Dose Rate Analysis to Support Radiolysis Assessment of Used CANDU Fuel

NWMO-TR-2022-02 R001 June 2023

Imelda Ariani

Candu Energy Inc., a member of the SNC-Lavalin group

NUCLEAR WASTE MANAGEMENT ORGANIZATION
SOCIÉTÉ DE GESTION DES DÉCHETS NUCLÉAIRES

Nuclear Waste Management Organization
22 St. Clair Avenue East, 4th Floor
Toronto, Ontario
M4T 2S3
Canada

Tel: 416-934-9814
Web: www.nwmo.ca

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Document History

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Authored by:	Imelda Ariani		
Verified by:	Kwok Tsang and Kevin Choi		
Approved by:	Robert Farkas		
Nuclear Waste Management Organization			
Reviewed by:	Chantal Medri, Mihaela Ion, and Mehran Behazin		
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Revision Summary		
Revision Number	**Date**	**Description of Changes/Improvements**
R000	2022-04	Initial issue
R001	2023-06	Added Appendix C (28-element fuel bundle). Added dose rates inside a UFC, with UFC filled with dry air (Section 8.3.1.1) or moist air (Section 8.3.1.2). Revised text in Sections 2.1 and 11 to reflect the additional calculations.

ABSTRACT

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Report No.: **NWMO-TR-2022-02 R001**
Author(s): Imelda Ariani
Company: Candu Energy Inc., a member of SNC-Lavalin group
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Abstract

Key to the geological repository design concept is the use of multiple natural and engineered barriers to isolate the nuclear waste from the environment. The used fuel container 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. In the event of the failure of a fuel container and breaching of Zircaloy cladding, groundwater could enter the container and contact the fuel. The solubility of uranium dioxide is low under the reducing conditions expected in a deep geological repository, and thus the used fuel dissolves very slowly. However, in a failed container the radiolysis of groundwater and H_2 generated by steel corrosion are expected to impact the rate of dissolution of fuel. Consequently, estimates of alpha, beta, gamma, and neutron dose rates in the water near the fuel container and fuel surfaces are required to support the assessment of the potential effects of the radiolysis of water on the integrity of the container and the used fuel.

This report provides an update to previous dose rate estimates using current used fuel inventories, container designs, and state-of-the-art computational method. Dose rate calculations include contributions from alpha, beta, gamma, and neutron sources in the used fuel. Dose rates are provided at ten decay times ranging from 30 to 10 million years and two burnup values: 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 from all Canadian CANDU reactors. Configurations involving presence of water being analysed include water submersion of used fuel bundle and used fuel container and water ingress into the used fuel element (cladding breach), the used fuel container (container breach), and into the placement room. The sensitivity of the calculated dose rates to the water density, host rock type (crystalline or sedimentary rock), and the within-fuel radiation source distribution has also been examined.

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LIST OF ABBREVIATIONS

APM	Adaptive Phase Management
CANDU	CANada Deuterium Uranium
DGR	Deep Geological Repository
ENDF	Evaluated Nuclear Data File
Gy	Gray. A dose of 1 Gray is equivalent to 1 Joule of energy deposited in a kilogram of a material.
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. In this plan, 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.

In the post closure safety assessment (NWMO 2018), groundwater is assumed to come into contact with the used fuel container. The radiolysis of groundwater could produce oxidizing species such as hydrogen peroxide and hydroxyl radicals, potentially affecting the container corrosion process. Furthermore, the safety assessment also considers a scenario where the UO_2 fuel could come into direct contact with the groundwater. The solubility of uranium dioxide is low under the reducing conditions expected in a deep geological repository and thus used fuel dissolves very slowly under such conditions. However, the radiolysis of groundwater, caused by the radiation emitted by used fuel, generates radiolytic products that can react with the fuel and increase the dissolution rate (Bradley and Shoesmith 2022). Thus, the fuel dissolution rate in a failed container is expected to be controlled by the rate of generation of radiolytic products in groundwater until the radiation fields have decayed to sufficiently low levels. Consequently, it is important to have an accurate estimate of the alpha, beta, gamma, and neutron dose rates in water near the UFC and fuel surfaces to support the assessment of the potential effects of the radiolysis of water on the integrity of the UFC and the used fuel.

1.2 Purpose and Scope of the Document

Dose rate estimates of used CANDU fuel and UFC configurations have been calculated in support of the APM program in the past (Garisto et al. 2009, Morco et al. 2017). However, those calculations were based on earlier used CANDU fuel radionuclide inventories and container designs. This report provides a revision to these calculations, using current used CANDU fuel inventories and container designs, and state-of-the art computational methods.

1.3 Limits and Applicability

Results documented in this report are based on specific CANDU fuel characteristics, used fuel container design, and storage configurations. The conclusions drawn from the study are subject to change if there are significant changes in the used fuel container design or storage configurations as the repository design evolves.

2. METHODOLOGY

2.1 General Methodology

The general methodology for estimating the alpha, beta, gamma, and neutron dose rates in water near the fuel surface involves estimating the radiation source emission from used fuel (Section 2.2) and simulating the particle transport from the source locations to the locations of interest (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. The study indicated that the differences in the source terms are small (approximately 5%). The reference bundle type was then selected to as the regular 37-element fuel bundle design because it is the most abundant fuel type in the current inventory of used CANDU fuel.

Unless otherwise noted, all calculations performed in this report apply the reference used fuel bundle characteristics[1]. Two burnup values are 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). Ten decay times ranging from 30 years to 10 million years are analysed.

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

Three different geometry configurations are analysed: i) single used fuel bundle, ii) single UFC, and iii) multiple UFCs in a post-closure placement room configuration. The cases analysed are discussed 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 radiological hazard purposes, the external dose rates from alpha and beta particles are typically ignored due to the particles’ short range. However, for the radiolysis assessment, the alpha and beta sources significantly impact the energy deposition in water in contact with fuel elements. The source terms from the alpha, beta, gamma, and neutrons were estimated using the ring-wise radionuclide inventory predicted by the ORIGEN-S code (Heckman and Edward 2020). The ring-wise designations are illustrated in Figure 1.

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[1] Calculations using the 28-element fuel bundle design characteristics were performed to assess whether the calculated dose rates using the reference used fuel bundle characteristics are bounding. Discussions are provided in Appendix C.

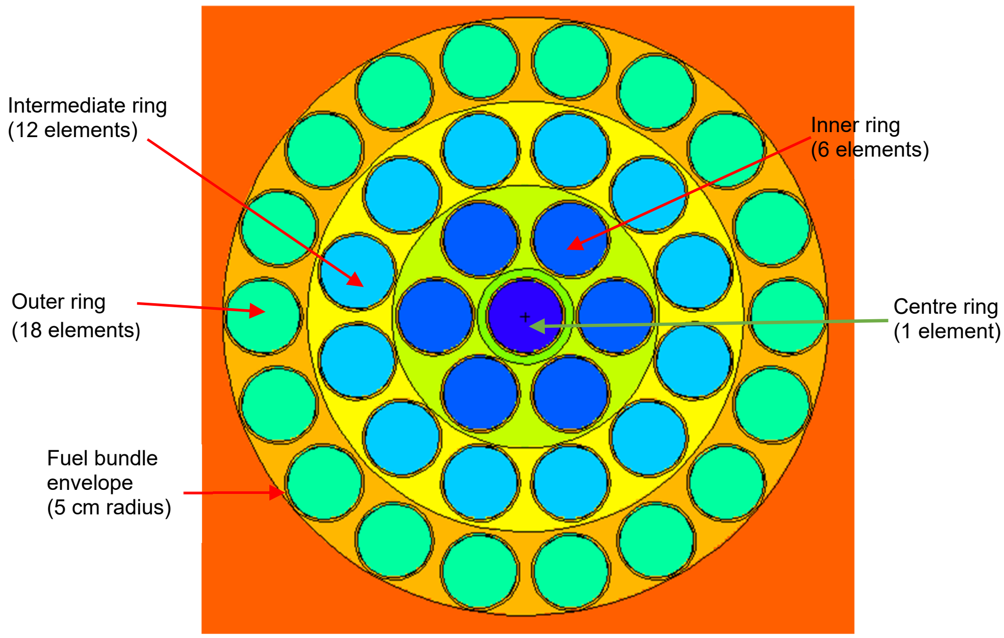


Figure 1: Ring-Wise Designation in Reference Used CANDU Fuel Bundle Model

2.2.1 Alpha Emissions

In the previous work (Garisto et al. 2009), the alpha emission from the used fuel was estimated using the bundle average radionuclide inventory and the radioactive decay data from Kocher 1981. Most of the alpha particles emitted from the radionuclides in the fuel elements were assumed to be in the range of 4 to 5 MeV. Before reaching the fuel surface, the alpha particles were assumed to be attenuated, resulting in a particle energy of 2.5 MeV. This was used to estimate the relative mass stopping power.

A more accurate approach for estimating the alpha emission, which does not depend on estimating stopping power, was applied in the current work. Alpha emission spectra were generated for each fuel ring (see Figure 1 for the fuel ring designation) using the radionuclide inventory estimated in Heckman and Edward 2020. Alpha source terms were calculated using the ORIGEN module from the SCALE code package, which uses an alpha decay library based on the ENDF/B-VII.1 nuclear data file. Alpha particles were modelled with discrete energies and the source spectrum was generated by straightforward binning into the desired energy group structure.

The alpha source intensity and average energy in each ring at different burnup and decay times are listed in Table 1 and Table 2, respectively. Figure 2 shows an illustration of the alpha emission energy spectra for outer ring fuel elements from a bundle with 290 MWh/kgU burnup. The alpha source intensity as a function of decay times is plotted in Figure 3. The alpha source intensity increases during the 30 to 100 years decay time due to the buildup of Am-241 (half-life = 432 years) from Pu-241 (half-life = 14 years) decay. Am-241 is the principal alpha source in the used fuel element during that decay period.

For the dose rate calculations, the alpha transport calculations were performed for sources emitted from the outer ring fuel element only since it emits more alphas compared to other fuel elements. Since the alpha particle range is short (less than 50 µm in water), the alpha particles emitted from a fuel element will not reach locations beyond its own fuel sheath. The dose rates obtained for the outer ring fuel element would bound the alpha dose rates from other fuel elements.

Table 1: Alpha Source Intensity

		Alpha/s source intensity per fuel element			
Bundle Burnup	**Decay Time (Years)**	**Centre**	**Inner**	**Intermediate**	**Outer**
220 MWh/kgU	30	1.88E+10	1.99E+10	2.26E+10	2.92E+10
	100	1.94E+10	2.06E+10	2.35E+10	3.04E+10
	200	1.72E+10	1.83E+10	2.07E+10	2.68E+10
	300	1.55E+10	1.64E+10	1.86E+10	2.39E+10
	500	1.30E+10	1.38E+10	1.55E+10	1.96E+10
	1E+03	9.51E+09	9.96E+09	1.10E+10	1.35E+10
	1E+04	3.82E+09	3.93E+09	4.16E+09	4.71E+09
	1E+05	2.28E+08	2.30E+08	2.36E+08	2.56E+08
	1E+06	7.18E+07	7.30E+07	7.60E+07	8.36E+07
	1E+07	5.47E+07	5.48E+07	5.50E+07	5.54E+07
290 MWh/kgU	30	2.67E+10	2.81E+10	3.14E+10	3.93E+10
	100	2.74E+10	2.88E+10	3.21E+10	3.99E+10
	200	2.40E+10	2.53E+10	2.81E+10	3.47E+10
	300	2.15E+10	2.25E+10	2.50E+10	3.08E+10
	500	1.78E+10	1.86E+10	2.05E+10	2.50E+10
	1E+03	1.26E+10	1.31E+10	1.42E+10	1.68E+10
	1E+04	4.59E+09	4.69E+09	4.91E+09	5.42E+09
	1E+05	2.49E+08	2.51E+08	2.57E+08	2.78E+08
	1E+06	8.03E+07	8.19E+07	8.56E+07	9.47E+07
	1E+07	5.53E+07	5.54E+07	5.55E+07	5.60E+07

Table 2: Alpha Source Average Energy

		Average MeV of Emitted Alpha Particles			
Bundle Burnup	**Decay Time (Years)**	**Centre**	**Inner**	**Intermediate**	**Outer**
220 MWh/kgU	30	5.38	5.38	5.39	5.40
	100	5.37	5.37	5.38	5.39
	200	5.35	5.35	5.35	5.37
	300	5.33	5.33	5.34	5.35
	500	5.30	5.30	5.31	5.32
	1E+03	5.24	5.24	5.25	5.26
	1E+04	5.15	5.15	5.15	5.15
	1E+05	5.20	5.20	5.20	5.19
	1E+06	5.48	5.48	5.49	5.52
	1E+07	5.35	5.35	5.35	5.34
290 MWh/kgU	30	5.39	5.40	5.40	5.41
	100	5.38	5.38	5.39	5.40
	200	5.36	5.36	5.37	5.38
	300	5.34	5.34	5.35	5.36
	500	5.31	5.32	5.32	5.33
	1E+03	5.25	5.26	5.26	5.27
	1E+04	5.15	5.15	5.15	5.15
	1E+05	5.20	5.20	5.19	5.18
	1E+06	5.51	5.51	5.52	5.54
	1E+07	5.35	5.34	5.34	5.34

