two geometry configurations were analysed: i) a single used fuel bundle and ii) fuel bundles inside a UFC. Additional information on the cases being analysed are given in Section 2.5.

2.2 Radiation Sources

Used CANDU fuel contains radioactive nuclides which can emit ionizing radiation in the form of alpha particles, beta particles, gamma rays, and neutrons. For the current analysis, the external dose rates from alpha and beta particles are ignored due to the particles' short range. The source terms from the gamma and neutron emissions were estimated using the ring-wise radionuclide inventory predicted using the ORIGEN-S code (Heckman and Edward 2020). The ring-wise designations are illustrated in Figure 1.

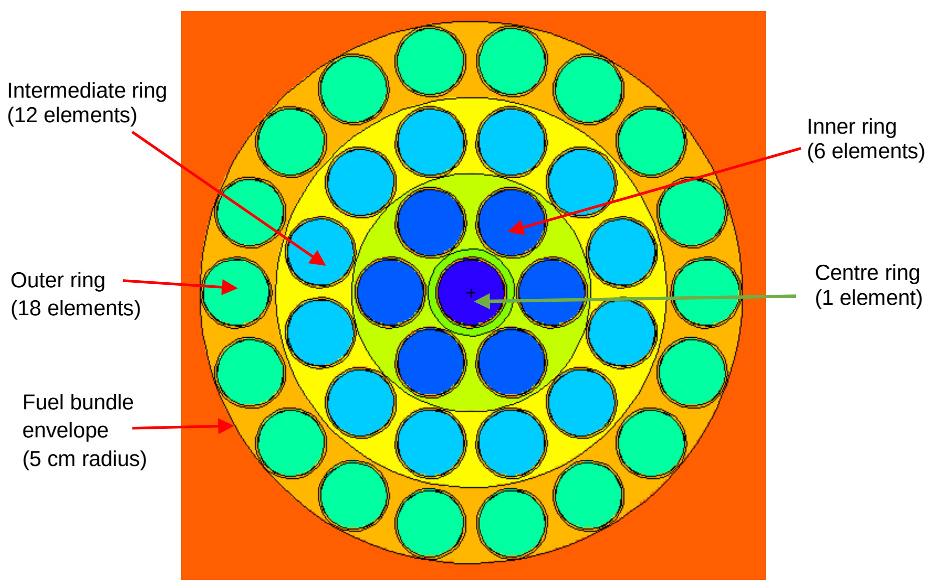


Figure 1: Ring-Wise Designation in Reference Used CANDU Fuel Bundle Model

2.2.1 Gamma Emissions

Ring-wise gamma emission spectra from Heckman and Edward 2020 were applied to the dose rate calculation models. The gamma source intensity and average energy in each ring at different burnup and decay times are listed in Table 1 and Table 2, respectively. The gamma emissions were binned into 20 energy groups. The gamma emission energy spectra at different decay times are shown in Figure 2. The gamma source intensities at different burnup and decay times are shown in Figure 3.

The principal gamma sources are:

- Various short-lived fission products at discharge,
- Ba-137m for 10 to 200 years decay time,
- Am-241 for 300 to 1000 years decay time,
- Pu-240 for 10,000 years decay time,
- Th-229, Pa-233, Np-237 for 1E+05 to 1E+06 years decay time, and
- Bi-214, Pb-214 for 1E+07 years decay time.

Table 1: Gamma Source Intensity

		Gamma/s Source Intensity per Fuel Element			
Bundle Burnup	**Decay Time (Years)**	**Centre Ring**	**Inner Ring**	**Intermediate Ring**	**Outer Ring**
220 MWh/kgU	0	5.52E+15	5.80E+15	6.47E+15	7.96E+15
	10	6.22E+11	6.54E+11	7.32E+11	8.98E+11
	30	3.65E+11	3.84E+11	4.27E+11	5.19E+11
	100	7.70E+10	8.10E+10	9.06E+10	1.11E+11
	200	1.35E+10	1.44E+10	1.64E+10	2.13E+10
	300	6.52E+09	7.02E+09	8.17E+09	1.11E+10
	500	4.39E+09	4.74E+09	5.54E+09	7.58E+09
	1E+03	2.24E+09	2.41E+09	2.80E+09	3.80E+09
	1E+04	2.42E+08	2.54E+08	2.82E+08	3.45E+08
	1E+05	4.50E+07	4.61E+07	4.89E+07	5.60E+07
	1E+06	4.61E+07	4.71E+07	4.95E+07	5.57E+07
	1E+07	3.11E+07	3.11E+07	3.12E+07	3.14E+07
290 MWh/kgU	0	5.59E+15	5.86E+15	6.51E+15	7.96E+15
	10	8.20E+11	8.60E+11	9.56E+11	1.17E+12
	30	4.76E+11	4.98E+11	5.51E+11	6.64E+11
	100	1.02E+11	1.06E+11	1.18E+11	1.43E+11
	200	1.88E+10	1.99E+10	2.23E+10	2.80E+10
	300	9.54E+09	1.02E+10	1.15E+10	1.49E+10
	500	6.49E+09	6.92E+09	7.87E+09	1.02E+10
	1E+03	3.29E+09	3.50E+09	3.97E+09	5.12E+09
	1E+04	3.31E+08	3.46E+08	3.82E+08	4.66E+08
	1E+05	5.37E+07	5.52E+07	5.86E+07	6.70E+07
	1E+06	5.30E+07	5.43E+07	5.72E+07	6.43E+07
	1E+07	3.13E+07	3.14E+07	3.14E+07	3.17E+07

Table 2: Gamma Source Average Energy

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Gamma Particles			
		Centre	Inner	Intermediate	Outer
220 MWh/kgU	0	0.48	0.48	0.47	0.46
	10	0.39	0.39	0.40	0.40
	30	0.39	0.39	0.39	0.39
	100	0.36	0.36	0.36	0.36
	200	0.22	0.22	0.22	0.21
	300	0.08	0.08	0.08	0.07
	500	0.04	0.04	0.04	0.04
	1E+03	0.04	0.04	0.04	0.04
	1E+04	0.04	0.04	0.04	0.04
	1E+05	0.23	0.23	0.22	0.21
	1E+06	0.30	0.30	0.29	0.27
	1E+07	0.39	0.39	0.39	0.39
290 MWh/kgU	0	0.47	0.47	0.46	0.46
	10	0.40	0.40	0.41	0.41
	30	0.39	0.39	0.40	0.40
	100	0.36	0.36	0.37	0.37
	200	0.22	0.21	0.21	0.21
	300	0.07	0.07	0.07	0.07
	500	0.04	0.04	0.04	0.04
	1E+03	0.04	0.04	0.04	0.04
	1E+04	0.04	0.04	0.04	0.05
	1E+05	0.22	0.22	0.21	0.21
	1E+06	0.28	0.27	0.26	0.24
	1E+07	0.39	0.39	0.39	0.39

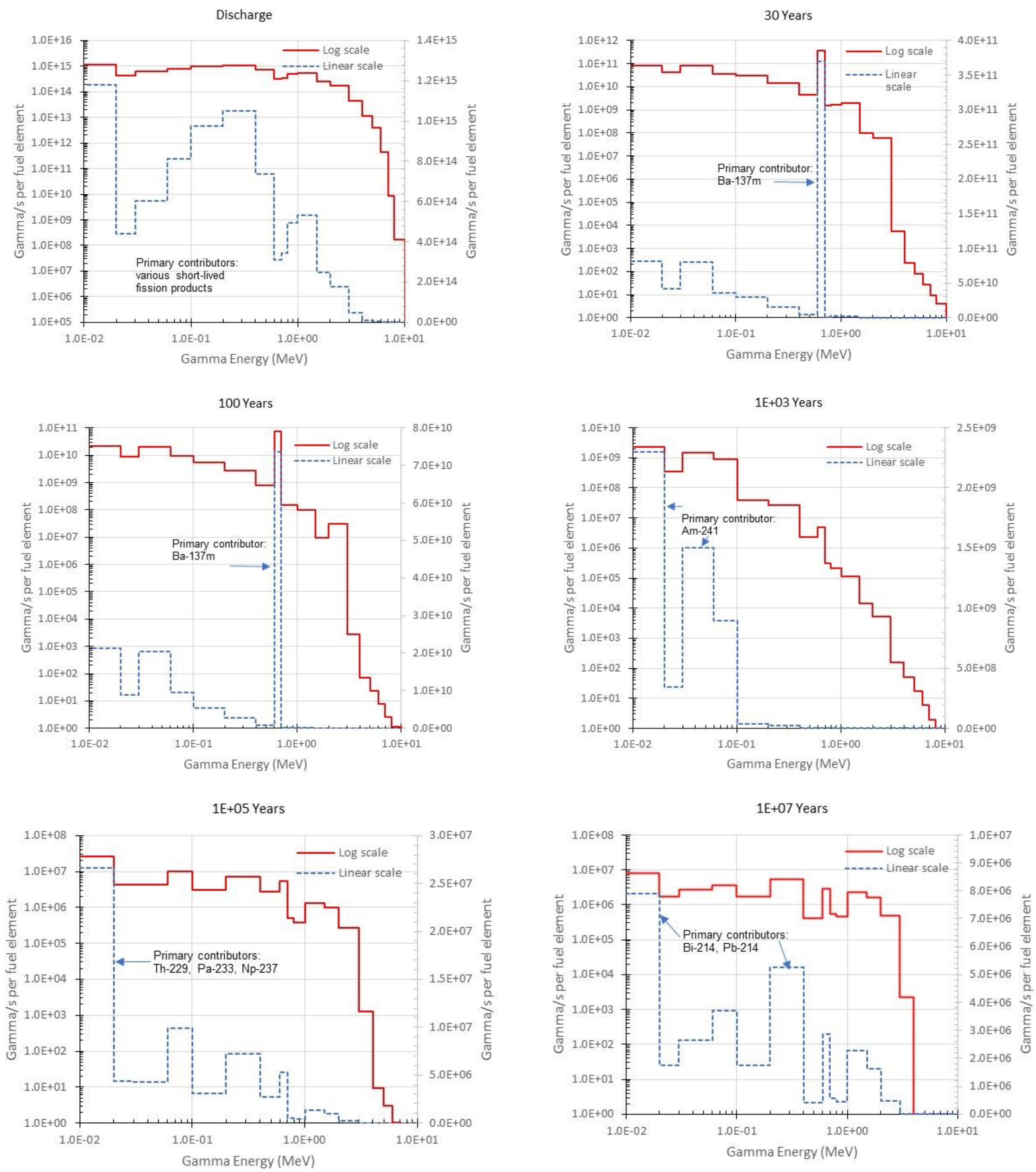


Figure 2: Gamma Emission Spectra at Various Decay Times

Note: Values plotted above correspond to the outer ring fuel elements from a bundle with 290 MWh/kgU burnup.

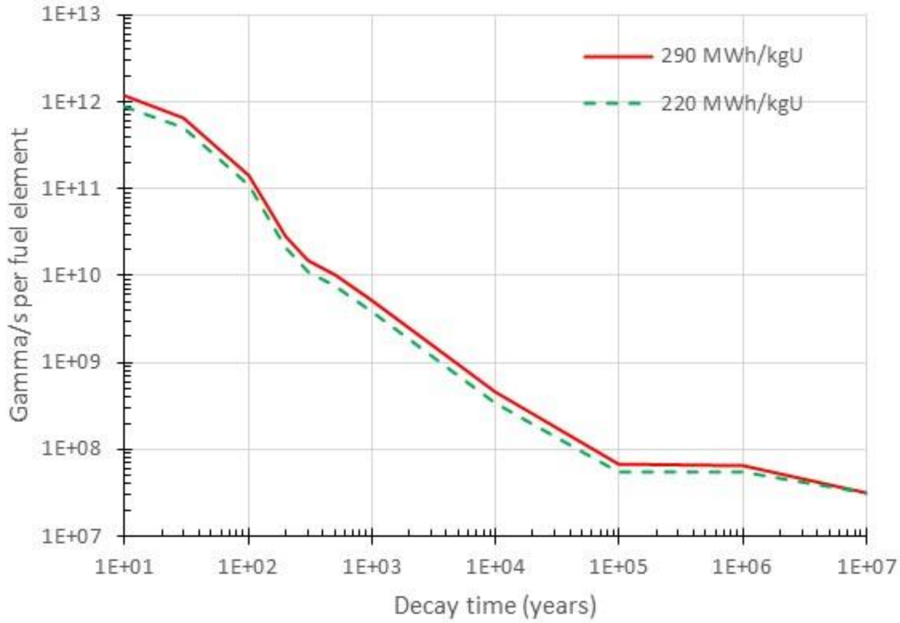


Figure 3: Gamma Source Intensity at Various Decay Times

Note: Values plotted above correspond to the outer ring fuel elements.

2.2.2 Neutron Emissions

Ring-wise neutron emission spectra from Heckman and Edward 2020 were applied to the dose rate calculation models. The neutron source intensity and average energy in each ring at different burnup and decay times are listed in Table 3 and Table 4, respectively. Neutron emissions were binned into 46 energy groups. The neutron emission spectra at different decay times are shown in Figure 4. The neutron source intensity at different burnup and decay times is shown in Figure 5.