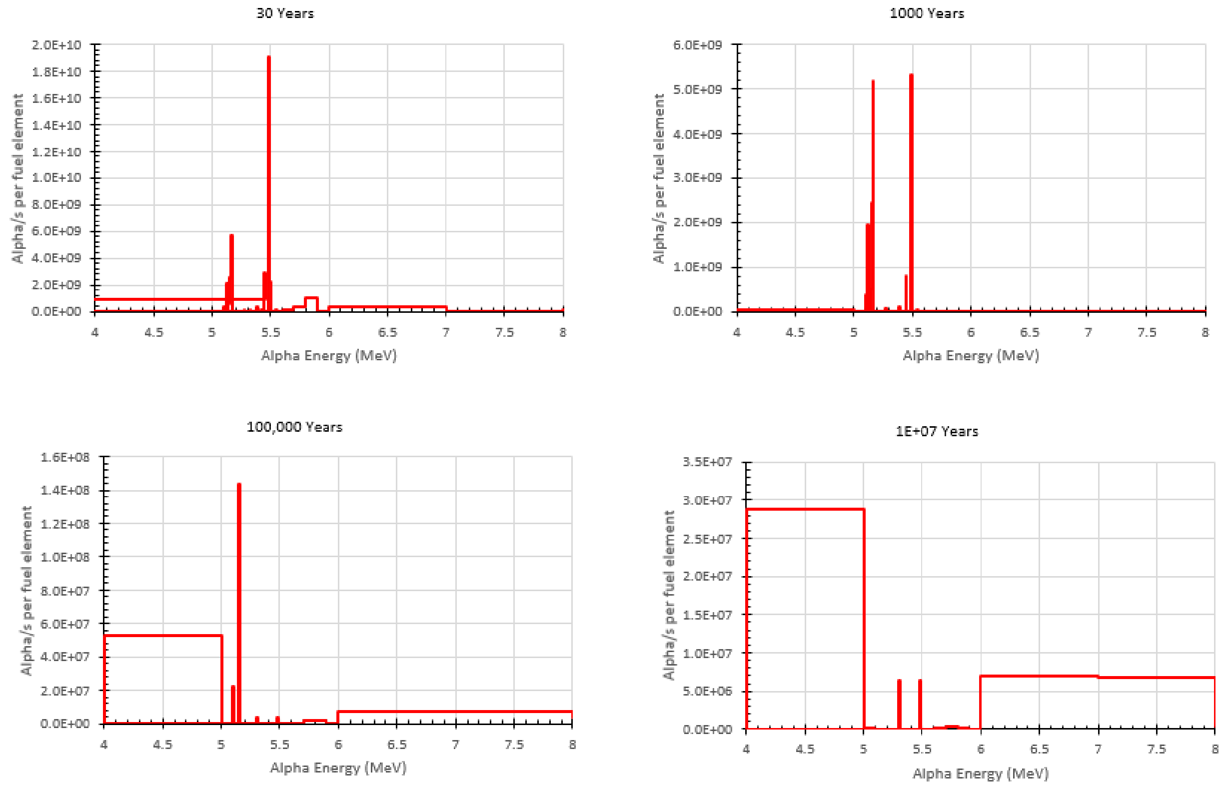


Figure 2: Alpha Emission Spectra at Various Decay Times

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

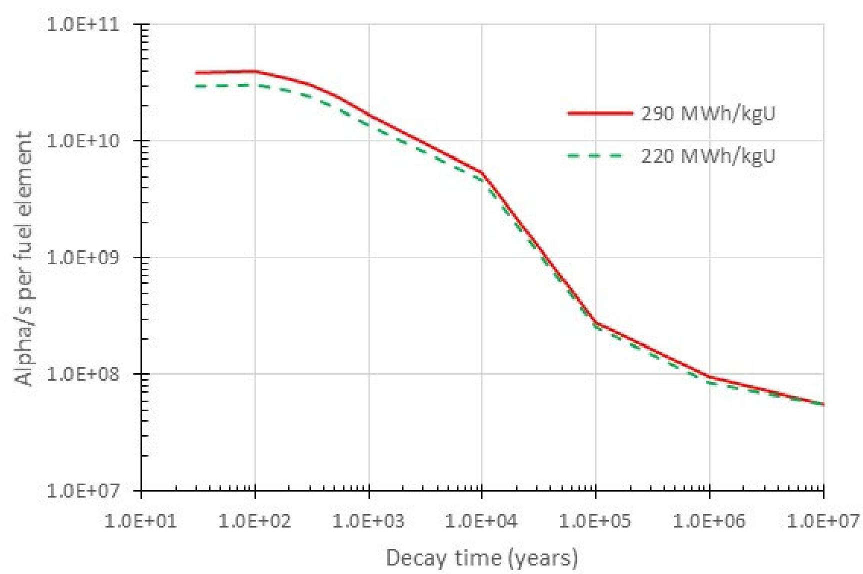


Figure 3: Alpha Source Intensity at Various Decay Times

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

2.2.2 Beta Emissions

In the previous work (Garisto et al. 2009), the beta emission was estimated using the bundle average radionuclide inventory and the radioactive decay data of from Kocher 1981. The average energy of the beta emission was assumed to be 0.3 MeV.

The beta emission spectra were generated for each fuel ring (see Figure 1 for the fuel ring designation) using the radionuclide inventory estimated in Heckman and Edward 2020. The beta emission energy spectra were determined using the ORIGEN module from the SCALE code package. In ORIGEN, beta emission rates and energy spectra were calculated using an analytic expression for the kinetic energy of the emitted beta particles. The beta decay data were derived from the Evaluated Nuclear Structure Data File, which includes beta decay information for 715 beta emitters. The beta decay data considers 8486 beta transition branches.

The beta source intensity and average energy in each ring at different burnup and decay times are listed in Table 3 and Table 4, respectively. The beta emission spectra at different decay times are shown in Figure 4. The beta source intensity at different burnup and decay times is shown in Figure 5. Between 30 and 300 years, the average energy of the emitted beta particles is approximately 0.3 MeV, for which the principal beta sources are Cs-137 and Sr-90. Between 300 and 100,000 years, the average beta energy ranges from 0.1 to 0.15 MeV, for which the principal beta sources during this period are Tc-99 and Np-239. Beyond 100,000 years, the average beta energy increases to 0.25-0.3 MeV. The principal beta sources during this period are Pb-214 and Bi-214 from the U-238 decay series.

Table 3: Beta Source Intensity

		Beta/s source intensity per fuel element			
Bundle Burnup	**Decay Time (Years)**	**Centre**	**Inner**	**Intermediate**	**Outer**
220 MWh/kgU	30	6.02E+11	6.33E+11	7.07E+11	8.68E+11
	100	1.00E+11	1.05E+11	1.16E+11	1.40E+11
	200	9.52E+09	9.96E+09	1.10E+10	1.32E+10
	300	1.10E+09	1.15E+09	1.26E+09	1.50E+09
	500	1.39E+08	1.45E+08	1.60E+08	1.98E+08
	1E+03	1.03E+08	1.09E+08	1.22E+08	1.57E+08
	1E+04	9.61E+07	1.01E+08	1.11E+08	1.37E+08
	1E+05	8.97E+07	9.28E+07	1.00E+08	1.16E+08
	1E+06	5.55E+07	5.65E+07	5.87E+07	6.41E+07
	1E+07	3.94E+07	3.95E+07	3.95E+07	3.97E+07
290 MWh/kgU	30	7.88E+11	8.26E+11	9.12E+11	1.10E+12
	100	1.29E+11	1.34E+11	1.48E+11	1.76E+11
	200	1.22E+10	1.27E+10	1.39E+10	1.65E+10
	300	1.40E+09	1.46E+09	1.60E+09	1.91E+09
	500	1.86E+08	1.95E+08	2.20E+08	2.86E+08
	1E+03	1.44E+08	1.52E+08	1.75E+08	2.36E+08
	1E+04	1.27E+08	1.33E+08	1.49E+08	1.88E+08
	1E+05	1.10E+08	1.14E+08	1.22E+08	1.42E+08
	1E+06	8.03E+07	8.19E+07	8.56E+07	9.47E+07
	1E+07	5.53E+07	5.54E+07	5.55E+07	5.60E+07

Table 4: Beta Source Average Energy

		Average MeV of Emitted Beta Particles			
Bundle Burnup	**Decay Time (Years)**	**Centre**	**Inner**	**Intermediate**	**Outer**
220 MWh/kgU	30	0.36	0.35	0.35	0.34
	100	0.40	0.40	0.40	0.39
	200	0.39	0.39	0.39	0.38
	300	0.33	0.33	0.33	0.33
	500	0.13	0.13	0.13	0.12
	1E+03	0.14	0.13	0.13	0.12
	1E+04	0.15	0.14	0.14	0.13
	1E+05	0.18	0.18	0.17	0.16
	1E+06	0.29	0.29	0.28	0.27
	1E+07	0.35	0.35	0.35	0.35
290 MWh/kgU	30	0.34	0.34	0.34	0.33
	100	0.39	0.39	0.39	0.38
	200	0.38	0.38	0.38	0.38
	300	0.32	0.32	0.32	0.32
	500	0.12	0.12	0.12	0.12
	1E+03	0.12	0.12	0.12	0.11
	1E+04	0.13	0.13	0.13	0.12
	1E+05	0.17	0.17	0.16	0.15
	1E+06	0.28	0.27	0.27	0.26
	1E+07	0.35	0.35	0.35	0.35

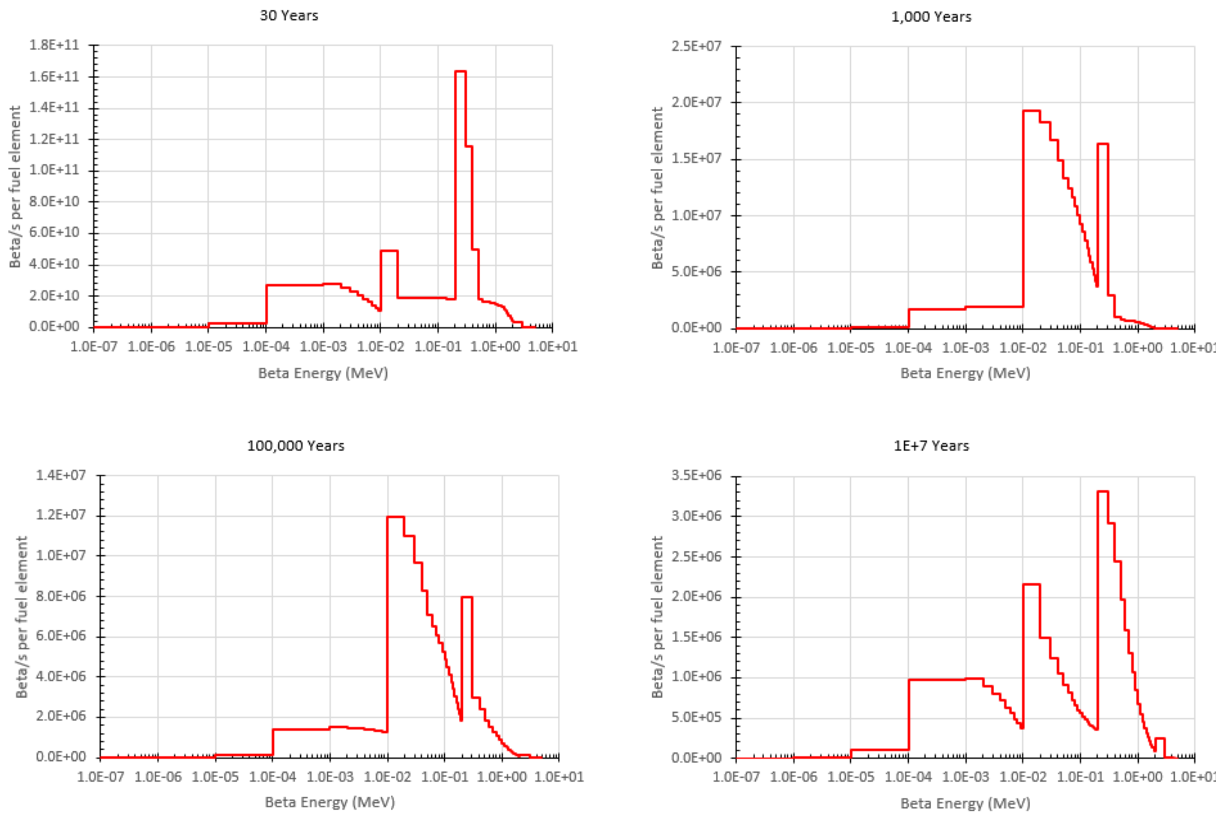


Figure 4: Beta Emission Spectra at Various Decay Times

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

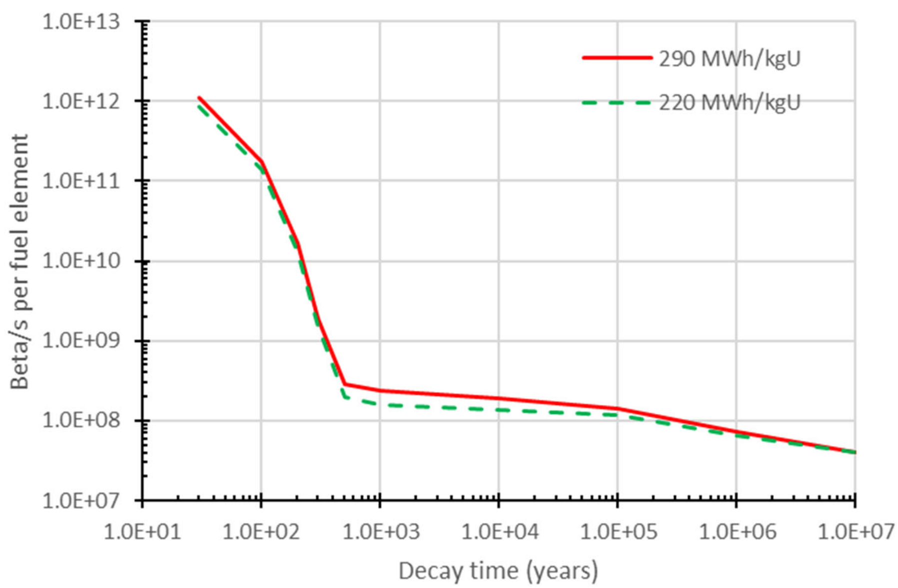


Figure 5: Beta Source Intensity at Various Decay Times

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

2.2.3 Gamma Emissions

Ring-wise gamma emission spectra from Heckman and Edward 2020 were applied. The gamma source intensity and average energy in each ring at different burnup and decay times are listed in Table 5 and Table 6, respectively. The gamma emissions were binned into 20 energy groups. The gamma emission energy spectra at different decay times are shown in Figure 6. The gamma source intensity at different burnup and decay times is shown in Figure 7.