Table 3: Neutron Source Intensity

		Neutron/s Source Intensity per Fuel Element			
Bundle Burnup	**Decay Time (Years)**	**Centre Ring**	**Inner Ring**	**Intermediate Ring**	**Outer Ring**
220 MWh/kgU	0	6.19E+12	6.37E+12	6.80E+12	7.62E+12
	10	1.54E+03	1.75E+03	2.34E+03	4.35E+03
	30	1.27E+03	1.40E+03	1.76E+03	2.89E+03
	100	9.87E+02	1.05E+03	1.21E+03	1.60E+03
	200	9.11E+02	9.69E+02	1.11E+03	1.42E+03
	300	8.67E+02	9.22E+02	1.05E+03	1.35E+03
	500	8.04E+02	8.53E+02	9.71E+02	1.24E+03
	1E+03	7.03E+02	7.46E+02	8.46E+02	1.07E+03
	1E+04	2.99E+02	3.18E+02	3.64E+02	4.75E+02
	1E+05	4.59E+01	5.16E+01	6.70E+01	1.13E+02
	1E+06	1.50E+01	1.61E+01	1.91E+01	2.79E+01
	1E+07	8.03E+00	8.03E+00	8.03E+00	8.02E+00

Bundle Burnup	Decay Time (Years)	Neutron/s Source Intensity per Fuel Element Centre Ring	Inner Ring	Intermediate Ring	Outer Ring
290 MWh/kgU	0	5.99E+12	6.13E+12	6.50E+12	7.25E+12
	10	3.86E+03	4.52E+03	6.40E+03	1.29E+04
	30	2.59E+03	2.94E+03	3.92E+03	7.19E+03
	100	1.47E+03	1.57E+03	1.80E+03	2.41E+03
	200	1.31E+03	1.38E+03	1.55E+03	1.94E+03
	300	1.24E+03	1.31E+03	1.47E+03	1.84E+03
	500	1.15E+03	1.21E+03	1.36E+03	1.69E+03
	1E+03	1.00E+03	1.05E+03	1.18E+03	1.47E+03
	1E+04	4.31E+02	4.58E+02	5.23E+02	6.80E+02
	1E+05	8.46E+01	9.58E+01	1.25E+02	2.09E+02
	1E+06	2.24E+01	2.46E+01	3.01E+01	4.61E+01
	1E+07	8.02E+00	8.02E+00	8.02E+00	8.01E+00

Table 4: Neutron Source Average Energy

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Neutron Centre Ring	Inner Ring	Intermediate Ring	Outer Ring
220 MWh/kgU	0	0.41	0.41	0.41	0.41
	10	2.10	2.11	2.11	2.12
	30	2.12	2.12	2.12	2.12
	100	2.12	2.12	2.12	2.12
	200	2.10	2.10	2.10	2.11
	300	2.09	2.09	2.09	2.10
	500	2.07	2.07	2.07	2.07
	1E+03	2.04	2.04	2.04	2.04
	1E+04	2.02	2.02	2.01	2.01
	1E+05	1.97	1.97	1.97	1.97
	1E+06	1.90	1.90	1.92	1.94
	1E+07	1.80	1.80	1.80	1.80
290 MWh/kgU	0	0.41	0.41	0.42	0.42
	10	2.12	2.12	2.12	2.12
	30	2.12	2.12	2.12	2.13
	100	2.12	2.12	2.12	2.12
	200	2.10	2.10	2.10	2.11
	300	2.09	2.09	2.09	2.09
	500	2.07	2.07	2.07	2.07
	1E+03	2.04	2.04	2.04	2.04
	1E+04	2.01	2.01	2.01	2.00
	1E+05	1.97	1.97	1.97	1.98
	1E+06	1.93	1.93	1.94	1.96
	1E+07	1.80	1.80	1.80	1.80

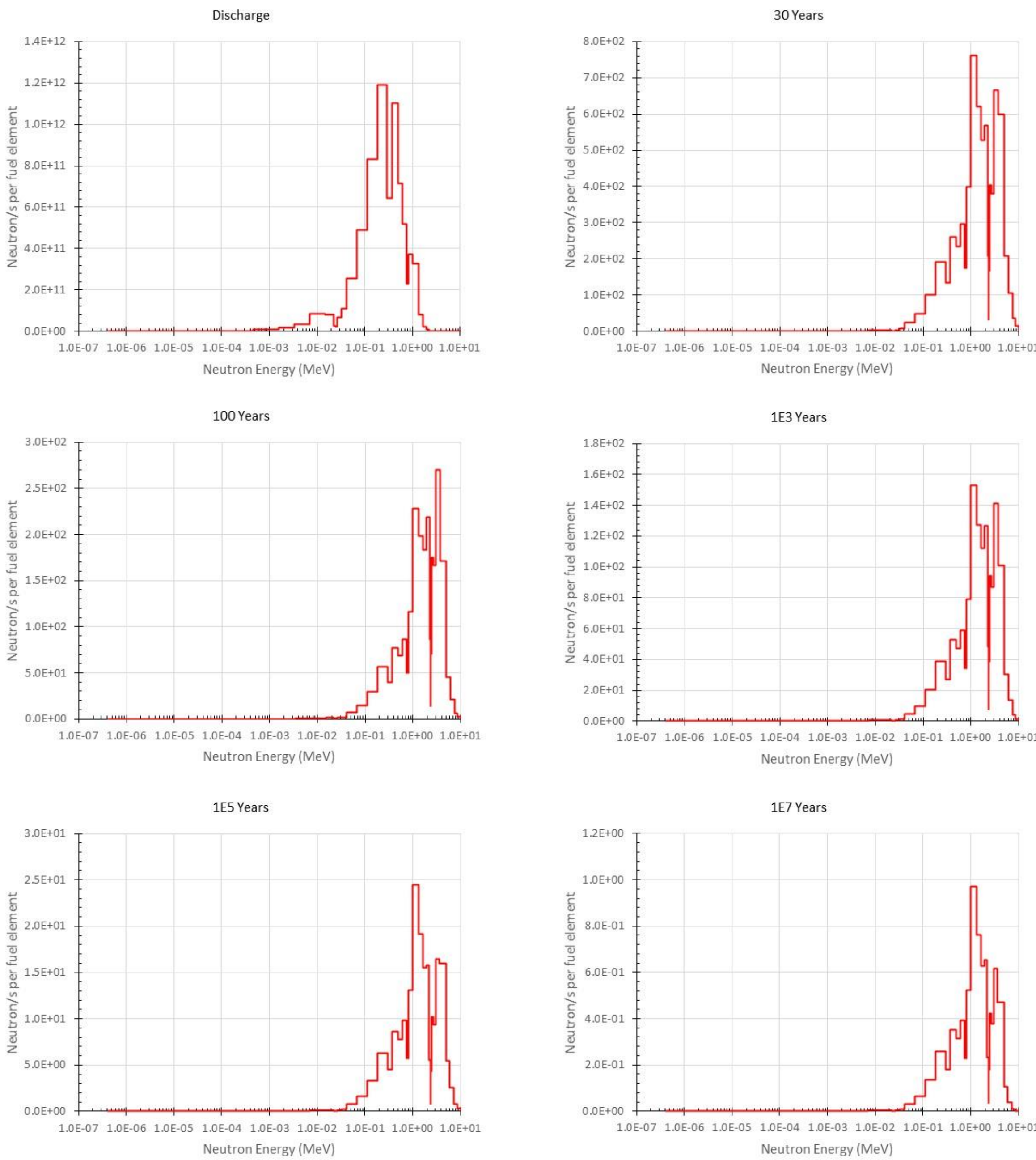


Figure 4: Neutron Emission Spectra at Various Decay Times

Note: Values plotted above correspond to the outer ring fuel elements from a bundle with 290 MWh/kgU burnup.

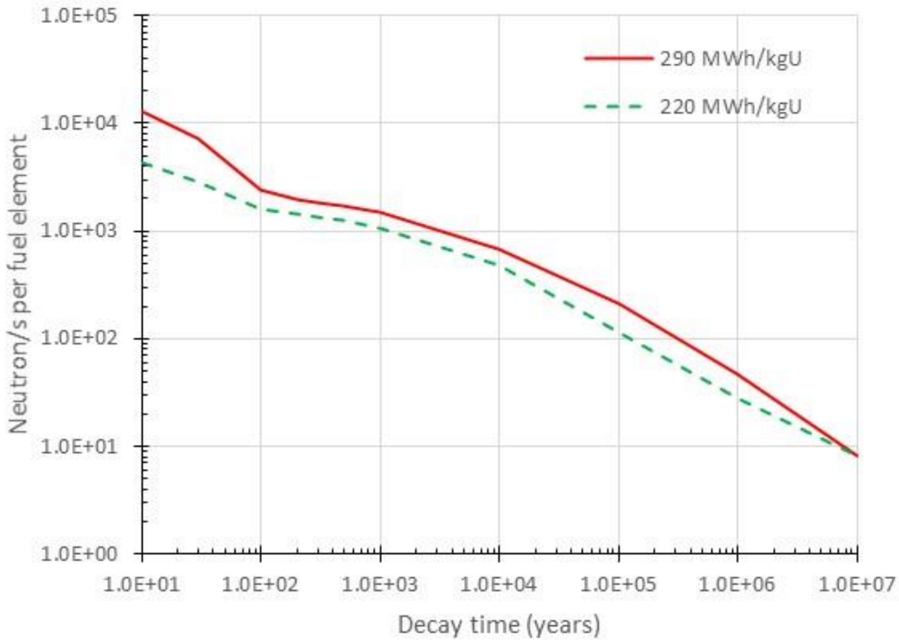


Figure 5: Neutron Source Intensity at Various Decay Times

Note: Values plotted above correspond to the outer ring fuel elements.

2.3 Particle Transport Calculations

The transport of gamma and neutron from the UO_2 fuel to the surrounding materials was simulated using the MCNP code version 6.1 (RSICC 2013), which applies a stochastic method of solving particle transport problems. Individual particles are simulated, tracked, and scored, instead of explicitly solving the transport equation. The Monte Carlo simulations create a series of life histories of the particles by using random sampling techniques to sample the probability laws that describe the real particle's behaviour. The simulation traces out, step by step, the particle's random walk through the medium. The history of a particle is followed until it no longer contributes information of interest to the problem at hand. Physical characteristics and interactions of photons and neutrons being considered in the MCNP simulations are discussed in following subsections.

2.3.1 Gamma Transport

Gamma photon transport calculations were performed using the photon (p) transport model (mode p) in MCNP for calculations in which the source terms are gammas from the fuel. For calculations in which the source terms are neutrons from the fuel and gammas are produced via neutron-capture reactions in the fuel and surrounding materials, the gamma transport is explicitly modeled by invoking the photon (p) mode along with the neutron (n) mode (i.e. mode n p in MCNP). The photon transport model considers photonuclear interactions, Raleigh scattering, Compton scattering, photoelectric effects, and pair production.

2.3.2 Neutron Transport

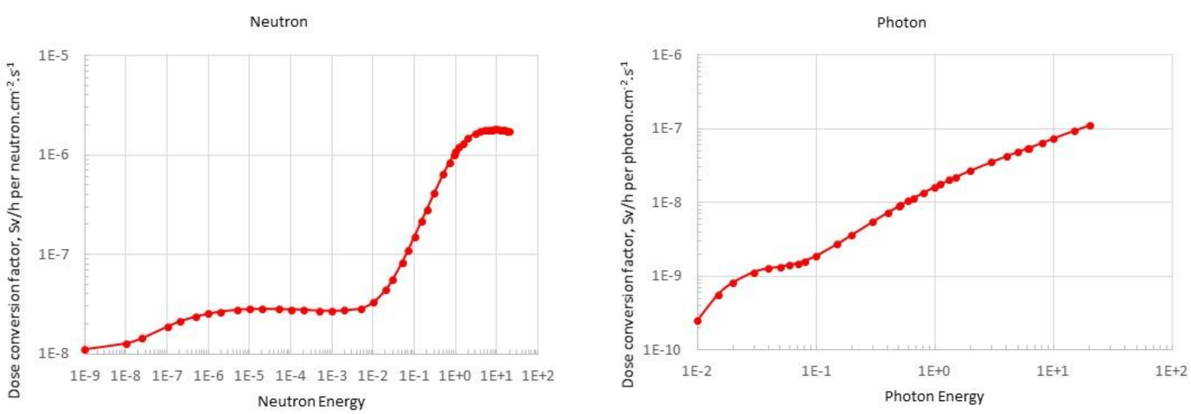
Neutron transport calculations were performed using both neutron (n) and photon (p) transport modes (mode n p in MCNP). The neutron transport model considers neutron capture, elastic and inelastic scattering, and some molecular scattering and temperature effects on scattering.

By invoking both neutron (n) and photon (p) transport modes (mode n p), neutron-induced photons are accounted for in all radionuclides for which photon-production cross-sections data exist. In addition, bremsstrahlung photons are also generated with a thick-target bremsstrahlung approximation.

2.3.3 Outputs

For the current analysis, the outputs of interest are the effective dose rates at various distances from a used fuel bundle or a UFC. The dose rates were calculated by multiplying photon and neutron fluxes obtained using the MCNP track length estimate of particle flux averaged over a mesh cell (F4 tally) and the dose conversion factors from Table A.1 (photon) and Table A.5 (neutron) of the ICRP Publication 116 (ICRP 2010). Dose conversion values associated with the Antero-Posterior (AP) geometry were selected since applying the dose conversion values for the AP geometry could result in the highest dose rates. In AP geometry, the ionising radiation is incident on the body in a direction orthogonal to its long axis. The photon and neutron dose conversion factors used in the effective dose rate calculations are plotted in Figure 6.

For completeness, the corresponding absorbed dose rate in air values are also provided in Appendix B.2. The absorbed dose rate values were calculated using the energy deposition tally in MCNP.



(a) Neutron (b) Gamma

Figure 6: Effective Dose Conversion Factors

2.4 Modeling Assumption

Parameter	Modeling Assumption	Basis and impact of assumption
Radiation source term spatial distribution within each fuel element	Uniform distribution within a fuel element in the reference case calculations.	Basis: The existing source term document (Heckman and Edward 2020) does not have data for spatial distribution within fuel element. Impact: The impact of skin effect on the gamma and neutron dose rates presented in this document is small (less than one percent).

2.5 Cases Analysed

Two geometry configurations were analysed:

a. Single used fuel bundle (Section 3) and
b. Single UFC (Section 4).

In each case, effective dose rates at various distances from the geometry containing used fuel bundle(s) were tabulated. The surrounding material is dry air. Contributions from neutron and gamma emissions, including the gammas emitted following neutron-capture reactions in fuel and surrounding materials, are considered. Two burnup values are considered: 220 and 290 MWh/kgU.

Decay times from discharge up to 10 million years are considered. It should be noted that the current Engineered Barrier System design is based on the heat generation of used 37-element CANDU fuel bundles with 10 years of decay time. As such, the calculated dose rates associated with this calculation case are very high but not realistic (i.e. will never be encountered in the actual APM activities). Nevertheless, the zero-decay time case was included for information.

3. DOSE RATES FROM A USED CANDU FUEL BUNDLE

In this scenario, unshielded dose rates from gamma, neutron, and neutron-capture gamma emitted by the used fuel bundle were tabulated at various distances from the used fuel bundle. The dose rate profiles are illustrated in Figure 7 (gamma) and Figure 8 (neutron).

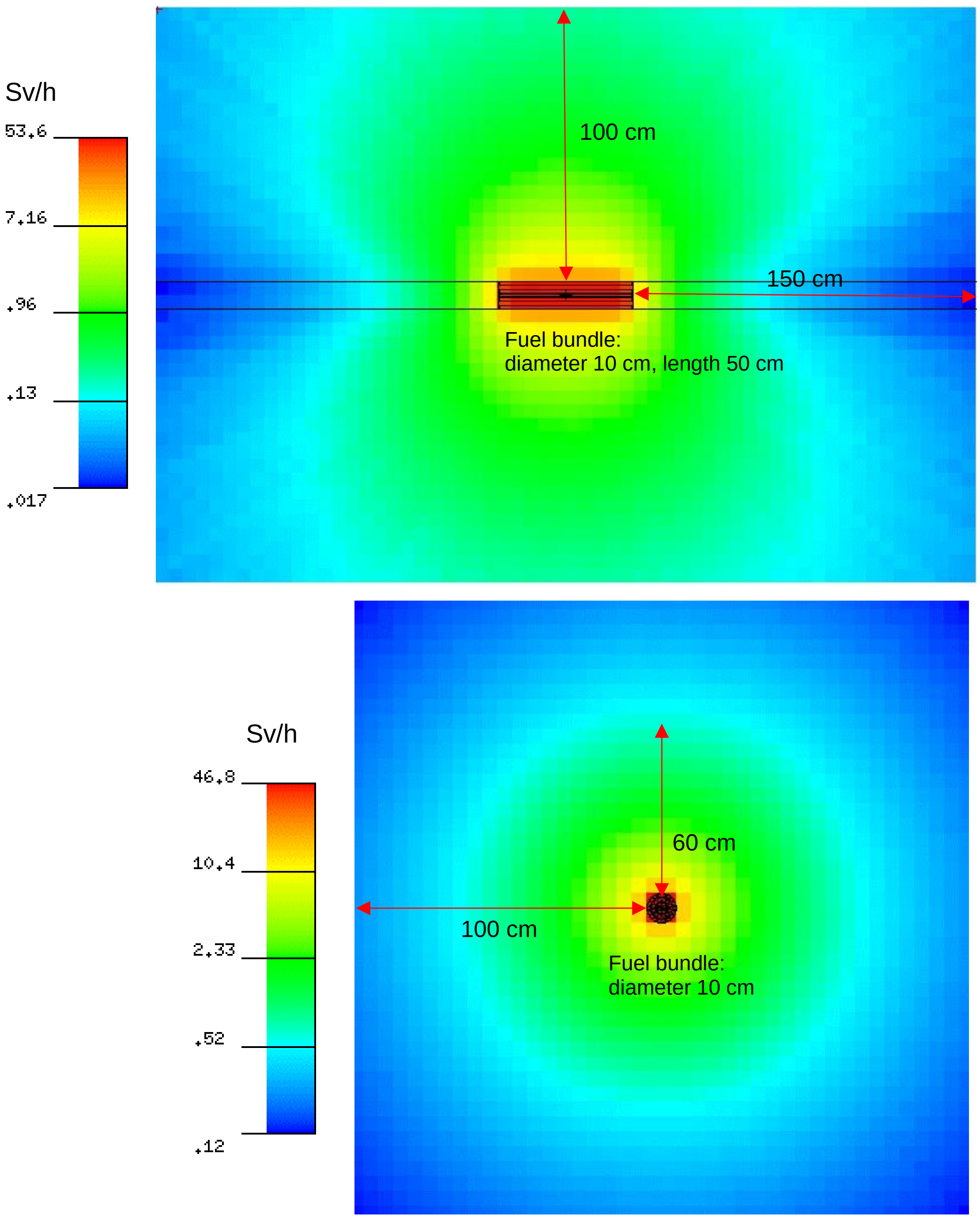


Figure 7: Dose Rate Profile from a Used Fuel Bundle: Gamma

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

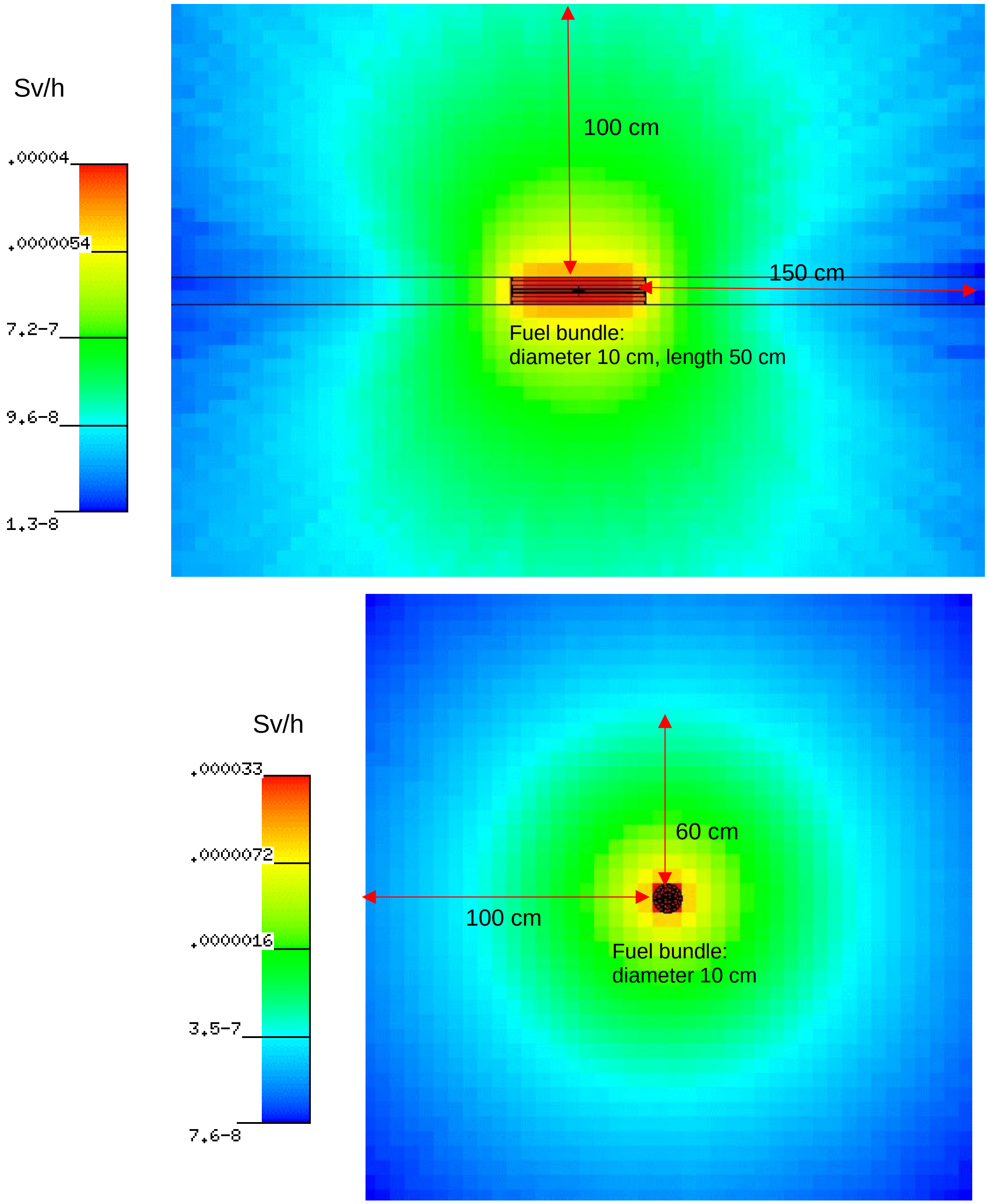


Figure 8: Dose Rate Profile from a Used Fuel Bundle: Neutron

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

Dose rates as a function of distance from a used fuel bundle are plotted in Figure 9. Contributions from neutron and neutron-capture gammas are small compared to that of primary gammas. Dose rates as a function of distance and bundle burnup and decay times are tabulated in Table 5.

The dose rates at 1 m away from a used CANDU fuel bundle are plotted in Figure 10 as a function of decay times. The dose rate decreases significantly during the first ten years after discharge due to decay of shorter-lived fission products. The dose rate decreases by a factor of approximately ten between 10 and 100 years after discharge. The dose rate profile versus time between ten and few hundred years after discharge follows the decay profile of Cs-137. The dose rate profile between 200 and a few thousand years follows the decay profile of Am-241. Beyond few thousand years decay, the dose rate profile is approximately flat since gamma emitters during that period are from long-lived actinides.

There is a slight increase in dose rates between 1E+04 and 1E+05 years because of the higher number of gammas emitted with energy above 0.3 MeV at 1E+05 years compared to that at 1E+04 years and higher energy gammas are more likely to escape the fuel elements. Although the gamma source intensity decreases during between 1E+04 and 1E+05 years (see Figure 3), the gamma spectrum becomes harder (i.e., higher average energy gammas are emitted from the fuel, see Table 2 and Figure 11). The primary gamma emitters are from the uranium (4n+2) decay chain, such as Bi-214.

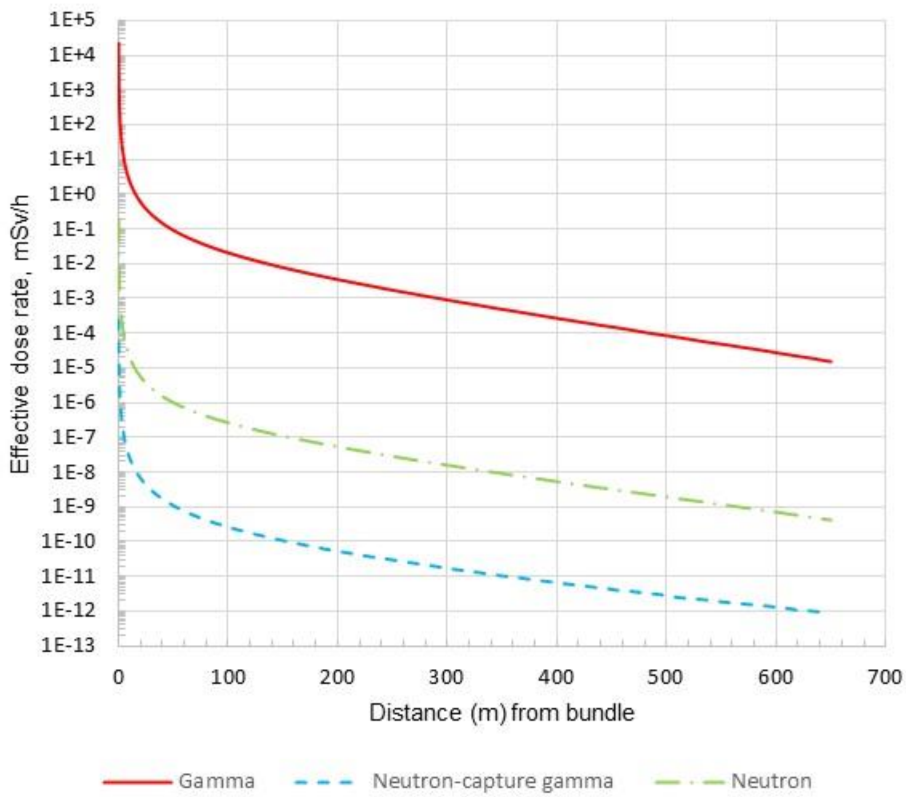


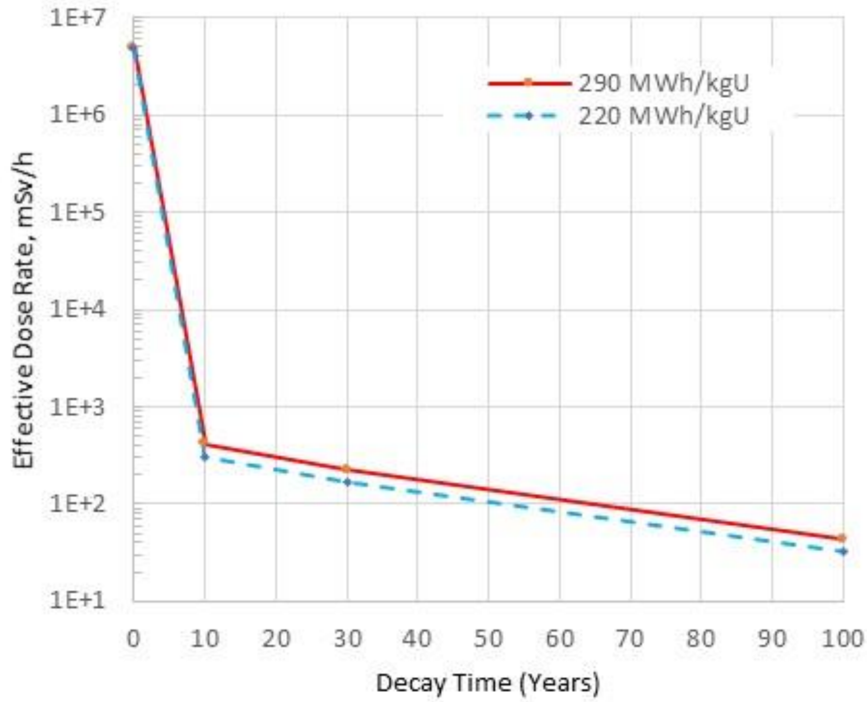
Figure 9: Dose Rates from an Unshielded Used Fuel Bundle