The principal gamma sources are:

- Ba-137m for 30 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 5: 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	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	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 6: Gamma Source Average Energy

		Average MeV of Emitted Gamma Particles			
Bundle Burnup	**Decay Time (Years)**	**Centre**	**Inner**	**Intermediate**	**Outer**
220 MWh/kgU	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	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

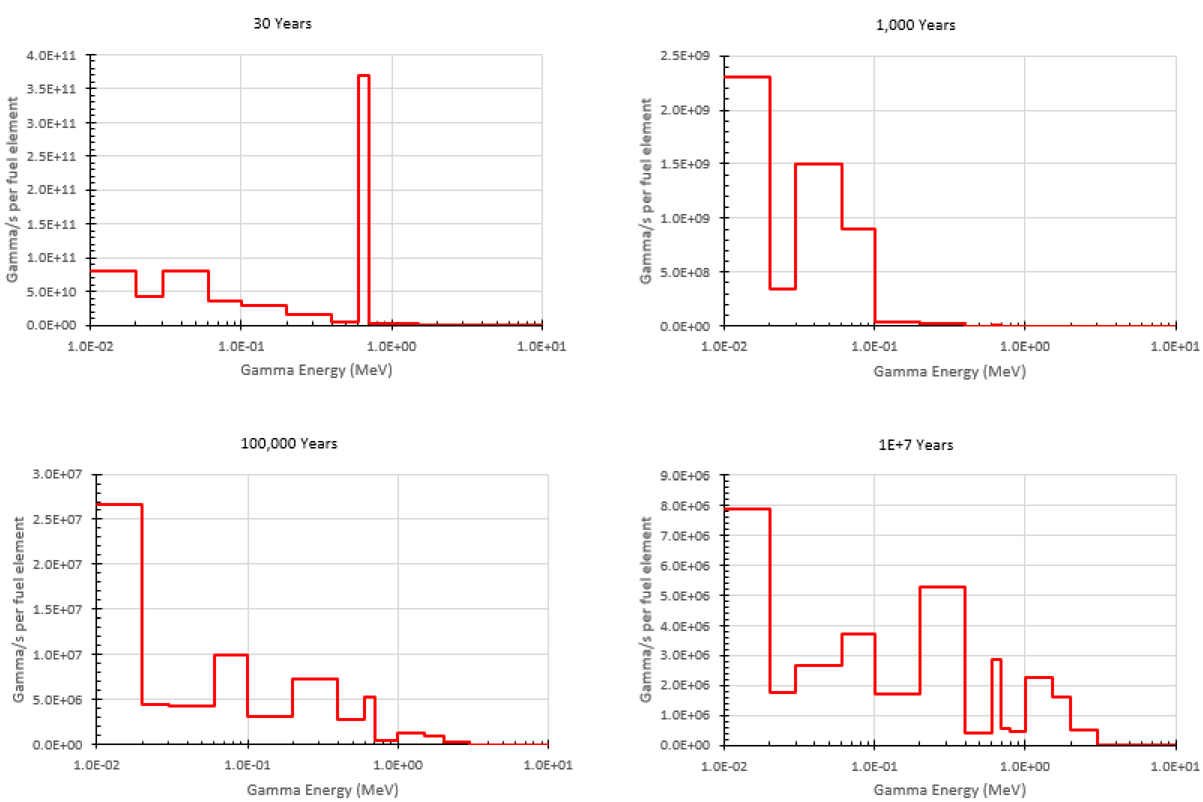


Figure 6: Gamma Emission Spectra at Various Decay Times

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

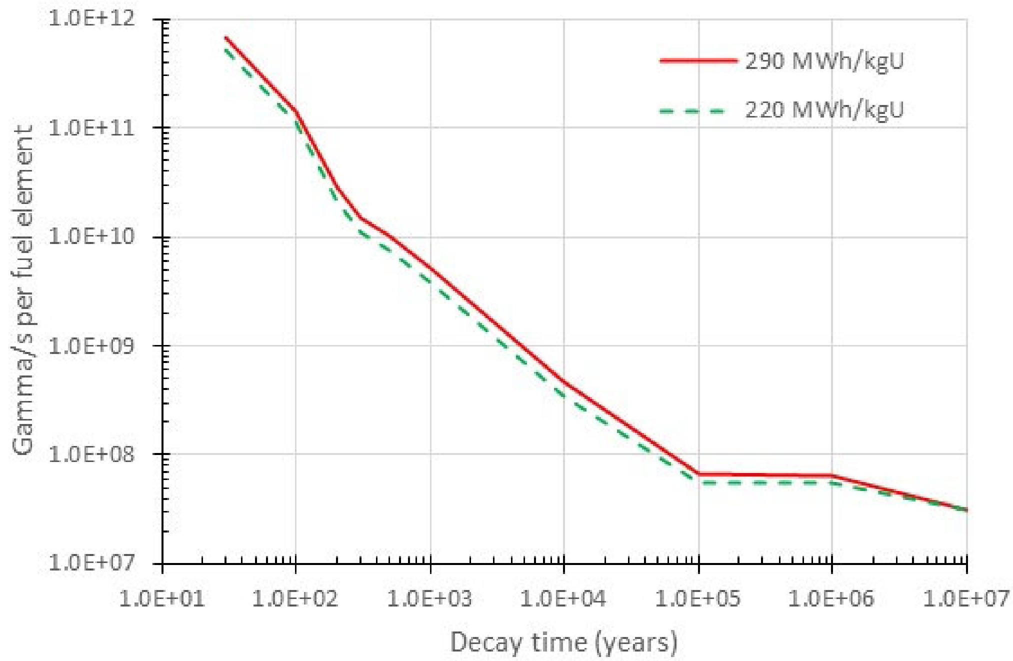


Figure 7: Gamma Source Intensity at Various Decay Times

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

2.2.4 Neutron Emissions

Ring-wise neutron emission spectra from Heckman and Edward 2020 were applied. The neutron source intensity and average energy in each ring at different burnup and decay times are listed in Table 7 and Table 8 respectively. Neutron emissions were binned into 46 energy groups. The neutron emission spectra at different decay times are shown in Figure 8. The neutron source intensity at different burnup and decay times is shown in Figure 9.

Table 7: 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	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
290 MWh/kgU	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 8: Neutron Source Average Energy

		Average MeV of Emitted Neutron			
Bundle Burnup	**Decay Time (Years)**	**Centre Ring**	**Inner Ring**	**Intermediate Ring**	**Outer Ring**
220 MWh/kgU	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	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

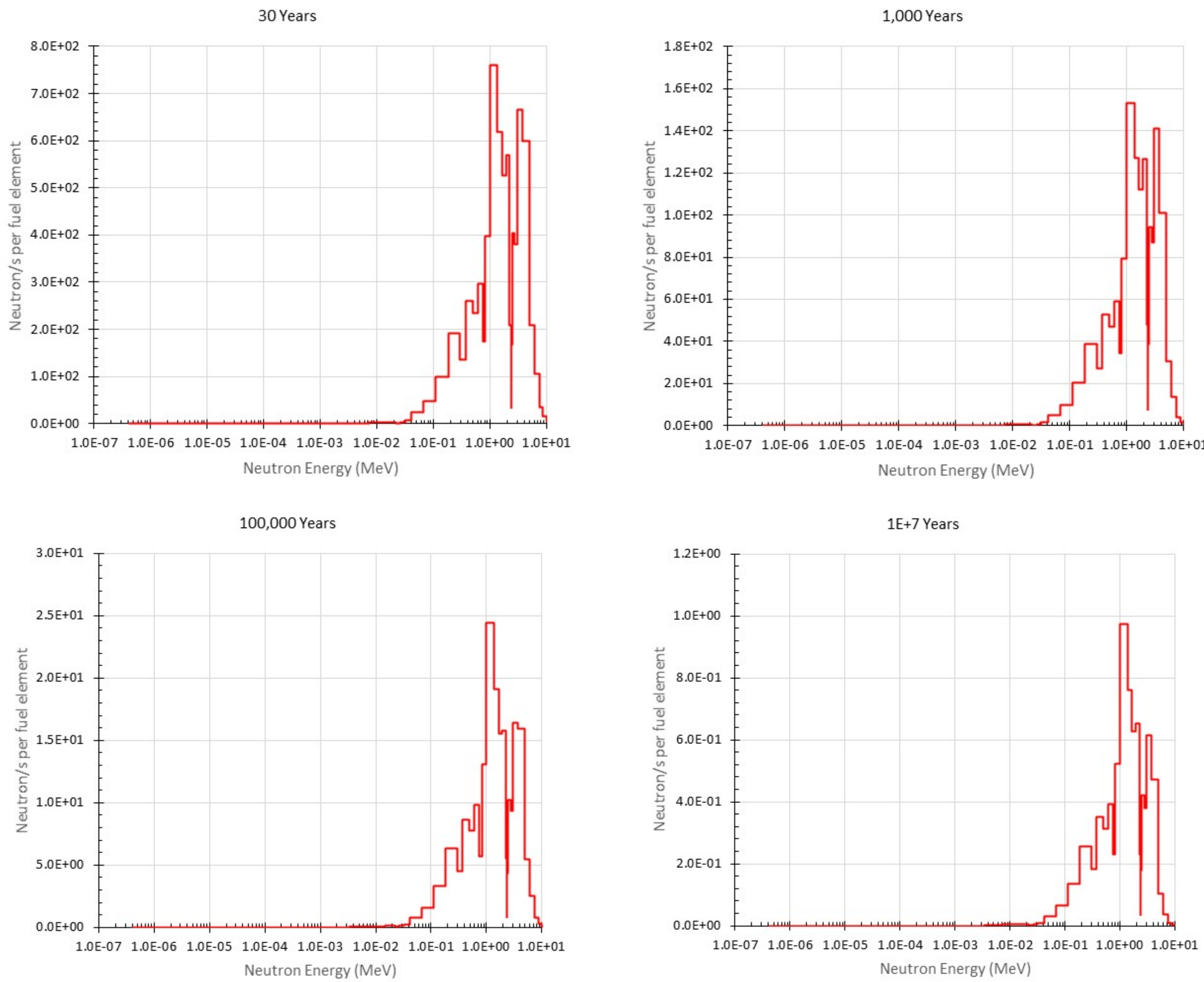


Figure 8: Neutron Emission Spectra at Various Decay Times

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

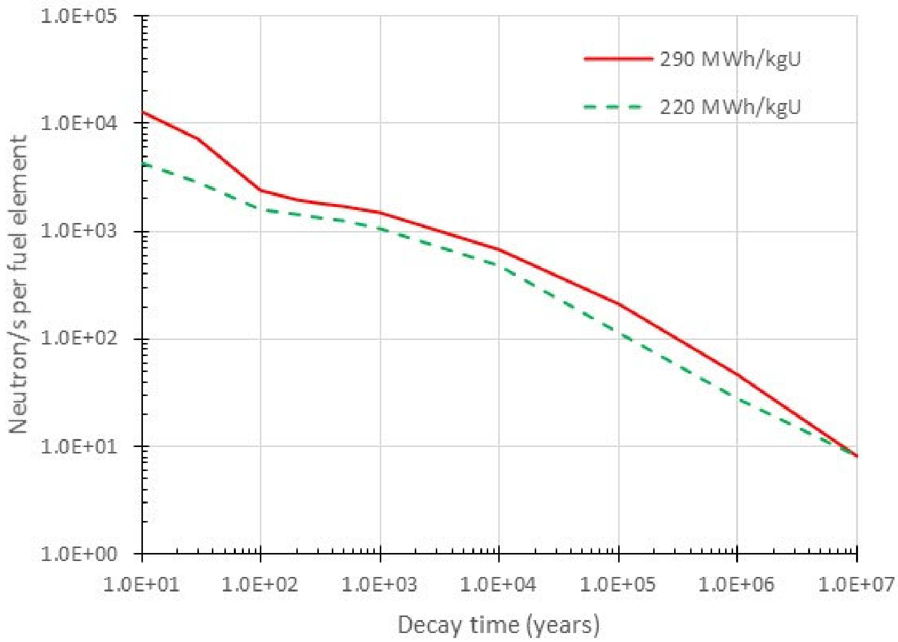


Figure 9: Neutron Source Intensity at Various Decay Times

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

2.3 Particle Transport Calculations

The transport of alpha, beta, 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 neutron, photon, alpha, and beta particles that are being considered in the MCNP simulations are discussed in following subsections.

For the current analysis, the outputs of interest are the dose rates in water in contact with fuel elements or container structures originating from alpha, beta, gamma, or neutron sources in used fuel. The dose rates were calculated using the MCNP F6 track length estimate of energy deposition tally for the corresponding particle. The tally outputs, which were in unit of MeV/s per gram of material, were converted into unit of Gray per hour (Gy/h).

2.3.1 Alpha Particle Transport

In the previous work (Garisto et al. 2009), dose rates from alpha particles emitted from the UO_2 fuel were estimated using the following formula:

$$D_{\alpha}(t) = 1.391 \times 10^{7} \Lambda_{\alpha} \sum_{j} f_{\alpha j}\, H_{act,j}(t) \qquad \text{Eqn. (1)}$$

where:

- $D_\alpha(t)$ is the alpha dose rate in water,
- Λ_α is the relative stopping power of alpha particles in water relative to uranium dioxide. A value of 3.25 was applied based on the assumption that the energy of alpha particles reaching the water at the fuel surface is 2.5 MeV,
- $f_{\alpha j}$ is the fraction of the decay energy of radionuclide *j* that is alpha energy, and
- $H_{act,j}$ is the total thermal decay power of radionuclide *j*.

In the current analysis, dose rates from alpha particles were calculated using the alpha (a) particle transport model in MCNP (mode a). The transport model considered the elastic and inelastic nuclear scattering, continuous slowing-down approximation, multiple Coulomb scattering for angular deflection and energy loss (including energy-loss straggling), and interaction with magnetic fields.

2.3.2 Beta Particle Transport

In the previous analysis (Garisto et al. 2009), dose rates from beta particles emitted from the UO_2 fuel were estimated using the following formula:

$$D_\beta(t) = 2.504 \times 10^7 \Lambda_\beta \sum_j f_{\beta j} H_{act,j}(t) \qquad \text{Eqn. (2)}$$

where:

- $D_\beta(t)$ is the beta dose rate in water,
- Λ_β is the relative stopping power of beta particles in water relative to uranium dioxide. A stopping power of 1.8, which corresponds to 0.3 MeV beta particle, was applied.
- $f_{\beta j}$ is the fraction of the decay energy of radionuclide *j* that is beta energy, and
- $H_{act,j}$ is the total thermal decay power of radionuclide *j*.

For the current analysis, dose rates from beta radiation sources were calculated using the electron (e) and photon (p) transport model in MCNP (mode e p). The transport model in MCNP considers multiple Coulomb scattering for angular deflection and energy loss (including energy-loss straggling), electro ionization, bremsstrahlung, and interaction with magnetic fields. The bremsstrahlung photons were tracked explicitly.

2.3.3 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., using the mode n p in MCNP). The photon transport model considers photonuclear interactions, Raleigh scattering, Compton scattering, photoelectric effects and pair production.

2.3.4 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.

2.4 Modeling Assumptions

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. The skin effect is studied as a sensitivity case.	Basis: The existing source term document (Heckman and Edward 2020) does not have data for spatial distribution within fuel element. Impact: Alpha and beta dose rate in water in contact with fuel element is underestimated. The impact on the gamma and neutron dose rates is small (around one percent).
Water density	0.9655 g/cm^3 for reference case calculations.	Basis: The value is the typical expected density of water in different fuel configuration being analysed. Impact: The impact of low temperature water has been assessed as a sensitivity case (Section 8.4).
Host rock	Crystalline rock for reference case calculations.	Basis: Closer distance between UFCs in the crystalline rock placement room configuration. Impact: The impact of sedimentary rock has been assessed as a sensitivity case (Section 8.5).

2.5 Cases Analysed

Dose rates in water associated with three used fuel geometry configurations were analysed:

a. Single fuel bundle (Section 3),
b. Single UFC (Sections 4 and 5), and
c. Multiple UFCs in a post-closure placement room (Sections 6 and 7).

The list of cases is provided in Table 9.

Dose rates were tabulated in water at the following locations:

- Inside the 50 μm gap between the fuel element and fuel sheath. Dose rates in fuel elements inside each of the four rings (see Figure 1) were tabulated separately.
- Within the fuel bundle envelope (see Figure 1). Dose rates in water region surrounding fuel elements in each of the four rings were tabulated separately.
- Inside of UFC, but outside of fuel bundle envelope.
- Outside of UFC (in water or in saturated bentonite).

Table 9: List of Analysed Cases

Case no.	Case description	Section discussion
1.1	Single fuel bundle, in water	Section 3
1.2	Inside of a water-filled UFC	Section 4.1 (alpha), Section 4.2 (beta), Sections 4.3.1 and 4.3.2 (gamma), and Sections 4.4.1 and 4.4.2 (neutron)
1.3	Outside of a water-filled UFC	Alpha and beta dose rates are negligible. Section 4.3.3 (gamma) and Section 4.4.3 (neutron)
1.4	Outside of a UFC filed with and surrounded by humid air	Section 5 for gamma and neutron
1.5	Multiple UFCs in placement room, inside water-filled UFC	Sections 6.1.1 and 6.1.2 (gamma) and Sections 6.2.1 and 6.2.2 (neutron)
1.6	Multiple UFCs in placement room, outside water-filled UFC	Section 6.1.3 (gamma) and Section 6.2.3 (neutron)
1.7	Multiple UFCs in placement room, outside moist air-filled UFC	Section 7

3. DOSE RATES FROM SINGLE FUEL BUNDLE

In this scenario, the used fuel bundle is submerged in water. The fuel sheath integrity is breached, allowing water ingress into the space or gap between the fuel elements and the fuel sheath. The water density is set to 0.9655 g/cm^3, which is the value at 93°C and 5 MPa. The 93°C temperature is based on the maximum container surface temperature associated with an example sedimentary rock placement room configuration (Guo 2018); the actual temperature would vary depending on the fuel burnup and age.

3.1 Alpha Dose Rates

Only alpha particles that are generated within the 20 μm from the fuel-water interface contribute to the dose rates in water. For a fuel element of 0.61 cm radius with a uniform alpha source distribution, around 99.3% of alpha particles generated in the fuel element do not reach the fuel-water interface. For the alpha particles which reach the fuel-water interface, the dose rates as a function of distance from the fuel element surface are plotted in Figure 10. Essentially all alpha particles escaping the fuel element are deposited within the 50 μm water-filled gap between the fuel element and the fuel sheath. Insignificant alpha particle energy is deposited in the fuel sheath and no energy deposition occurs beyond the fuel sheath.

Maximum dose rates in water in contact with the fuel surface at different decay times are listed in Table 10. The values are plotted in Figure 11. The alpha dose rate increases during the 30 to 100 years decay time due to the buildup of Am-241 (half-life = 432 years) from Pu-241 (half-life = 14 years) decay. Am-241 is the principal alpha source in the used fuel element during that decay period.

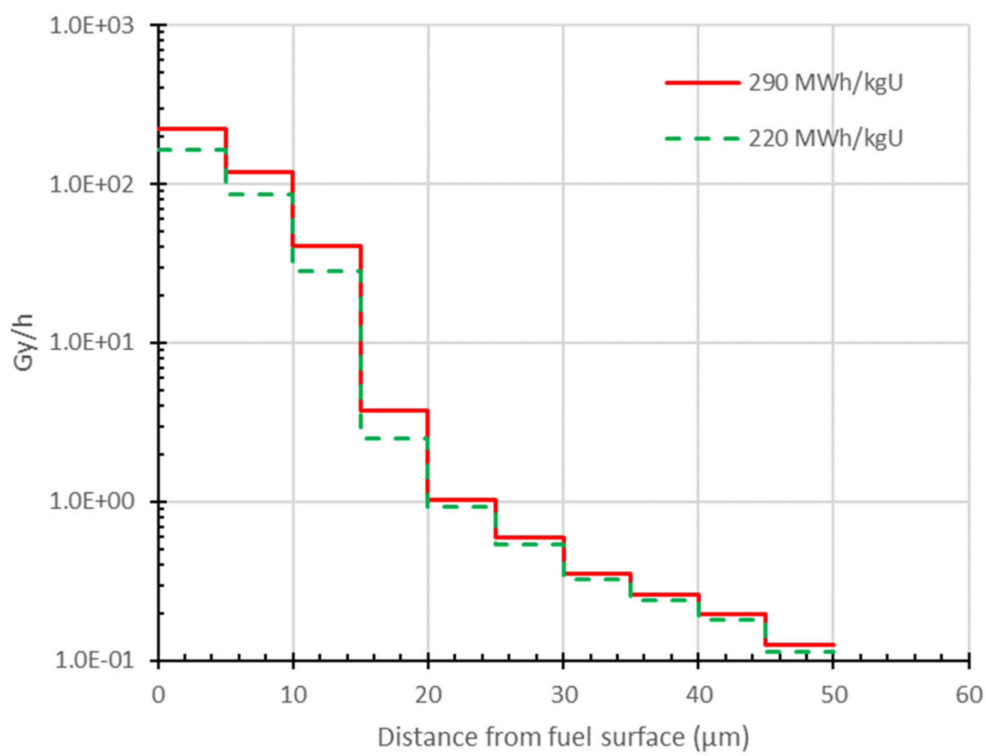


Figure 10: Alpha Dose Rates in Water as a Function of Distance from Fuel Surface

Note: Values plotted above correspond to the outer ring fuel elements at 30 years decay time.

Table 10: Maximum Alpha Dose Rates at Fuel-Water Interface

Bundle Burnup	Decay Time (Years)	Gy/h	Bundle Burnup	Decay Time (Years)	Gy/h
220 MWh/kgU	30	1.65E+02	290 MWh/kgU	30	2.23E+02
	100	1.72E+02		100	2.26E+02
	200	1.50E+02		200	1.95E+02
	300	1.33E+02		300	1.72E+02
	500	1.08E+02		500	1.38E+02
	1E+03	7.30E+01		1E+03	9.10E+01
	1E+04	2.44E+01		1E+04	2.80E+01
	1E+05	1.29E+00		1E+05	1.38E+00
	1E+06	4.25E-01		1E+06	4.84E-01
	1E+07	2.65E-01		1E+07	2.67E-01
Note: the maximum dose rate occurs within the first 5 μm into the water layer.					

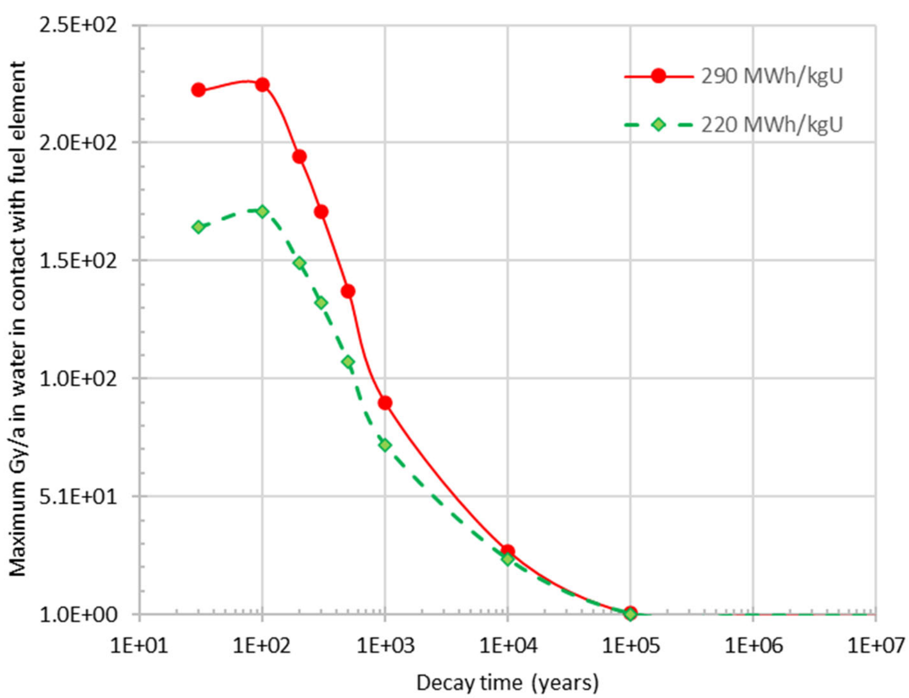
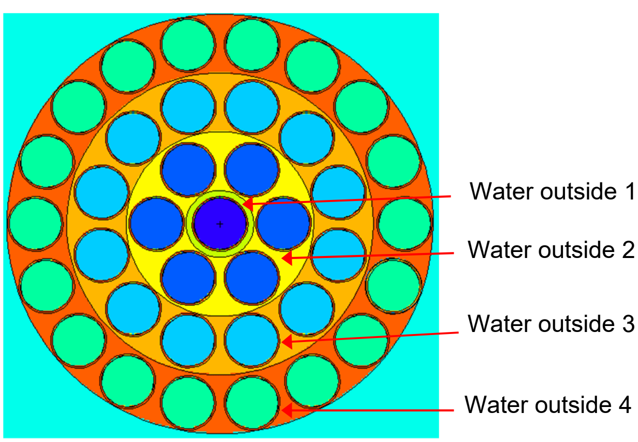