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

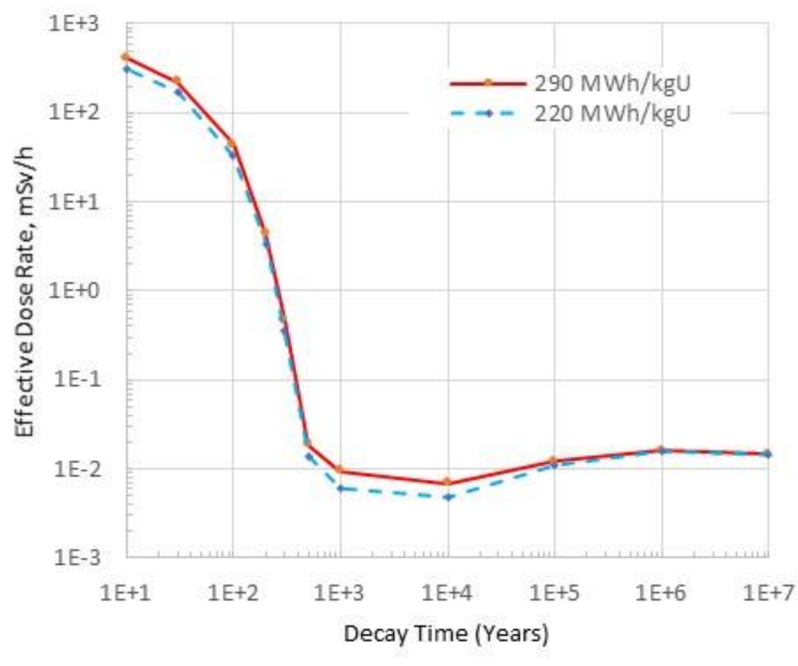
Table 5: Effective Dose Rates from a Used CANDU Fuel Bundle

Bundle Burnup	Decay Time (Years)	Effective Dose Rate (mSv/h) at Distance from Fuel Bundle Envelope							
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m	650 m
220 MWh/kgU	0	5.18E+08	3.61E+07	4.96E+06	1.34E+06	5.63E+04	2.29E+03	5.35E+02	9.12E-01
	10	3.15E+04	2.20E+03	3.04E+02	8.15E+01	3.38E+00	1.30E-01	2.91E-02	2.24E-05
	30	1.75E+04	1.23E+03	1.69E+02	4.55E+01	1.89E+00	7.26E-02	1.63E-02	1.20E-05
	100	3.44E+03	2.41E+02	3.33E+01	8.93E+00	3.71E-01	1.43E-02	3.19E-03	2.33E-06
	200	3.46E+02	2.43E+01	3.36E+00	9.00E-01	3.74E-02	1.44E-03	3.22E-04	2.45E-07
	300	3.67E+01	2.57E+00	3.56E-01	9.54E-02	3.97E-03	1.53E-04	3.45E-05	2.81E-08
	500	1.37E+00	9.85E-02	1.38E-02	3.71E-03	1.57E-04	6.45E-06	1.40E-06	1.32E-09
	1E+03	6.17E-01	4.45E-02	6.22E-03	1.68E-03	7.15E-05	2.97E-06	6.53E-07	3.53E-10
	1E+04	5.09E-01	3.58E-02	4.94E-03	1.33E-03	5.53E-05	2.17E-06	4.90E-07	4.73E-10
	1E+05	1.12E+00	7.89E-02	1.09E-02	2.92E-03	1.21E-04	4.62E-06	1.04E-06	1.80E-09
	1E+06	1.62E+00	1.14E-01	1.58E-02	4.24E-03	1.76E-04	6.65E-06	1.49E-06	2.93E-09
	1E+07	1.50E+00	1.06E-01	1.46E-02	3.93E-03	1.63E-04	6.16E-06	1.38E-06	2.75E-09
290 MWh/kgU	0	5.07E+08	3.53E+07	4.86E+06	1.31E+06	5.51E+04	2.24E+03	5.23E+02	8.88E-01
	10	4.25E+04	2.97E+03	4.11E+02	1.10E+02	4.57E+00	1.76E-01	3.94E-02	3.11E-05
	30	2.30E+04	1.61E+03	2.23E+02	5.98E+01	2.48E+00	9.54E-02	2.14E-02	1.57E-05
	100	4.50E+03	3.16E+02	4.36E+01	1.17E+01	4.86E-01	1.87E-02	4.19E-03	3.04E-06
	200	4.53E+02	3.18E+01	4.39E+00	1.18E+00	4.89E-02	1.88E-03	4.22E-04	3.15E-07
	300	4.79E+01	3.36E+00	4.65E-01	1.25E-01	5.19E-03	1.99E-04	4.50E-05	3.81E-08
	500	1.89E+00	1.36E-01	1.90E-02	5.12E-03	2.17E-04	8.91E-06	1.95E-06	1.64E-09
	1E+03	9.38E-01	6.76E-02	9.45E-03	2.55E-03	1.09E-04	4.52E-06	9.99E-07	4.87E-10
	1E+04	7.01E-01	4.93E-02	6.82E-03	1.83E-03	7.65E-05	3.02E-06	6.81E-07	6.05E-10
	1E+05	1.25E+00	8.81E-02	1.22E-02	3.27E-03	1.35E-04	5.17E-06	1.16E-06	1.87E-09
	1E+06	1.66E+00	1.17E-01	1.61E-02	4.33E-03	1.80E-04	6.81E-06	1.53E-06	2.95E-09
	1E+07	1.50E+00	1.06E-01	1.46E-02	3.92E-03	1.62E-04	6.15E-06	1.38E-06	2.76E-09

Note: Dose rate values presented in the table include contributions from gammas, neutrons, and neutron-capture gammas.



(a) Discharge to 100 years



(b) 10 to 10 million years

Figure 10: Dose Rates at 1 m from an Unshielded Used Fuel Bundle

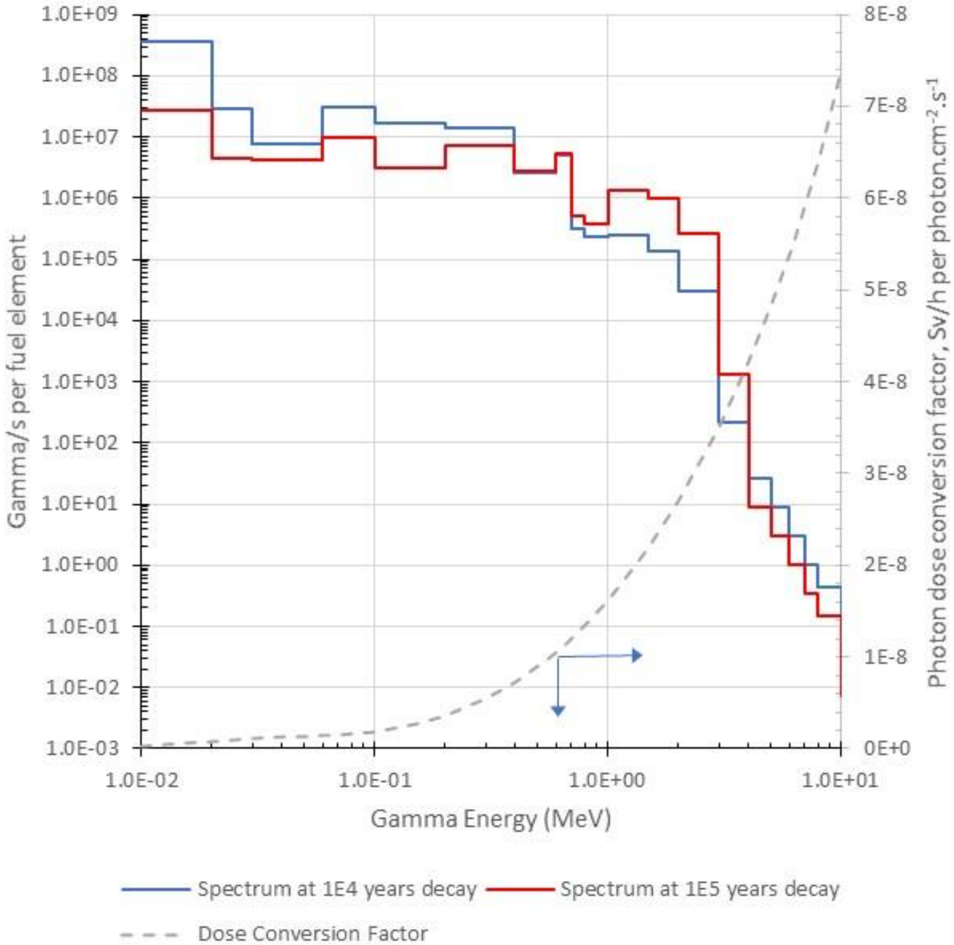


Figure 11: Gamma Source Spectra at 1E+04 and 1E+05 Years

Note: Values plotted above correspond to an outer ring fuel element from a bundle with 290 MWh/kgU burnup.

4. DOSE RATES FROM A USED FUEL CONTAINER

In this scenario, dose rates from a UFC containing 48 used CANDU fuel bundles of the same burnup and decay times were calculated at various distances from the UFC. The dose rate profiles are illustrated in Figure 12 (gamma) and Figure 13 (neutron).