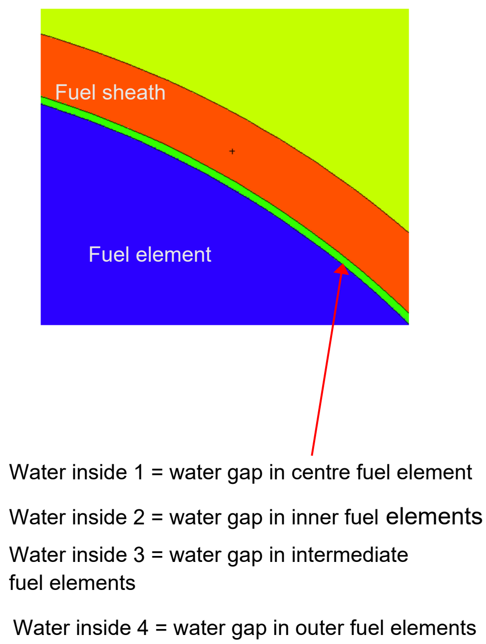
Figure 11: Maximum Alpha Dose Rates in Water

3.2 Beta Dose Rates

For the beta dose rate calculations, contributions from electrons and bremsstrahlung photons were considered. The bremsstrahlung photons were explicitly tracked by invoking “mode p” along with the “mode e” calculations. Approximately 97.9% of the beta energy is deposited within the fuel elements. Around 1.2% of the beta energy is deposited in the zircaloy component of the fuel bundle, 0.1% is deposited in the gap (filled with water) between the fuel element and the fuel sheath, and the remaining 0.8% is deposited in water outside of the fuel sheath. The beta dose rate distribution in water within the fuel bundle envelope and outside of the bundle are illustrated in Figure 12. Figure 13 shows the beta dose rate as a function of distance from the fuel bundle envelope.

The maximum beta dose rates in water are listed in Table 11 and are plotted in Figure 14. The maximum dose rates occur in water that fills the gap between the fuel element and the fuel sheath. At that location, the contribution from bremsstrahlung photons is slightly less than 1% of the total dose rate.





Water location	Gy/h
Water inside 1	1.63E+02
Water inside 2	1.72E+02
Water inside 3	1.86E+02
Water inside 4	2.16E+02
Water outside 1	1.41E+01
Water outside 2	1.18E+01
Water outside 3	1.25E+01
Water outside 4	1.17E+01

Figure 12: Beta Dose Rates in Water inside of Fuel Bundle Envelope (Single Fuel Bundle)

Notes: Values listed above correspond to 290 MWh/kgU bundle at 30 years decay time.

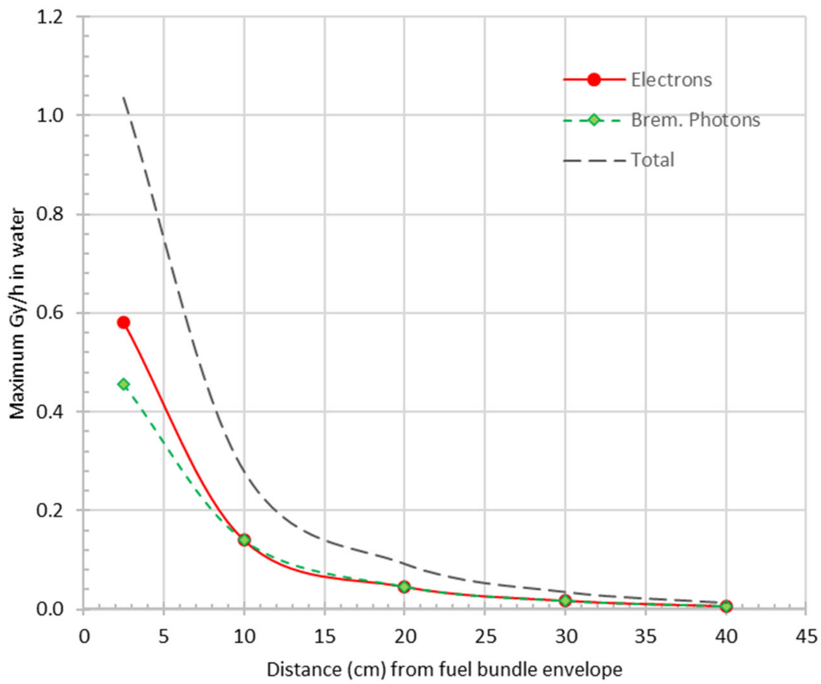


Figure 13: Beta Dose Rates in Water outside of Fuel Bundle Envelope (Single Fuel Bundle)

Notes: Values plotted above correspond to a fuel bundle with 290 MWh/kgU burnup and 30 years decay time.

Table 11: Maximum Beta Dose Rates at Fuel Water Interface

Bundle Burnup	Decay Time (Years)	Electron Gy/h	Brem. Photons Gy/h	Total Gy/h
220 MWh/kgU	30	1.72E+02	1.26E+00	1.73E+02
	100	3.22E+01	2.34E-01	3.25E+01
	200	2.98E+00	2.12E-02	3.00E+00
	300	2.87E-01	1.96E-03	2.89E-01
	500	1.15E-02	5.43E-05	1.16E-02
	1E+03	8.75E-03	3.56E-05	8.78E-03
	1E+04	1.15E-02	3.82E-05	1.15E-02
	1E+05	1.16E-02	5.13E-05	1.16E-02
	1E+06	1.05E-02	6.30E-05	1.05E-02
	1E+07	9.53E-03	7.37E-05	9.60E-03
290 MWh/kgU	30	2.12E+02	1.53E+00	2.13E+02
	100	3.95E+01	2.84E-01	3.98E+01
	200	3.66E+00	2.58E-02	3.69E+00
	300	3.53E-01	2.38E-03	3.56E-01
	500	1.56E-02	6.10E-05	1.57E-02
	1E+03	1.19E-02	3.93E-05	1.20E-02
	1E+04	1.47E-02	4.04E-05	1.47E-02
	1E+05	1.28E-02	5.39E-05	1.28E-02
	1E+06	1.13E-02	6.43E-05	1.13E-02
	1E+07	9.68E-03	7.99E-05	9.76E-03

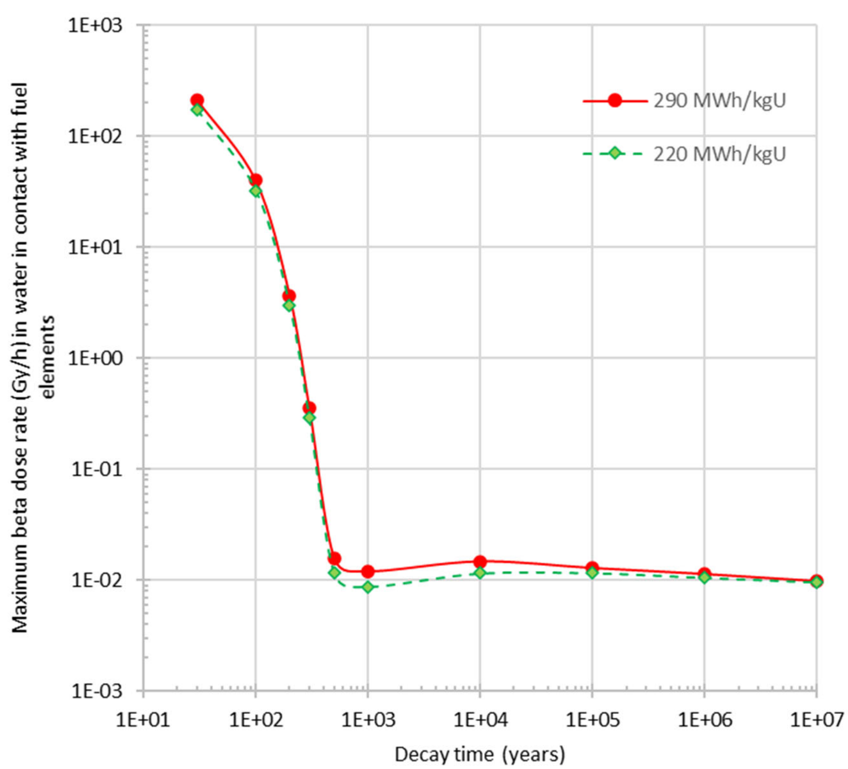
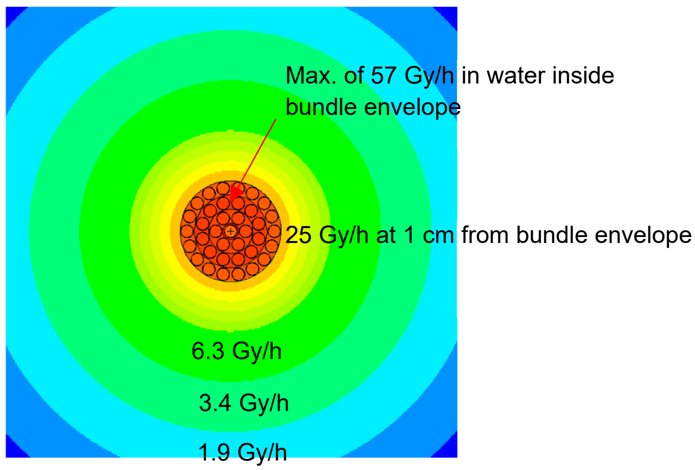


Figure 14: Maximum Beta Dose Rates in Water (Single Fuel Bundle)

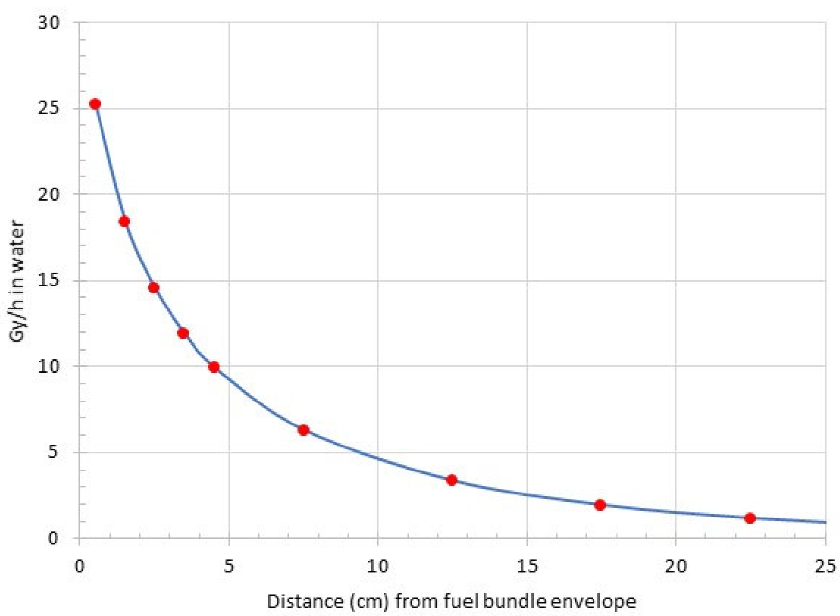
Notes: Values plotted above include contributions from electrons and bremsstrahlung photons.

3.3 Gamma Dose Rates

Approximately 80% of the gamma source energy is deposited within the fuel elements. Around 2.4% of the beta energy is deposited in the zircaloy component of the fuel bundle, 1.4% is deposited in water inside the fuel bundle envelope, and the remaining is deposited outside of the fuel bundle envelope. The gamma dose rate distribution outside of the bundle is illustrated in Figure 15. The maximum dose rates in water are listed in Table 12 and are plotted in Figure 16. The maximum dose rates occur in the interstitial water within the fuel bundle envelope.



(a) Illustration of gamma dose rate tallies outside of a fuel bundle



(b) Plot of gamma dose rate profile outside of a fuel bundle

Figure 15: Gamma Dose Rates in Water outside of Fuel Bundle Envelope

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

Table 12: Maximum Gamma Dose Rate at Fuel-Water Interface (Single Fuel Bundle)

Bundle Burnup	Decay Time (years)	Gy/h	Bundle Burnup	Decay Time (years)	Gy/h
220 MWh/kgU	30	4.32E+01	290 MWh/kgU	30	5.70E+01
	100	8.49E+00		100	1.12E+01
	200	8.57E-01		200	1.13E+00
	300	9.17E-02		300	1.20E-01
	500	3.98E-03		500	5.49E-03
	1E+03	1.84E-03		1E+03	2.72E-03
	1E+04	1.25E-03		1E+04	1.68E-03
	1E+05	3.07E-03		1E+05	3.39E-03
	1E+06	4.61E-03		1E+06	4.70E-03
	1E+07	4.34E-03		1E+07	4.33E-03

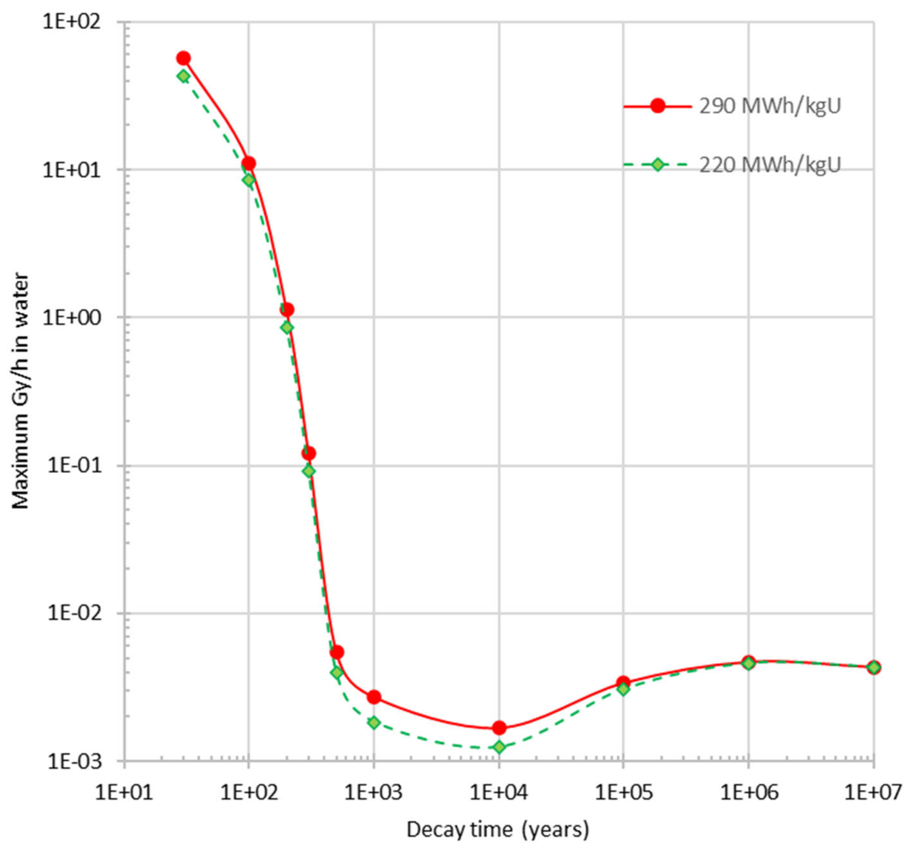
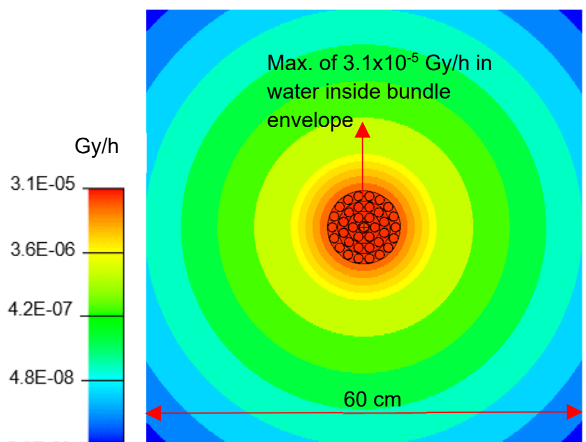


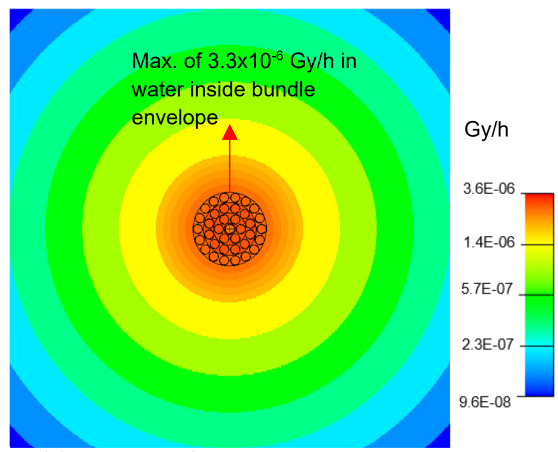
Figure 16: Maximum Gamma Dose Rates in Water (Single Fuel Bundle)

3.4 Neutron Dose Rates

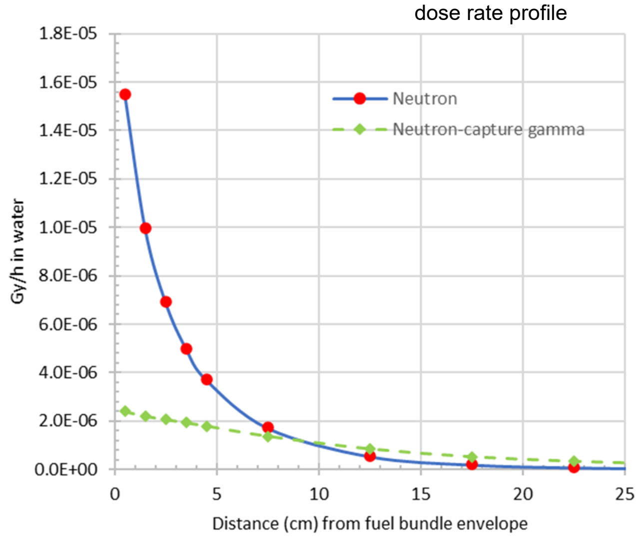
Approximately 80% of the generated energy from neutron sources is deposited in the fuel elements. Small percentages of neutron energy are deposited in the zircaloy component of the fuel bundle, water in gap between fuel element and fuel sheath, and water inside the fuel bundle envelope (0.3%, 0.1%, and 0.2%, respectively). The neutron dose rate distribution outside of the bundle is illustrated in Figure 17. The maximum dose rates in water inside the used fuel bundle envelope are listed in Table 13 and are plotted in Figure 18. Contributions from neutrons and neutron-capture gammas are accounted for. The neutron-capture gamma accounts for approximately 10% of the total dose rate in water inside the fuel bundle envelope. The neutron-capture gamma percentage contribution increases away from the fuel bundle envelope and would dominate the neutron dose rate at 30 cm from the fuel bundle envelope. Dose rates in water from neutron sources are small compared to the dose rates from alpha, beta, and gamma sources described in Sections 3.1, 3.2, and 3.3, respectively.



(a) Illustration of neutron dose rate profile



(b) Illustration of neutron-capture gamma dose rate profile



(c) Plot of dose rates generated by neutron emissions

Figure 17: Neutron Dose Rates in Water outside of Fuel Bundle Envelope (Single Fuel Bundle)

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

Table 13: Maximum Neutron Dose Rate at Fuel-Water Interface (Single Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Gy/h	Bundle Burnup	Decay Time (Years)	Gy/h
220 MWh/kgU	30	1.52E-05	290 MWh/kgU	30	3.46E-05
	100	1.02E-05		100	1.52E-05
	200	9.22E-06		200	1.29E-05
	300	8.74E-06		300	1.22E-05
	500	8.01E-06		500	1.12E-05
	1E+03	6.93E-06		1E+03	9.65E-06
	1E+04	2.97E-06		1E+04	4.26E-06
	1E+05	5.57E-07		1E+05	1.03E-06
	1E+06	1.55E-07		1E+06	2.45E-07
	1E+07	6.22E-08		1E+07	6.22E-08

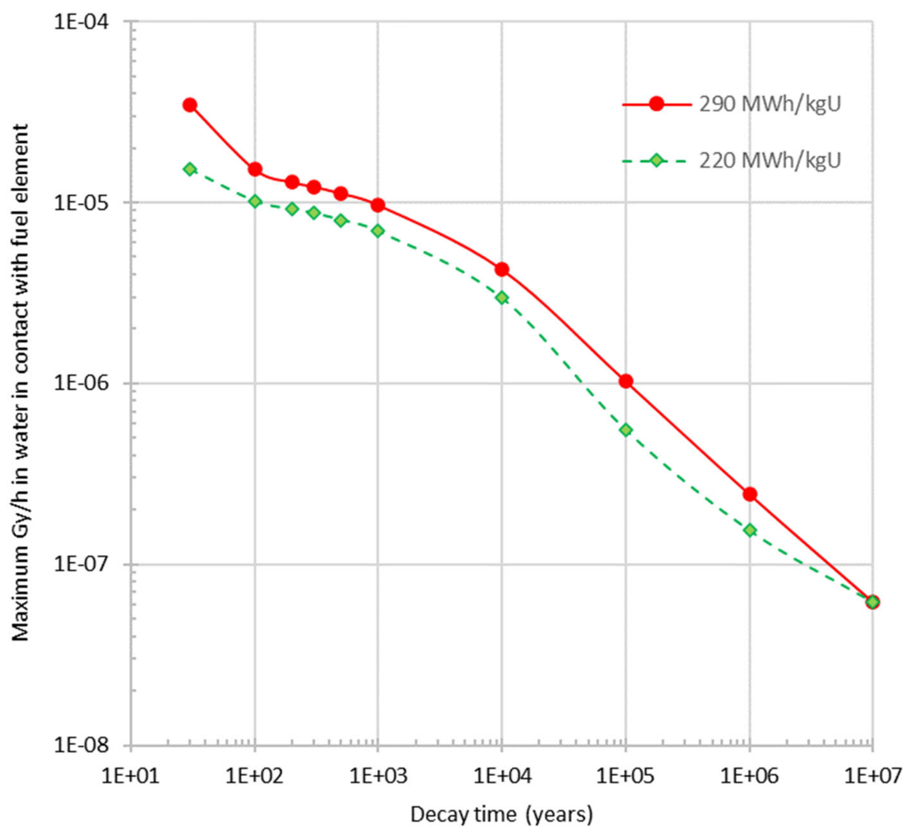
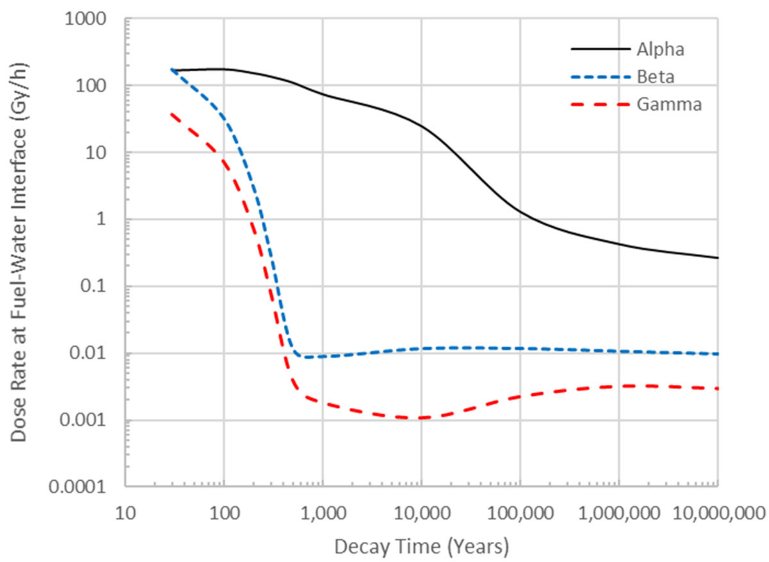


Figure 18: Maximum Neutron Dose Rate in Water (Single Fuel Bundle)

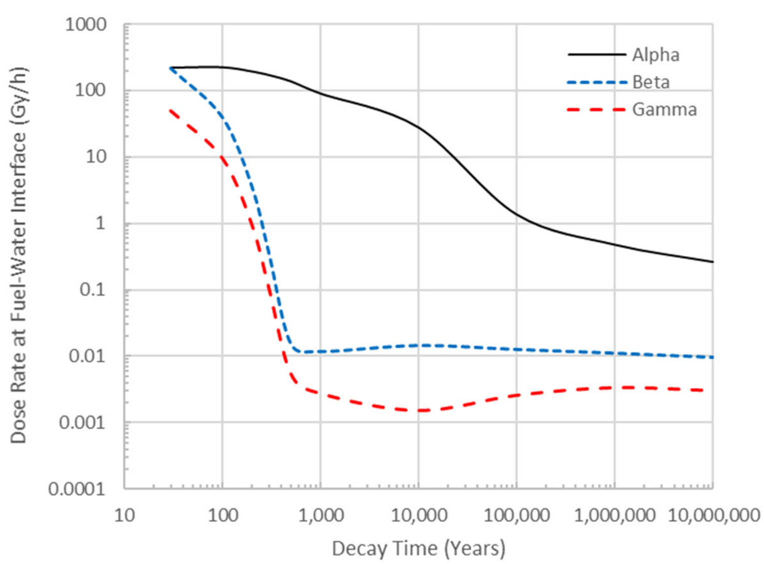
3.5 Total Dose Rates

The maximum total dose rate in water from alpha, beta, gamma, and neutron for the single fuel bundle configuration occurs in water that fills the gap between the outer-ring fuel element and the fuel sheath. The maximum dose rates as a function of decay times are shown in Figure 19.