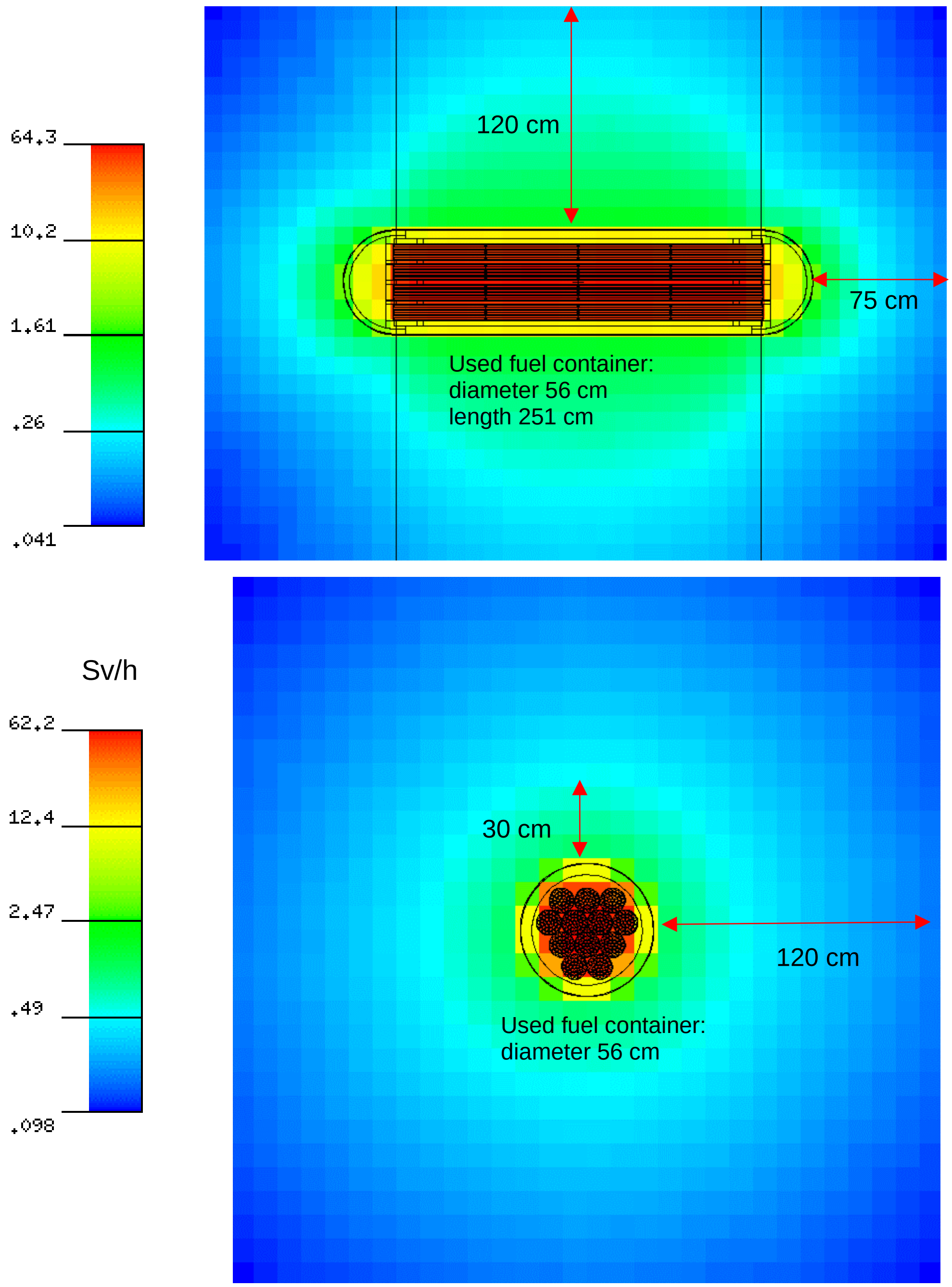


Figure 12: Dose Rate Profile from a Used Fuel Container: Gamma

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

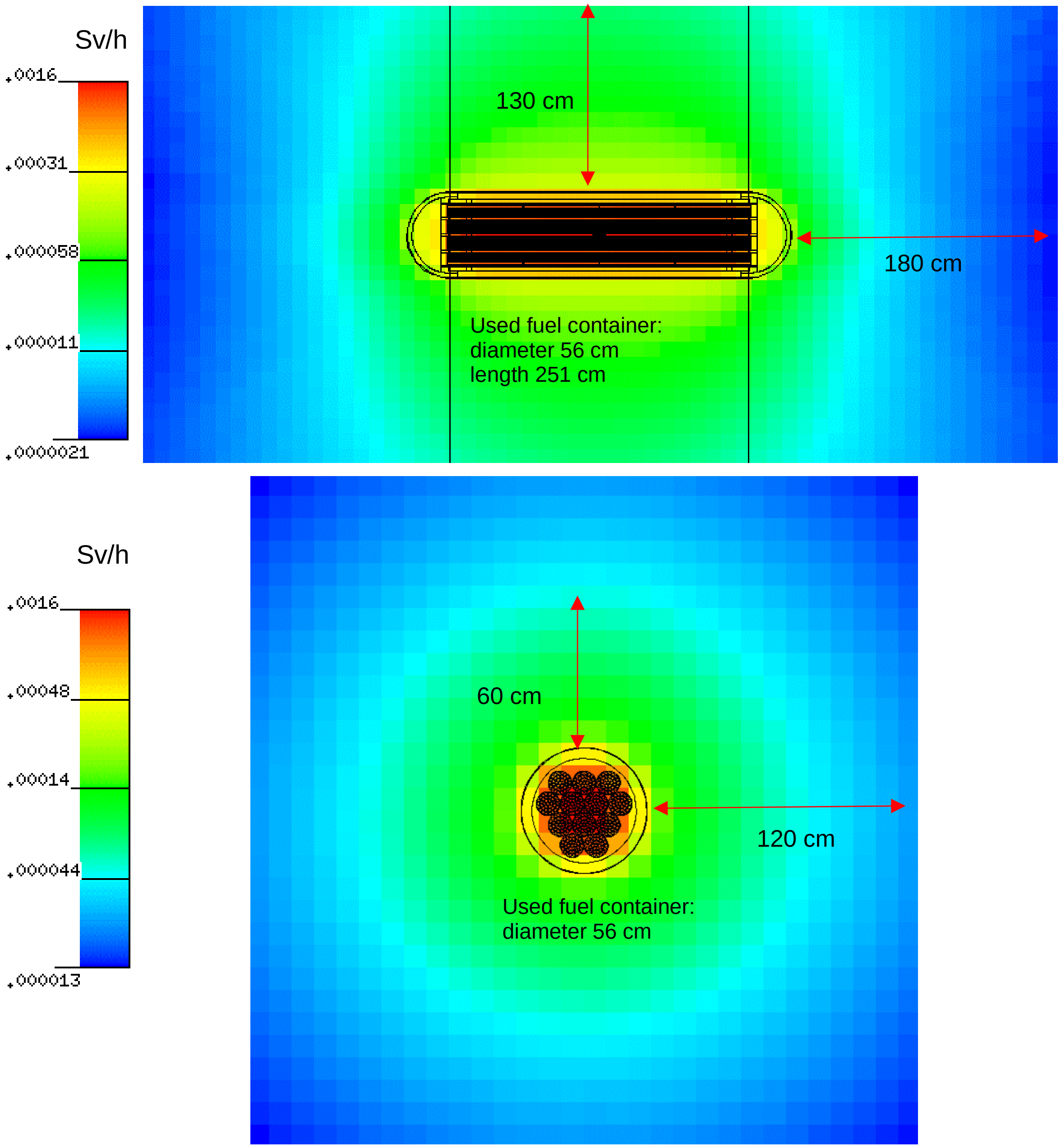


Figure 13: Dose Rate Profile from a Used Fuel Container: Neutron

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

Dose rates as a function of distance from a UFC are plotted in Figure 14. Since the UFC construct does not contain significant shielding material to attenuate the gamma radiation emitted from the fuel, the primary contributor to the dose rate outside of a UFC is that of gamma radiation. Contributions from neutrons and neutron-capture gammas are small. Dose rates as a function of distance, bundle burnup, and decay times are shown in Table 6. The dose rate reduction as a function of distance is less than that for a single fuel bundle discussed in Section 3 due to the geometry effect (larger size of the source region, i.e. the size of the UFC versus the used fuel bundle).

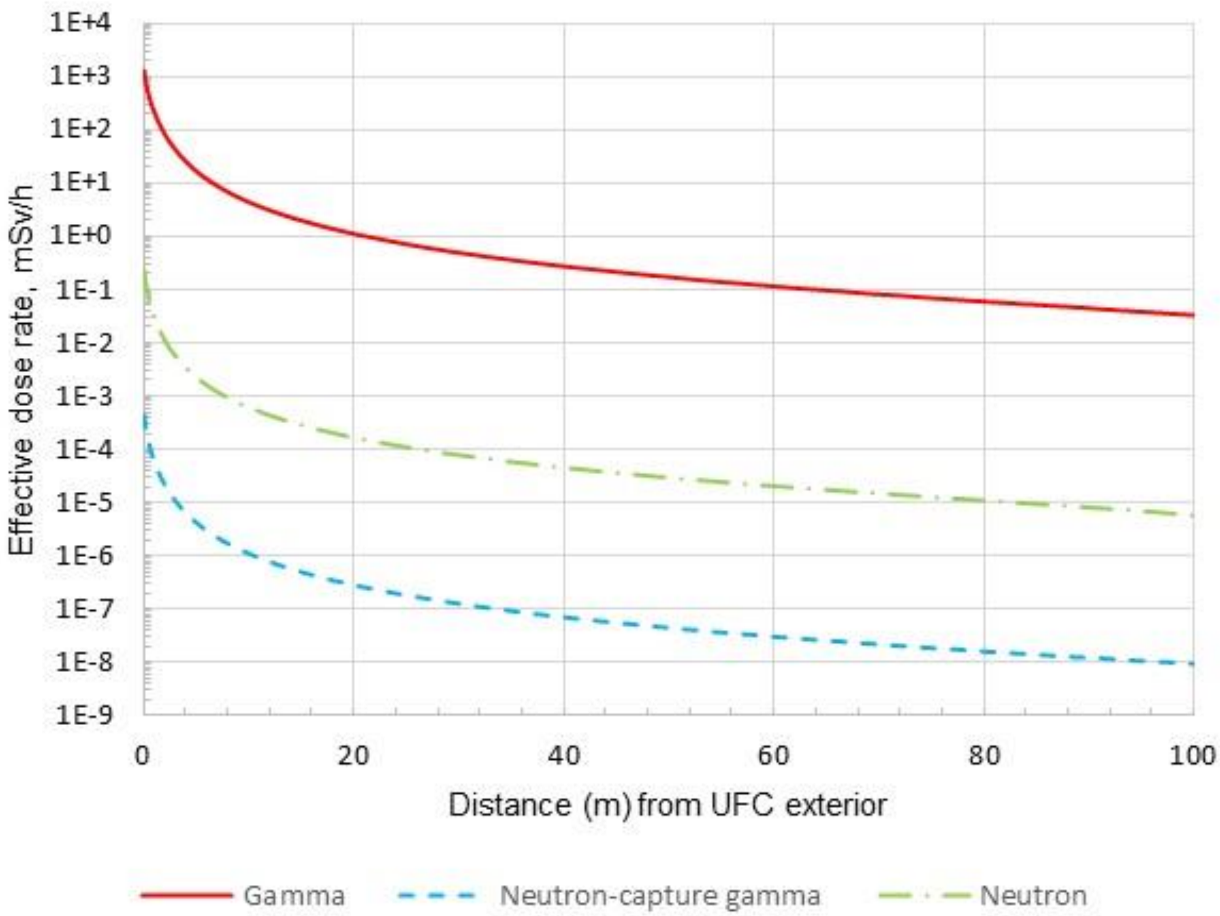


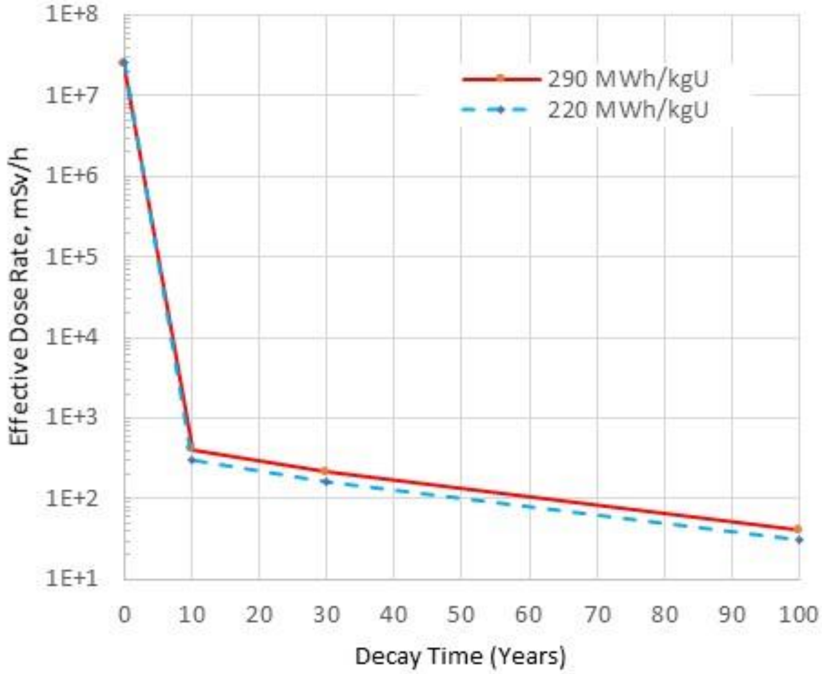
Figure 14: Dose Rate at Different Distances from a UFC

Values plotted above correspond to 290 MWh/kgU bundle with 30 years decay time.

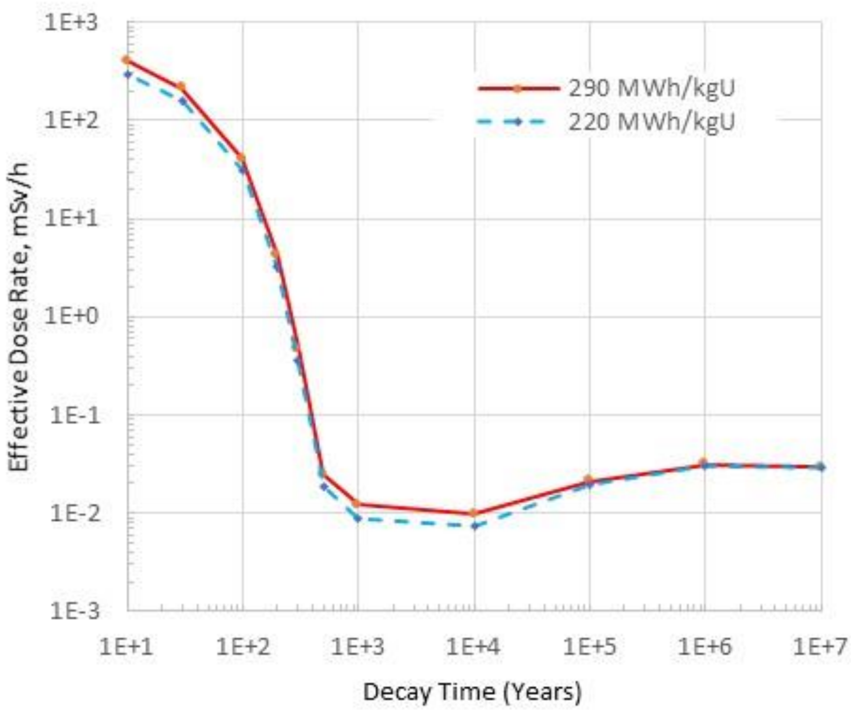
Table 6: Effective Dose Rates from a Used Fuel Container

Bundle Burnup	Decay Time (Years)	Effective dose rate (mSv/h) at distance from UFC exterior						
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m
220 MWh/kgU	0	1.90E+08	7.69E+07	2.58E+07	9.57E+06	5.30E+05	2.46E+04	5.05E+03
	10	1.85E+03	8.05E+02	2.97E+02	1.16E+02	6.32E+00	2.48E-01	5.09E-02
	30	9.97E+02	4.34E+02	1.60E+02	6.25E+01	3.42E+00	1.34E-01	2.72E-02
	100	1.94E+02	8.47E+01	3.12E+01	1.22E+01	6.68E-01	2.63E-02	5.33E-03
	200	2.00E+01	8.72E+00	3.21E+00	1.25E+00	6.87E-02	2.67E-03	5.51E-04
	300	2.27E+00	9.90E-01	3.63E-01	1.40E-01	7.75E-03	3.00E-04	6.25E-05
	500	1.28E-01	5.39E-02	1.91E-02	7.23E-03	3.99E-04	1.65E-05	3.45E-06
	1E+03	6.27E-02	2.56E-02	8.79E-03	3.31E-03	1.83E-04	8.05E-06	1.71E-06
	1E+04	5.02E-02	2.13E-02	7.44E-03	2.82E-03	1.55E-04	6.53E-06	1.37E-06
	1E+05	1.21E-01	5.30E-02	1.92E-02	7.39E-03	4.00E-04	1.56E-05	3.26E-06
	1E+06	1.94E-01	8.55E-02	3.10E-02	1.19E-02	6.39E-04	2.49E-05	5.35E-06
	1E+07	1.84E-01	8.12E-02	2.94E-02	1.13E-02	6.07E-04	2.35E-05	4.98E-06
290 MWh/kgU	0	1.84E+08	7.45E+07	2.49E+07	9.27E+06	5.13E+05	2.38E+04	4.89E+03
	10	2.52E+03	1.10E+03	4.04E+02	1.57E+02	8.60E+00	3.39E-01	6.89E-02
	30	1.31E+03	5.73E+02	2.11E+02	8.24E+01	4.50E+00	1.78E-01	3.60E-02
	100	2.55E+02	1.11E+02	4.09E+01	1.60E+01	8.74E-01	3.45E-02	6.98E-03
	200	2.61E+01	1.14E+01	4.18E+00	1.63E+00	8.95E-02	3.49E-03	7.18E-04
	300	2.93E+00	1.28E+00	4.69E-01	1.81E-01	9.96E-03	3.90E-04	8.02E-05
	500	1.65E-01	6.95E-02	2.45E-02	9.28E-03	5.08E-04	2.08E-05	4.60E-06
	1E+03	8.71E-02	3.56E-02	1.22E-02	4.62E-03	2.54E-04	1.12E-05	2.36E-06
	1E+04	6.59E-02	2.79E-02	9.69E-03	3.67E-03	2.03E-04	8.60E-06	1.82E-06
	1E+05	1.33E-01	5.82E-02	2.10E-02	8.08E-03	4.37E-04	1.70E-05	3.61E-06
	1E+06	1.97E-01	8.67E-02	3.14E-02	1.20E-02	6.48E-04	2.53E-05	5.40E-06
	1E+07	1.84E-01	8.11E-02	2.94E-02	1.13E-02	6.06E-04	2.35E-05	4.97E-06

Notes:
Dose rate values presented in the table include contributions from gammas, neutrons, and neutron-capture gammas.
The current UFC design is not intended to handle used fuel bundles with less than ten years decay time. Nevertheless, the zero-decay time case was included in the table for information.