Alpha and beta sources contribute almost equally at 30 years decay time. The alpha dose rates become dominant at decay times greater than 100 years since beta and gamma emitting fission products in the fuel have decayed significantly.



(a) Bundle Burnup = 220 MWh/kgU



(b) Bundle Burnup = 290 MWh/kgU

Figure 19: Maximum Dose Rates at Fuel-Water Interface (Single Fuel Bundle)

Table 14: Maximum Dose Rates at Fuel-Water Interface (Single Fuel Bundle)

			Percent Contribution			
Burnup	**Decay Time (years)**	**Total Gy/h**	**Alpha**	**Beta**	**Gamma**	**Neutron**
220 MWh/kgU	30	3.76E+02	44.0%	46.1%	10.0%	0.0%
	100	2.12E+02	81.2%	15.3%	3.5%	0.0%
	200	1.54E+02	97.6%	2.0%	0.5%	0.0%
	300	1.34E+02	99.7%	0.2%	0.1%	0.0%
	500	1.08E+02	100.0%	0.0%	0.0%	0.0%
	1E+03	7.30E+01	100.0%	0.0%	0.0%	0.0%
	1E+04	2.44E+01	99.9%	0.0%	0.0%	0.0%
	1E+05	1.30E+00	98.9%	0.9%	0.2%	0.0%
	1E+06	4.39E-01	96.9%	2.4%	0.7%	0.0%
	1E+07	2.78E-01	95.5%	3.5%	1.1%	0.0%
290 MWh/kgU	30	4.86E+02	46.0%	43.9%	10.1%	0.0%
	100	2.75E+02	82.0%	14.5%	3.5%	0.0%
	200	2.00E+02	97.7%	1.8%	0.5%	0.0%
	300	1.72E+02	99.7%	0.2%	0.1%	0.0%
	500	1.38E+02	100.0%	0.0%	0.0%	0.0%
	1E+03	9.10E+01	100.0%	0.0%	0.0%	0.0%
	1E+04	2.80E+01	99.9%	0.1%	0.0%	0.0%
	1E+05	1.40E+00	98.9%	0.9%	0.2%	0.0%
	1E+06	4.99E-01	97.1%	2.3%	0.7%	0.0%
	1E+07	2.80E-01	95.4%	3.5%	1.1%	0.0%

4. DOSE RATES FROM SINGLE USED FUEL CONTAINER (WATER FILLED)

In this scenario, the UFC is submerged in water. The UFC is breached, allowing water ingress into the space inside the UFC. The fuel sheath integrity is also breached, allowing water ingress into the space or gap between the fuel element and the fuel sheath. The water density is the same as that of the single fuel bundle scenario in Section 3. Dose rates were evaluated at three locations:

1) Fuel-water interface
2) Internal UFC-water interface
3) External UFC-water interface

4.1 Alpha Dose Rates

4.1.1 Fuel-Water Interface

Due to the short range of alpha particles, the maximum dose rate in water in contact with fuel element is not impacted by neighbouring fuel bundles in a single UFC. Dose rates presented in Section 3.1 apply.

4.1.2 Internal UFC-Water Interface

Due to the short range of alpha particles, the alpha dose rate at the internal UFC-water interface is zero.

4.1.3 External UFC-Water Interface

Due to the short range of alpha particles, the alpha dose rate at the external UFC-water interface is zero.

4.2 Beta Dose Rates

4.2.1 Fuel-Water Interface

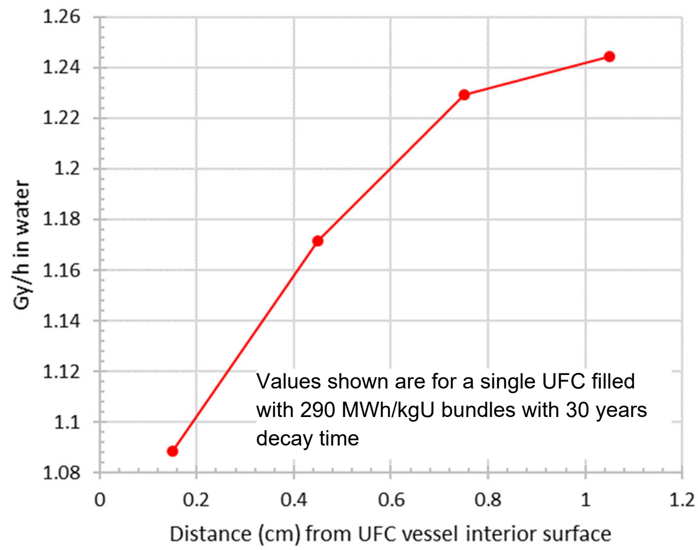
The maximum dose rates at the fuel-water interface for the single UFC configuration are listed in Table 15. The values are slightly higher than the ones given in Table 16 (single fuel bundle).

Table 15: Maximum Beta Dose Rates at Fuel-Water Interface (Single UFC)

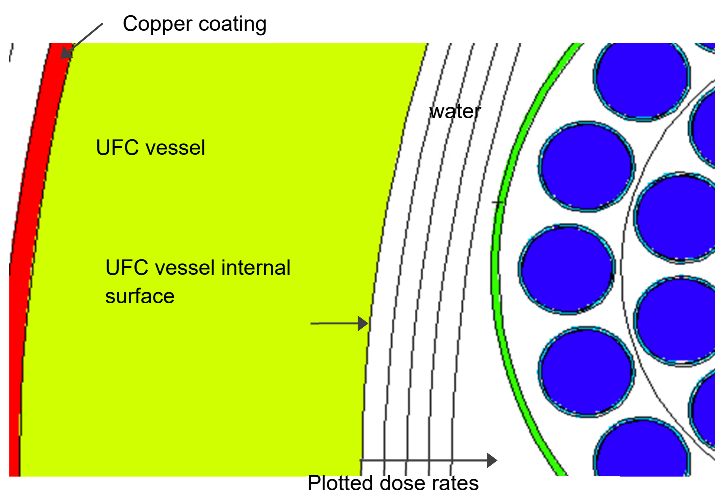
Bundle Burnup	Decay Time (Years)	Gy/h	Bundle Burnup	Decay Time (Years)	Gy/h
220 MWh/kgU	30	1.77E+02	290 MWh/kgU	30	2.16E+02
	100	3.31E+01		100	4.03E+01
	200	3.06E+00		200	3.76E+00
	300	2.97E-01		300	3.61E-01
	500	1.18E-02		500	1.59E-02
	1E+03	8.92E-03		1E+03	1.22E-02
	1E+04	1.16E-02		1E+04	1.48E-02
	1E+05	1.19E-02		1E+05	1.29E-02
	1E+06	1.08E-02		1E+06	1.14E-02
	1E+07	1.04E-02		1E+07	1.05E-02

4.2.2 Internal UFC-Water Interface

Beta dose rates at the UFC-water interface were tabulated at different distances from the UFC vessel internal surface. The dose rate profile near the internal surface of the UFC is shown in Figure 20. The maximum dose rates inside the UFC outside of the fuel bundle envelope at different bundle burnup and decay times are tabulated in Table 16.



(a) Plot of beta dose rates vs. locations inside the UFC



(b) Illustration of the locations of dose rates plotted in Figure (a)

Figure 20: Beta Dose Rates in Water Inside UFC (Single UFC)

Table 16: Beta Dose Rates in Water Inside UFC (Single UFC)

Bundle Burnup	Decay Time (years)	Water A (Gy/h)	Water B (Gy/h)
220 MWh//kgU	30	1.58E+00	9.19E-01
	100	2.94E-01	1.74E-01
	200	2.65E-02	1.54E-02
	300	2.46E-03	1.43E-03
	500	6.51E-05	3.75E-05
	1E+03	4.28E-05	2.50E-05
	1E+04	4.45E-05	2.83E-05
	1E+05	6.25E-05	3.95E-05
	1E+06	7.82E-05	4.49E-05
	1E+07	7.15E-05	4.38E-05
290 MWh//kgU	30	1.92E+00	1.09E+00
	100	3.60E-01	2.09E-01
	200	3.19E-02	1.86E-02
	300	3.01E-03	1.67E-03
	500	7.23E-05	4.16E-05
	1E+03	4.52E-05	2.84E-05
	1E+04	4.78E-05	2.60E-05
	1E+05	6.51E-05	3.95E-05
	1E+06	7.78E-05	4.70E-05
	1E+07	7.22E-05	4.35E-05

Notes:
Water A = water inside UFC but outside of fuel bundle envelope.
Water B = water closest to the internal surface of the UFC (at 3 mm from the internal UFC surface).

4.2.3 External UFC-Water Interface

Due to the short range of beta particles, the beta dose rate at the external UFC-water interface is zero.

4.3 Gamma Dose Rates

4.3.1 Fuel-Water Interface

The maximum gamma dose rates in water for the single UFC configuration are listed in Table 17. The maximum dose rate occurs in water between fuel elements in the outer ring of the fuel bundle. The maximum dose rates in water within the bundle envelope are 15 to 30% higher than the dose rate from a single fuel bundle due to the presence of additional gamma sources in neighbouring bundles inside the UFC.

Table 17: Maximum Gamma Dose Rate at Fuel-Water Interface (Single UFC)

Bundle Burnup	Decay Time (Years)	Gy/h	Ratio to dose rate from single bundle configuration*
220 MWh/kgU	30	5.15E+01	1.19
	100	1.01E+01	1.19
	200	1.02E+00	1.19
	300	1.10E-01	1.20
	500	5.27E-03	1.32
	1E+03	2.21E-03	1.20
	1E+04	1.52E-03	1.22
	1E+05	3.57E-03	1.16
	1E+06	5.40E-03	1.17
	1E+07	5.10E-03	1.17
290 MWh/kgU	30	6.77E+01	1.19
	100	1.32E+01	1.18
	200	1.34E+00	1.19
	300	1.44E-01	1.20
	500	7.09E-03	1.29
	1E+03	3.27E-03	1.20
	1E+04	2.08E-03	1.23
	1E+05	3.93E-03	1.16
	1E+06	5.53E-03	1.18
	1E+07	5.09E-03	1.18
* See Table 12 for the single bundle dose rates.			

4.3.2 Internal UFC-Water Interface

Dose rates at the UFC-water interface were tabulated at different distances from the UFC vessel internal surfaces. The dose rates are plotted in Figure 21. The maximum dose rates inside the UFC outside of the fuel bundle envelope at different bundle burnup and decay times are tabulated in Table 18.

Table 18: Gamma Dose Rates in Water Inside UFC (Single UFC)

Bundle Burnup	Decay Time (years)	Water A (Gy/h)	Water B (Gy/h)
220 MWh/kgU	30	2.42E+01	1.40E+01
	100	4.75E+00	2.75E+00
	200	4.79E-01	2.78E-01
	300	5.06E-02	2.95E-02
	500	1.67E-03	9.61E-04
	1E+03	7.01E-04	3.93E-04
	1E+04	6.84E-04	3.99E-04
	1E+05	1.64E-03	9.71E-04
	1E+06	2.44E-03	1.45E-03
	1E+07	2.27E-03	1.35E-03
290 MWh/kgU	30	3.18E+01	1.85E+01
	100	6.22E+00	3.61E+00
	200	6.27E-01	3.63E-01
	300	6.61E-02	3.84E-02
	500	2.26E-03	1.28E-03
	1E+03	1.06E-03	5.94E-04
	1E+04	9.27E-04	5.40E-04
	1E+05	1.82E-03	1.08E-03
	1E+06	2.48E-03	1.48E-03
	1E+07	2.27E-03	1.35E-03

Notes:
Water A = water inside UFC but outside of the fuel bundle envelope.
Water B = water closest to the internal surface of the UFC (at 3 mm from the internal UFC surface).