(a) Discharge to 100 years



(b) 10 to 10 million years

Figure 15: Dose Rate at 1 m from a Used Fuel Container

5. RESULT UNCERTAINTY

Contributors to uncertainties in calculated dose rates presented in this report include the following:

- Uncertainty in radiation source terms (source intensity and energy spectra).
  The uncertainty in the radionuclide predictions for used CANDU fuel is typically chosen as 20% (Garisto et al. 2009), regardless of decay time and energy group discretization.
- Statistical uncertainty in the MCNP simulations.
  For the unshielded (fuel bundle) or the lightly shielded (UFC) configurations, the statistical uncertainties of the quoted dose rates are one percent or less for dose rates up to 100 m away from the bundle or UFC. See Appendix B.1 for details on the statistical uncertainties.
- Uncertainty in the MCNP cross-section data for neutron and gamma transport calculations.
  The pointwise cross-section data from the Evaluated Nuclear Data File (ENDF) are comprehensive and detailed. Extensive data testing and benchmarking of the ENDF/B-VII cross-section dataset have been performed (LANL 2013). The uncertainty in the cross-section data is not expected to be dominant in the calculated dose rates for the current analysis.
- Modeling simplifications, such as not modeling fuel bundle bearing pads and split spacers.
  The uncertainty introduced from such modeling simplification is judged to be minimal.
- Uncertainty in radiation source term spatial distribution.
  The burnup and decay time dependent ring-wise source term distributions have been adopted in the current analysis. The ring-wise distributions were based on the calculations using the TRITON and ORIGEN-S modules in SCALE code package (Heckman and Edward 2020). The uncertainty associated with the ring-wise distribution is expected to be small.

Based on information above, the suggested result uncertainties for dose rates presented in Table 5 and Table 6 are ±20% of the listed values.

It should be noted that the burnup values selected for the analysis are conservative with respect to the average value; therefore, the calculated dose rates are conservative if applied to an average used CANDU fuel bundle or an average UFC.

6. SUMMARY

Effective dose rates associated with various used fuel handling configurations have been calculated using the updated radionuclide inventory estimates of the NWMO reference used fuel bundle (Heckman and Edward 2020). Contributions from neutron and gamma radiation emitted from the used fuel were considered. External dose rates from alpha and beta emissions are negligible at the dose rate receptor locations of interest. Note, however, their contributions to the thermal assessment are important. In this analysis, the radiation source term distribution within the bundle was considered by applying ring-wise source term distributions instead of assuming a uniform flat distribution. The application of this type of source distribution improves the accuracy of the dose rate calculations.

Two geometry configurations were analysed: i) single fuel bundle and ii) fuel bundles inside a UFC. Dose rate receptor locations up to 100 m from the source have been considered. Two burnup values were examined: 220 and 290 MWh/kgU. Decay times from discharge up to 1E+07 years were considered. It should be noted that the current Engineered Barrier System design is based on the heat generation of used 37-element CANDU fuel bundles with 10 years of decay time and dose rates decrease significantly during the first ten years of decay time. Nevertheless, the zero-decay time case was included for information.

The effective dose rates at the minimum cooling time for storage in UFC (10 years) are plotted in Figure 16 and Figure 17.

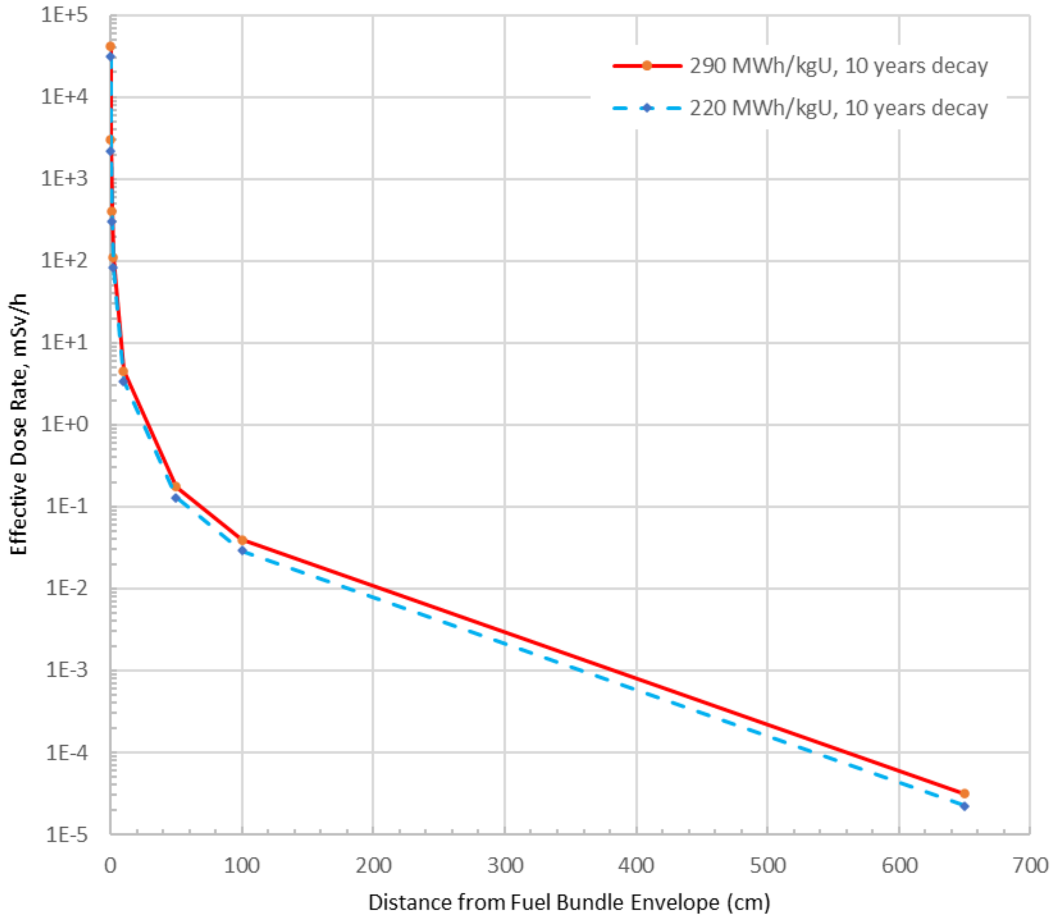


Figure 16: Effective Dose Rates from a Used CANDU Fuel Bundle

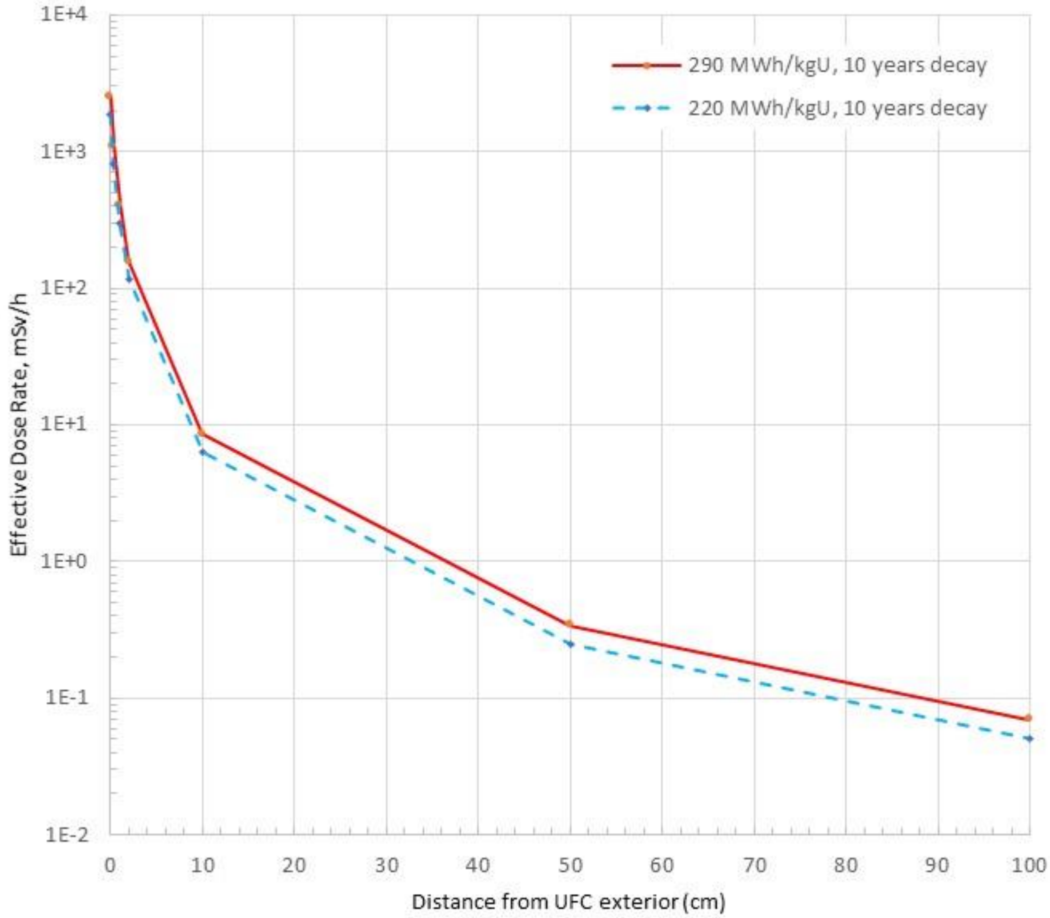


Figure 17: Effective Dose Rates from a Used Fuel Container

REFERENCES

Garisto, F. et al. 2009. Alpha, Beta and Gamma Dose Rates in Water in Contact with Used CANDU Fuel. Nuclear Waste Management Organization Technical Report NWMO-TR-2009-27. Toronto, Canada.

Heckman, K and Edward, J. 2020. Radionuclide Inventory for Reference CANDU Fuel Bundles. Nuclear Waste Management Organization Technical Report NWMO-TR-2020-05. Toronto, Canada.

ICRP. 2010. Conversion Coefficients for Radiological Protection Quantities for External Radiation Exposures. International Commission on Radiological Protection Publication 116. Ann ICRP 40(2-5).

LANL. 2013. Verification and Validation of the ENDF/B-VII.1-based Continuous Energy Data Tables for MCNP6. Los Alamos National Laboratory LA-UR-13-26742.

RSICC. 2013. MCNP6.1: Monte Carlo N-Particle Transport Code System. Available from the Radiation Safety Information Computational Center (https://rsicc.ornl.gov) as code package CCC-810.

APPENDIX A : DETAILS ON CALCULATION MODELS

CONTENTS

Page

A.1 GEOMETRY REPRESENTATION

A.1.1 Used Fuel Bundle

The reference used fuel bundle for the dose rate analysis is the standard length 37R fuel bundle (Heckman and Edward 2020). Input parameters and geometry pertaining to the reference used fuel bundle are listed in Table A-1 and Figure A-1.

The MCNP representation of the used fuel bundle is shown in Figure A-2.

Table A-1: Used Fuel Bundle Parameters

Parameter	Value
Number of fuel elements per bundle	37
UO_2 pellet diameter	12.2 mm
Sheath outer diameter Sheath thickness Sheath inner diameter	13.1 mm 0.4 mm 12.3 mm
Gap between UO_2 pellet and sheath	0.05 mm
Stack length	481 mm
U loading	19.25 kg
UO_2 loading	21.84 kg
UO_2 density	10.4967 g/cm^3
Bundle length	495 mm
End plate diameter End plate thickness	91 mm 1.6 mm
Material for sheath, end plates, end caps	Zircaloy-4

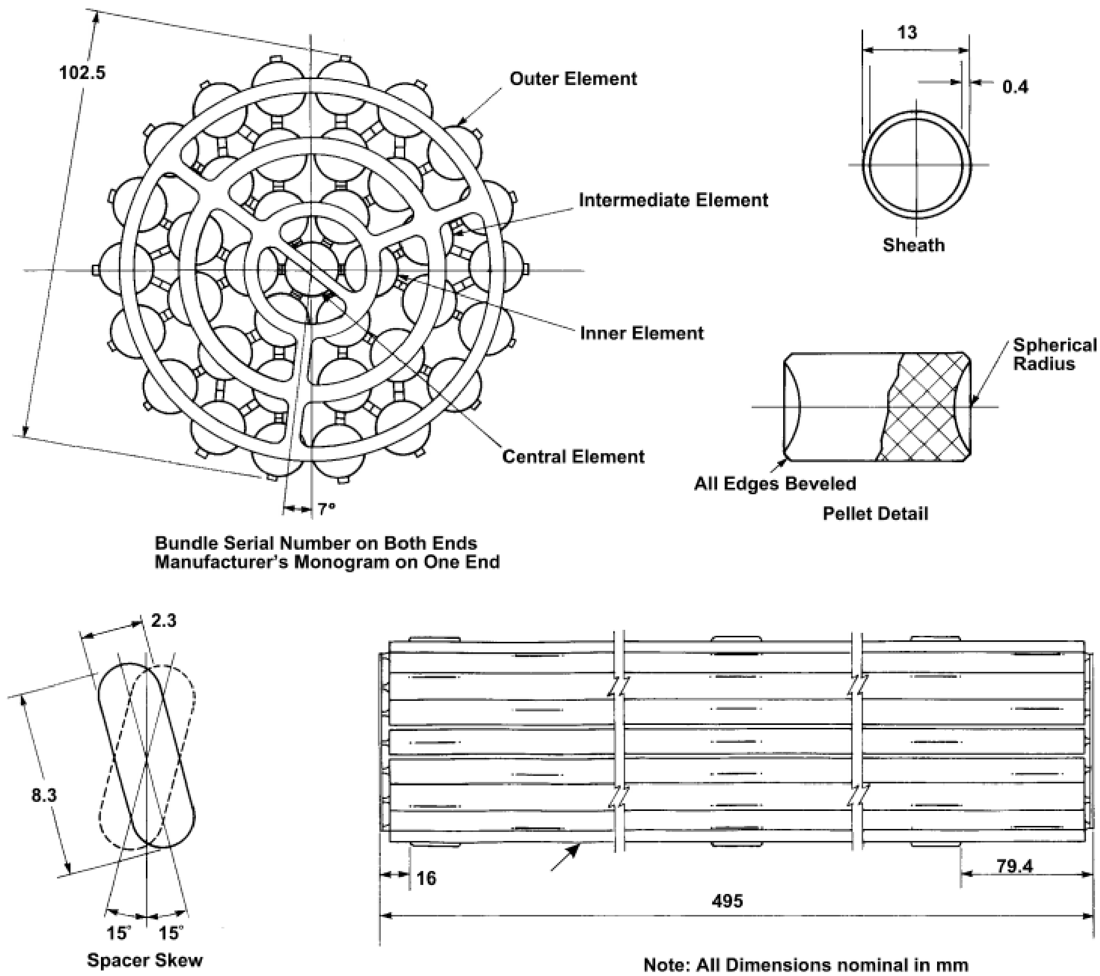


Figure A-1: Standard 37R Fuel Bundle Design

Source: Figure 2 of Heckman and Edwards 2020.

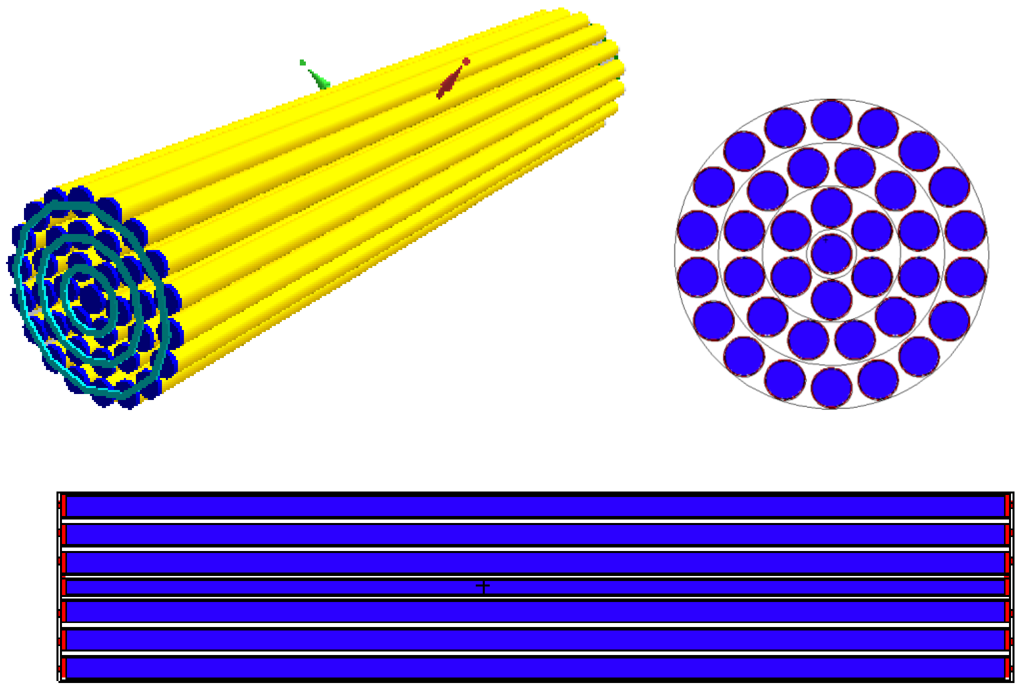


Figure A-2: MCNP Representation of Used Fuel Bundle

A.1.2 Used Fuel Container

The UFC consists of an inner vessel of carbon steel, which provides the structural strength to withstand repository loads, and an outer layer of copper which functions as a corrosion barrier. For the dose rate calculations, each UFC is filled with 48 reference used fuel bundles of the same burnup. The UFC geometry is shown in Figure A-3. The MCNP representation of the fuel bundles inside a UFC is shown in Figure A-4.

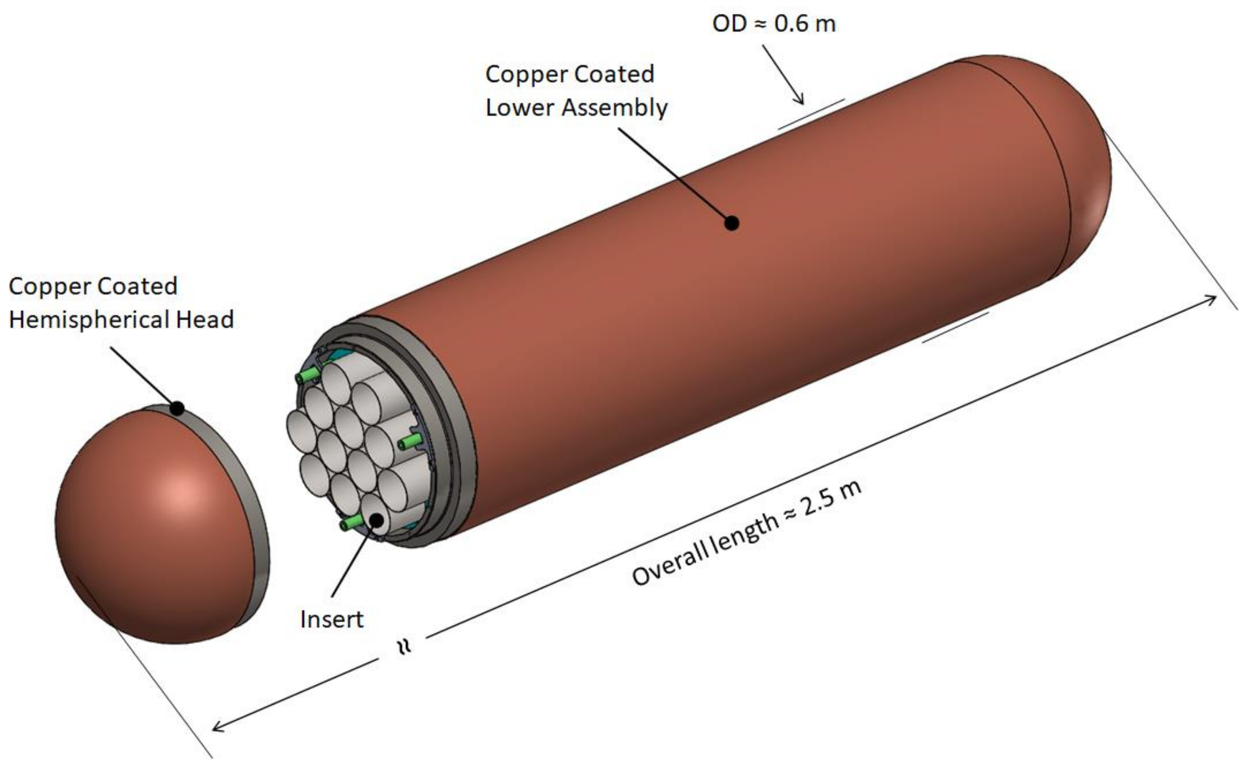


Figure A-3: Used Fuel Container

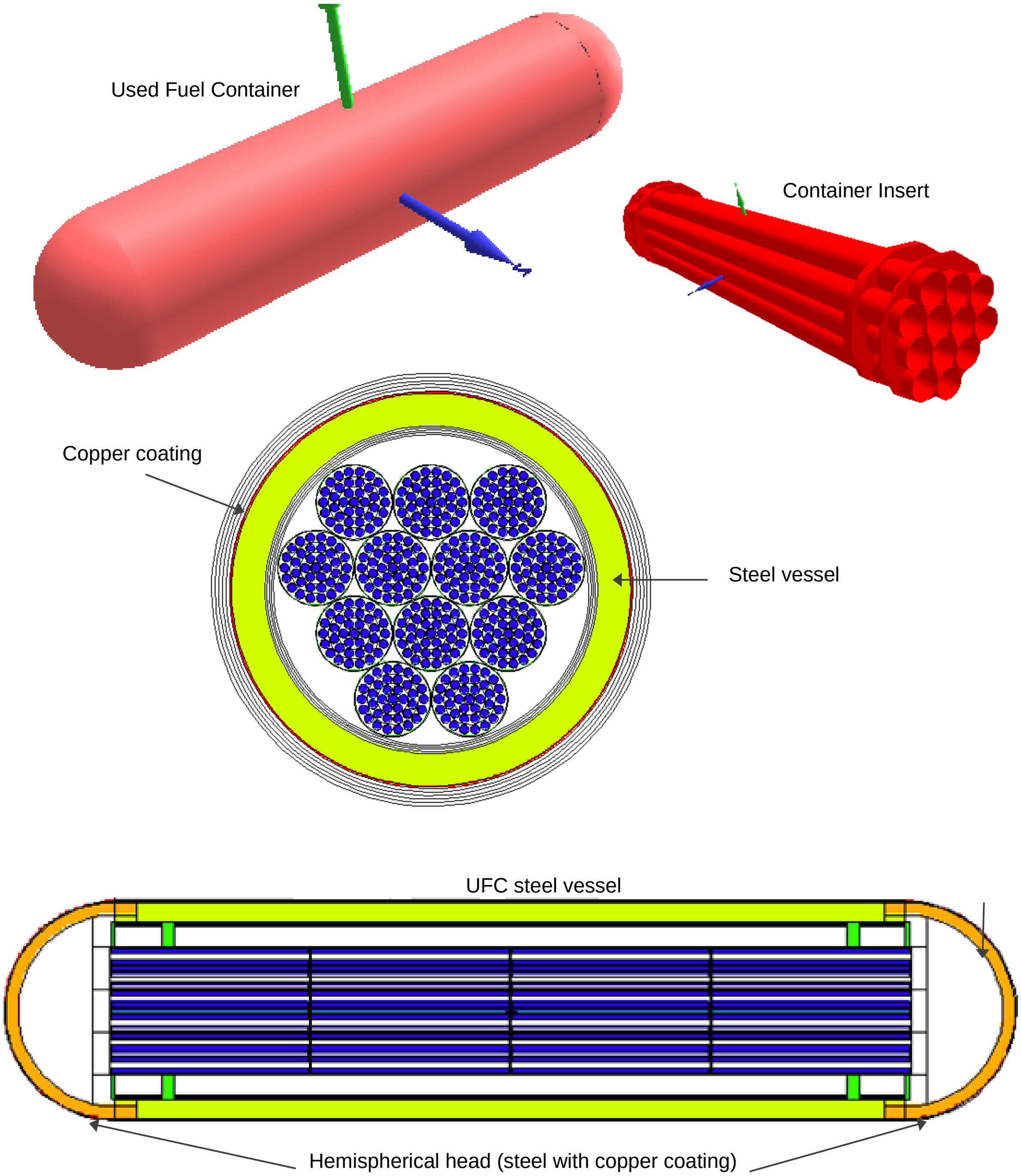


Figure A-4: MCNP Representation of Used Fuel Container and Container Insert

A.2 PARTICLE TRANSPORT CALCULATION OPTIONS

A.2.1 Gamma Transport

The following options (specified via PHYS:p particle physics option card) were employed for the photon particle transport:

- Generation of bremsstrahlung photons with thick-target bremsstrahlung model was turned on,
- Coherent (Thomson) scattering was turned on,
- Photonuclear particle production was turned off,
- Photon Doppler energy broadening was turned on, and
- No photofission prompt gammas.

A.2.2 Neutron Transport

The following options (specified via PHYS:n particle physics option card) were applied:

- Unresolved resonance range probability table treatment is turned on when data tables are available,
- Data tables are used up to their upper limit for each nuclide. Physics models are applied above that limit,
- Secondary photons are produced,
- All neutron interactions, including secondaries and inelastic collisions, are processed, and
- Treatment of nuclear elastic scattering by Prael/Liu/Striganov model.

In addition, the generation of neutrons from fissions in fuel is turned off by invoking the NONU card. Fissile material will be treated as purely a transport medium. Fission will be treated as capture with gammas produced.

A.3 REFERENCES

Heckman, K and Edward, J. 2020. Radionuclide Inventory for Reference CANDU Fuel Bundles. Nuclear Waste Management Organization Technical Report NWMO-TR-2020-05. Toronto, Canada.

APPENDIX B : ADDITIONAL DOSE RATE TABLES

B.1 EFFECTIVE DOSE RATES STATISTICAL UNCERTAINTIES

The MCNP statistical uncertainties of dose rate values given in Table 5 and Table 6 are listed in Table B-1 and Table B-2, respectively.

Table B-1: Statistical Uncertainty of Dose Rates in Table 5

Bundle Burnup	Decay Time (Years)	Distance from Fuel Bundle Envelope							
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m	650 m
220 MWh/kgU	0	0.01%	0.01%	0.02%	0.02%	0.05%	0.1%	0.1%	1.3%
	10	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.3%
	30	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.2%
	100	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.3%
	200	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.2%	2.0%
	300	0.04%	0.04%	0.06%	0.08%	0.18%	0.4%	0.5%	4.7%
	500	0.02%	0.02%	0.03%	0.05%	0.10%	0.2%	0.3%	3.9%
	1E+03	0.04%	0.05%	0.07%	0.09%	0.20%	0.4%	0.5%	4.2%
	1E+04	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.2%	2.4%
	1E+05	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	2.2%
	1E+06	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.3%	1.7%
	1E+07	0.01%	0.01%	0.02%	0.03%	0.07%	0.1%	0.2%	1.4%
290 MWh/kgU	0	0.01%	0.01%	0.02%	0.02%	0.05%	0.1%	0.1%	1.3%
	10	0.01%	0.01%	0.02%	0.02%	0.06%	0.1%	0.2%	1.3%
	30	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.2%
	100	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.3%
	200	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.2%	1.9%
	300	0.04%	0.04%	0.06%	0.08%	0.18%	0.4%	0.5%	5.1%
	500	0.02%	0.02%	0.03%	0.04%	0.10%	0.2%	0.3%	4.0%
	1E+03	0.04%	0.04%	0.06%	0.09%	0.18%	0.4%	0.5%	4.0%
	1E+04	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.2%	2.5%
	1E+05	0.02%	0.02%	0.03%	0.04%	0.10%	0.2%	0.3%	2.2%
	1E+06	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	1.8%
	1E+07	0.01%	0.01%	0.02%	0.03%	0.07%	0.2%	0.2%	1.4%

Table B-2: Statistical Uncertainty of Dose Rates in Table 6

Bundle Burnup	Decay Time (Years)	Effective dose rate at distance from UFC exterior						
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m
220 MWh/kgU	0	0.04%	0.04%	0.04%	0.05%	0.1%	0.2%	0.3%
	10	0.1%	0.1%	0.1%	0.1%	0.3%	0.5%	0.8%
	30	0.1%	0.1%	0.1%	0.1%	0.3%	0.5%	0.8%
	100	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	200	0.2%	0.2%	0.2%	0.2%	0.4%	0.8%	1.1%
	300	0.3%	0.3%	0.4%	0.4%	0.8%	1.8%	2.6%
	500	0.4%	0.4%	0.4%	0.5%	1.0%	2.0%	3.0%
	1E+03	0.2%	0.2%	0.3%	0.3%	0.6%	1.2%	1.7%
	1E+04	0.4%	0.4%	0.4%	0.5%	1.0%	2.0%	2.9%
	1E+05	0.1%	0.1%	0.1%	0.2%	0.3%	0.8%	1.1%
	1E+06	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.9%
	1E+07	0.1%	0.1%	0.1%	0.1%	0.2%	0.5%	0.7%
290 MWh/kgU	0	0.0%	0.0%	0.0%	0.1%	0.1%	0.2%	0.3%
	10	0.1%	0.1%	0.1%	0.1%	0.2%	0.5%	0.8%
	30	0.1%	0.1%	0.1%	0.1%	0.3%	0.5%	0.8%
	100	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	200	0.2%	0.2%	0.2%	0.2%	0.4%	0.8%	1.1%
	300	0.4%	0.3%	0.4%	0.4%	0.9%	1.8%	2.6%
	500	0.4%	0.4%	0.4%	0.5%	1.0%	2.0%	3.1%
	1E+03	0.2%	0.2%	0.3%	0.3%	0.6%	1.2%	1.7%
	1E+04	0.4%	0.4%	0.4%	0.5%	1.0%	1.9%	2.7%
	1E+05	0.1%	0.1%	0.1%	0.2%	0.4%	0.8%	1.1%
	1E+06	0.1%	0.1%	0.1%	0.2%	0.3%	0.7%	1.0%
	1E+07	0.1%	0.1%	0.1%	0.1%	0.2%	0.5%	0.7%

B.2 ABSORBED DOSE RATES IN AIR

The absorbed dose rates in air are presented in Table B-3 (from a used CANDU fuel bundle) along with the corresponding statistical uncertainties.

Table B-3: Absorbed Dose Rates from a Used CANDU Fuel Bundle

Bundle Burnup	Decay Time (Years)	Absorbed Dose Rate in Air (mGv/h) at Distance from Fuel Bundle Envelope							
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m	650 m
220 MWh/kgU	0	4.18E+08	2.90E+07	3.99E+06	1.07E+06	4.42E+04	1.65E+03	3.69E+02	8.23E-01
	10	3.08E+04	2.15E+03	2.97E+02	7.96E+01	3.29E+00	1.24E-01	2.71E-02	2.04E-05
	30	1.71E+04	1.20E+03	1.66E+02	4.44E+01	1.83E+00	6.89E-02	1.51E-02	1.08E-05
	100	3.36E+03	2.35E+02	3.25E+01	8.71E+00	3.60E-01	1.35E-02	2.97E-03	2.11E-06
	200	3.39E+02	2.37E+01	3.28E+00	8.78E-01	3.63E-02	1.36E-03	3.00E-04	2.22E-07
	300	3.58E+01	2.51E+00	3.47E-01	9.29E-02	3.84E-03	1.45E-04	3.21E-05	2.55E-08
	500	1.27E+00	9.06E-02	1.26E-02	3.38E-03	1.40E-04	5.60E-06	1.24E-06	1.22E-09
	1E+03	5.49E-01	3.93E-02	5.47E-03	1.47E-03	6.08E-05	2.48E-06	5.43E-07	2.52E-10
	1E+04	4.80E-01	3.37E-02	4.66E-03	1.25E-03	5.17E-05	1.97E-06	4.34E-07	4.12E-10
	1E+05	1.10E+00	7.78E-02	1.07E-02	2.88E-03	1.19E-04	4.46E-06	9.89E-07	1.72E-09
	1E+06	1.61E+00	1.14E-01	1.57E-02	4.21E-03	1.74E-04	6.49E-06	1.44E-06	2.83E-09
	1E+07	1.50E+00	1.06E-01	1.46E-02	3.91E-03	1.61E-04	6.03E-06	1.34E-06	2.65E-09
290 MWh/kgU	0	4.11E+08	2.85E+07	3.92E+06	1.05E+06	4.34E+04	1.63E+03	3.62E+02	8.00E-01
	10	4.16E+04	2.91E+03	4.01E+02	1.08E+02	4.44E+00	1.67E-01	3.67E-02	2.83E-05
	30	2.25E+04	1.58E+03	2.18E+02	5.84E+01	2.41E+00	9.06E-02	1.99E-02	1.42E-05
	100	4.40E+03	3.09E+02	4.26E+01	1.14E+01	4.72E-01	1.77E-02	3.89E-03	2.75E-06
	200	4.43E+02	3.11E+01	4.29E+00	1.15E+00	4.75E-02	1.79E-03	3.93E-04	2.85E-07
	300	4.67E+01	3.28E+00	4.53E-01	1.21E-01	5.02E-03	1.89E-04	4.18E-05	3.48E-08
	500	1.74E+00	1.24E-01	1.73E-02	4.63E-03	1.92E-04	7.73E-06	1.71E-06	1.44E-09
	1E+03	8.35E-01	5.98E-02	8.32E-03	2.23E-03	9.26E-05	3.77E-06	8.31E-07	3.47E-10
	1E+04	6.56E-01	4.62E-02	6.38E-03	1.71E-03	7.09E-05	2.71E-06	5.98E-07	5.21E-10
	1E+05	1.23E+00	8.66E-02	1.20E-02	3.21E-03	1.33E-04	4.97E-06	1.10E-06	1.79E-09
	1E+06	1.65E+00	1.16E-01	1.60E-02	4.29E-03	1.77E-04	6.63E-06	1.47E-06	2.84E-09
	1E+07	1.49E+00	1.05E-01	1.45E-02	3.90E-03	1.61E-04	6.02E-06	1.33E-06	2.66E-09

Table B-4: Statistical Uncertainty of Dose Rates in Table B-3

Bundle Burnup	Decay Time (Years)	Distance from Fuel Bundle Envelope							
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m	650 m
220 MWh/kgU	0	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	10	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	30	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.3%
	100	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	200	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.2%	2.2%
	300	0.04%	0.04%	0.06%	0.08%	0.18%	0.4%	0.5%	5.1%
	500	0.02%	0.03%	0.04%	0.05%	0.11%	0.2%	0.3%	4.4%
	1E+03	0.05%	0.05%	0.07%	0.10%	0.22%	0.5%	0.6%	5.6%
	1E+04	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	2.7%
	1E+05	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	2.3%
	1E+06	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.3%	1.8%
	1E+07	0.01%	0.01%	0.02%	0.03%	0.07%	0.2%	0.2%	1.4%
290 MWh/kgU	0	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	10	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	30	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.3%
	100	0.01%	0.01%	0.02%	0.03%	0.06%	0.1%	0.2%	1.4%
	200	0.02%	0.02%	0.03%	0.04%	0.08%	0.2%	0.2%	2.1%
	300	0.04%	0.04%	0.06%	0.08%	0.18%	0.4%	0.5%	5.6%
	500	0.02%	0.02%	0.04%	0.05%	0.11%	0.2%	0.3%	4.6%
	1E+03	0.04%	0.05%	0.07%	0.10%	0.21%	0.4%	0.6%	5.4%
	1E+04	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	2.9%
	1E+05	0.02%	0.02%	0.03%	0.04%	0.10%	0.2%	0.3%	2.4%
	1E+06	0.02%	0.02%	0.03%	0.04%	0.09%	0.2%	0.3%	1.9%
	1E+07	0.01%	0.01%	0.02%	0.03%	0.07%	0.2%	0.2%	1.4%

Table B-5: Absorbed Dose Rates from a Used Fuel Container

Bundle Burnup	Decay Time (Years)	Effective dose rate (mSv/h) at distance from UFC exterior						
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m
220 MWh/kgU	0	5.16E+07	2.28E+07	8.23E+06	3.14E+06	1.69E+05	6.51E+03	1.37E+03
	10	1.76E+03	7.68E+02	2.84E+02	1.11E+02	6.02E+00	2.31E-01	4.69E-02
	30	9.46E+02	4.13E+02	1.53E+02	5.97E+01	3.25E+00	1.25E-01	2.50E-02
	100	1.84E+02	8.06E+01	2.98E+01	1.16E+01	6.34E-01	2.44E-02	4.88E-03
	200	1.90E+01	8.29E+00	3.06E+00	1.19E+00	6.52E-02	2.47E-03	5.05E-04
	300	2.12E+00	9.29E-01	3.42E-01	1.32E-01	7.28E-03	2.74E-04	5.65E-05
	500	7.88E-02	3.45E-02	1.27E-02	4.91E-03	2.69E-04	1.02E-05	2.08E-06
	1E+03	2.12E-02	9.23E-03	3.45E-03	1.36E-03	7.36E-05	2.79E-06	5.77E-07
	1E+04	3.15E-02	1.39E-02	5.03E-03	1.93E-03	1.05E-04	4.09E-06	8.43E-07
	1E+05	1.15E-01	5.07E-02	1.84E-02	7.10E-03	3.83E-04	1.46E-05	3.05E-06
	1E+06	1.90E-01	8.40E-02	3.05E-02	1.17E-02	6.28E-04	2.41E-05	5.16E-06
	1E+07	1.81E-01	8.01E-02	2.91E-02	1.12E-02	5.98E-04	2.28E-05	4.82E-06
290 MWh/kgU	0	5.04E+07	2.22E+07	8.03E+06	3.07E+06	1.65E+05	6.35E+03	1.34E+03
	10	2.39E+03	1.05E+03	3.87E+02	1.51E+02	8.20E+00	3.15E-01	6.34E-02
	30	1.25E+03	5.45E+02	2.02E+02	7.87E+01	4.28E+00	1.65E-01	3.30E-02
	100	2.41E+02	1.05E+02	3.90E+01	1.52E+01	8.30E-01	3.20E-02	6.40E-03
	200	2.47E+01	1.08E+01	3.99E+00	1.55E+00	8.50E-02	3.23E-03	6.58E-04
	300	2.73E+00	1.20E+00	4.41E-01	1.71E-01	9.33E-03	3.56E-04	7.21E-05
	500	9.76E-02	4.27E-02	1.57E-02	6.07E-03	3.29E-04	1.21E-05	2.73E-06
	1E+03	2.98E-02	1.29E-02	4.84E-03	1.90E-03	1.03E-04	3.93E-06	7.97E-07
	1E+04	3.92E-02	1.73E-02	6.25E-03	2.40E-03	1.32E-04	5.13E-06	1.07E-06
	1E+05	1.24E-01	5.45E-02	1.98E-02	7.63E-03	4.11E-04	1.56E-05	3.30E-06
	1E+06	1.92E-01	8.49E-02	3.08E-02	1.18E-02	6.34E-04	2.44E-05	5.18E-06
	1E+07	1.81E-01	7.99E-02	2.90E-02	1.11E-02	5.97E-04	2.29E-05	4.81E-06

Notes:
Dose rate values presented in the table include contributions from gammas, neutrons, and neutron-capture gammas.
The current UFC design is not intended to handle used fuel bundles with less than ten years decay time. Nevertheless, the zero-decay time case was included in the table for information.

Table B-6: Statistical Uncertainty of Dose Rates in Table B-5

Bundle Burnup	Decay Time (Years)	Effective dose rate (mSv/h) at distance from UFC exterior						
		Near Contact	0.3 m	1 m	2 m	10 m	50 m	100 m
220 MWh/kgU	0	0.09%	0.09%	0.09%	0.11%	0.21%	0.48%	0.71%
	10	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	30	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	100	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.9%
	200	0.2%	0.2%	0.2%	0.2%	0.4%	0.8%	1.2%
	300	0.4%	0.4%	0.4%	0.4%	0.9%	1.9%	2.8%
	500	0.6%	0.6%	0.6%	0.7%	1.5%	3.3%	5.0%
	1E+03	0.6%	0.6%	0.6%	0.8%	1.5%	3.3%	5.0%
	1E+04	0.6%	0.6%	0.6%	0.7%	1.4%	3.1%	4.6%
	1E+05	0.1%	0.1%	0.2%	0.2%	0.4%	0.8%	1.2%
	1E+06	0.12%	0.12%	0.12%	0.15%	0.29%	0.64%	0.94%
	1E+07	0.09%	0.09%	0.10%	0.12%	0.23%	0.51%	0.75%
290 MWh/kgU	0	0.09%	0.09%	0.09%	0.11%	0.22%	0.48%	0.72%
	10	0.1%	0.1%	0.1%	0.1%	0.3%	0.5%	0.8%
	30	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	100	0.1%	0.1%	0.1%	0.1%	0.3%	0.6%	0.8%
	200	0.2%	0.2%	0.2%	0.2%	0.4%	0.8%	1.2%
	300	0.4%	0.4%	0.4%	0.5%	0.9%	2.0%	2.8%
	500	0.6%	0.6%	0.6%	0.8%	1.5%	3.4%	5.1%
	1E+03	0.6%	0.6%	0.6%	0.7%	1.5%	3.2%	4.8%
	1E+04	0.6%	0.6%	0.6%	0.7%	1.4%	3.1%	4.6%
	1E+05	0.2%	0.2%	0.2%	0.2%	0.4%	0.8%	1.2%
	1E+06	0.1%	0.1%	0.1%	0.2%	0.3%	0.7%	1.0%
	1E+07	0.09%	0.09%	0.10%	0.12%	0.23%	0.52%	0.76%