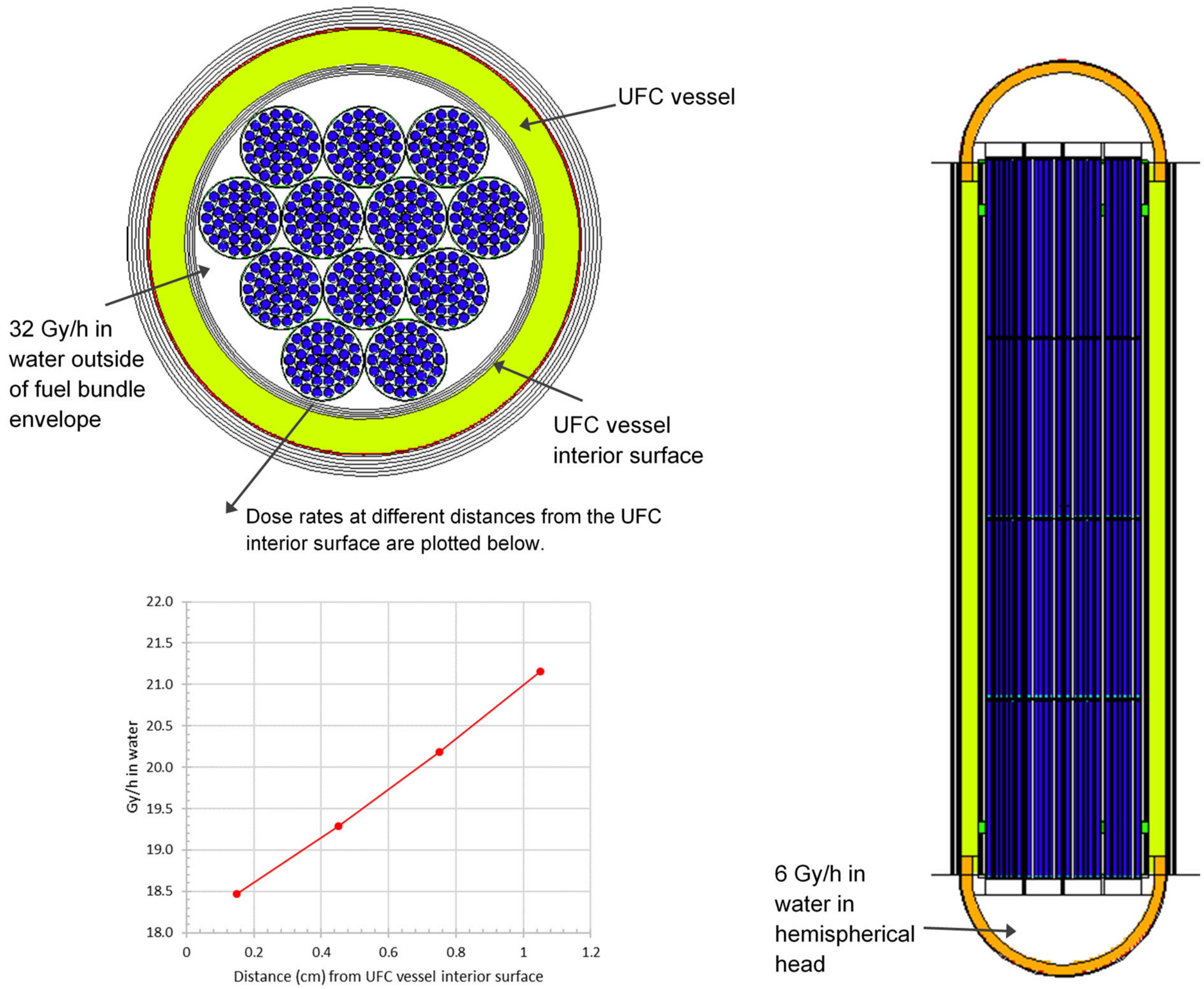
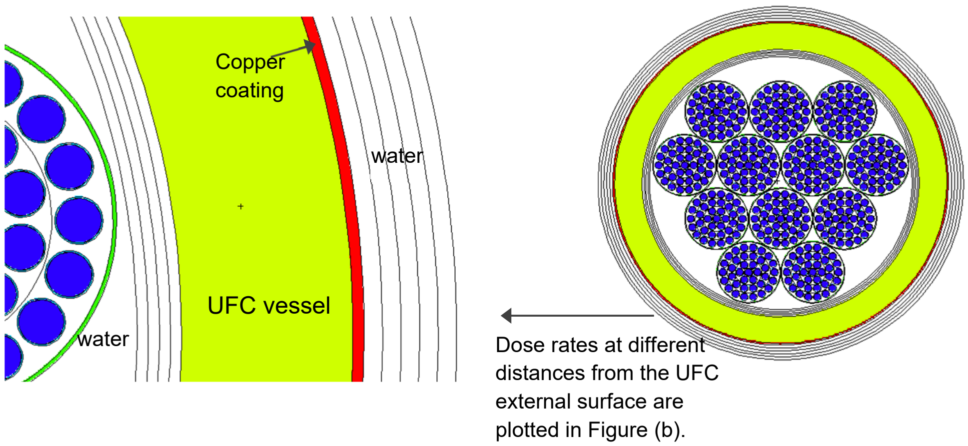


Figure 21: Gamma Dose Rates in Water inside UFC (Single UFC)

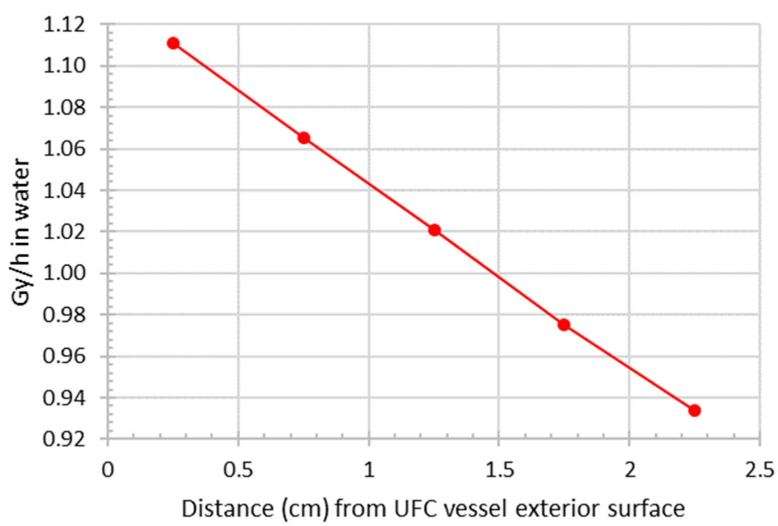
Note: Values plotted above correspond to a UFC filled with bundles with 290 MWh/kgU burnup and 30 years decay time.

4.3.3 External UFC-Water Interface

Dose rates in water as a function of distance from the UFC external surface are shown in Figure 22.The maximum dose rates outside of a UFC associated with different bundle burnup and decay times are listed in Table 19.



(a) Illustration of locations of dose rates plotted in Figure (b)



(b) Plot of gamma dose rates vs. water locations

Figure 22: Gamma Dose Rates in Water Outside UFC (Single UFC)

Note: Values plotted above correspond to a UFC filled with bundles with 290 MWh/kgU burnup and 30 years decay time.

Table 19: Gamma Dose Rates in Water Outside UFC (Single UFC)

Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC	Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC
220 MWh/kgU	30	8.43E-01	290 MWh/kgU	30	1.11E+00
	100	1.64E-01		100	2.15E-01
	200	1.69E-02		200	2.20E-02
	300	1.92E-03		300	2.44E-03
	500	6.56E-05		500	8.80E-05
	1E+03	1.85E-05		1E+03	2.52E-05
	1E+04	2.77E-05		1E+04	3.40E-05
	1E+05	1.02E-04		1E+05	1.09E-04
	1E+06	1.67E-04		1E+06	1.70E-04
	1E+07	1.60E-04		1E+07	1.60E-04

4.4 Neutron Dose Rates

Dose rates presented in this section includes the contributions from neutrons and neutron-capture gammas.

4.4.1 Fuel-Water Interface

The maximum neutron dose rates in water for the single UFC configuration are listed in Table 20. The maximum dose rate occurs in the interstitial water between fuel elements in the fuel bundle. The maximum dose rates in water within the bundle envelope are approximately twice the dose rates from a single fuel bundle due to the presence of additional neutron sources in neighbouring bundles inside the UFC.

Table 20: Maximum Neutron Dose Rate at Fuel-Water Interface (Single UFC)

Bundle Burnup	Decay Time (Years)	Gy/h	Ratio to dose rate from single bundle configuration*
220 MWh/kgU	30	3.26E-05	2.14
	100	2.05E-05	2.01
	200	1.84E-05	1.99
	300	1.73E-05	1.98
	500	1.59E-05	1.99
	1.E+03	1.37E-05	1.98
	1.E+04	5.90E-06	1.98
	1.E+05	1.20E-06	2.15
	1.E+06	3.12E-07	2.02
	1.E+07	1.15E-07	1.85
290 MWh/kgU	30	7.75E-05	2.24
	100	3.05E-05	2.01
	200	2.56E-05	1.98
	300	2.41E-05	1.98
	500	2.21E-05	1.98
	1.E+03	1.90E-05	1.97
	1.E+04	8.47E-06	1.99
	1.E+05	2.22E-06	2.16
	1.E+06	5.05E-07	2.06
	1.E+07	1.15E-07	1.85

* See Table 13 for the single bundle dose rates.

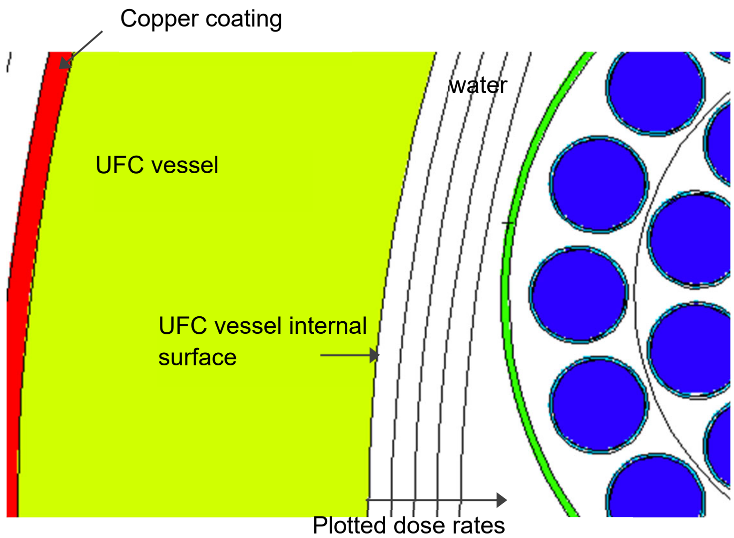
4.4.2 Internal UFC-Water Interface

Dose rates at the UFC-water interface were tabulated at different distances from the internal UFC surface. The dose rates are plotted in Figure 23. The maximum dose rates inside the UFC outside of the fuel bundle envelope at different bundle burnup and decay times are tabulated in Table 21.

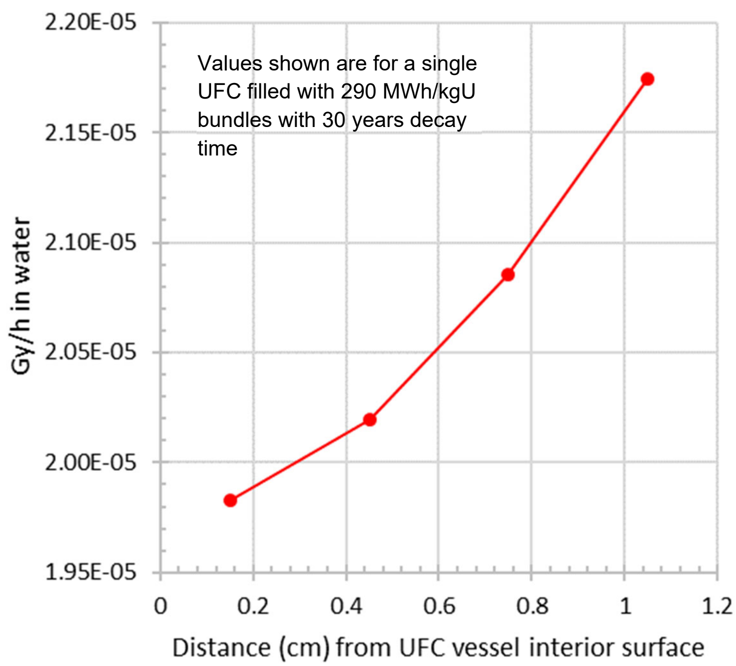
Table 21: Neutron Dose Rates in Water Inside UFC (Single UFC)

Bundle Burnup	Decay Time (years)	Water A (Gy/h)	Water B (Gy/h)
220 MWh/kgU	30	1.37E-05	8.36E-06
	100	8.39E-06	5.13E-06
	200	7.49E-06	4.57E-06
	300	7.08E-06	4.32E-06
	500	6.48E-06	3.96E-06
	1E+03	5.56E-06	3.39E-06
	1E+04	2.41E-06	1.47E-06
	1E+05	5.05E-07	3.07E-07
	1E+06	1.31E-07	7.97E-08
	1E+07	4.37E-08	2.67E-08
290 MWh/kgU	30	3.26E-05	1.98E-05
	100	1.25E-05	7.65E-06
	200	1.04E-05	6.33E-06
	300	9.77E-06	5.96E-06
	500	8.93E-06	5.45E-06
	1E+03	7.66E-06	4.67E-06
	1E+04	3.44E-06	2.10E-06
	1E+05	9.35E-07	5.69E-07
	1E+06	2.13E-07	1.30E-07
	1E+07	4.37E-08	2.67E-08

Notes:
Water A = water inside UFC but outside of the fuel bundle envelope.
Water B = water closest to the internal surface of the UFC (at 3 mm from the internal UFC surface).



(a) Illustration of locations of dose rates plotted in Figure (b)



(b) Plot of neutron dose rates vs. water locations

Figure 23: Neutron Dose Rates in Water inside UFC (Single UFC)

4.4.3 External UFC-Water Interface

Dose rates in water as a function of distance from the UFC external surface are shown in Figure 24.The maximum dose rates outside of a UFC associated with different bundle burnup and decay times are listed in Table 22.

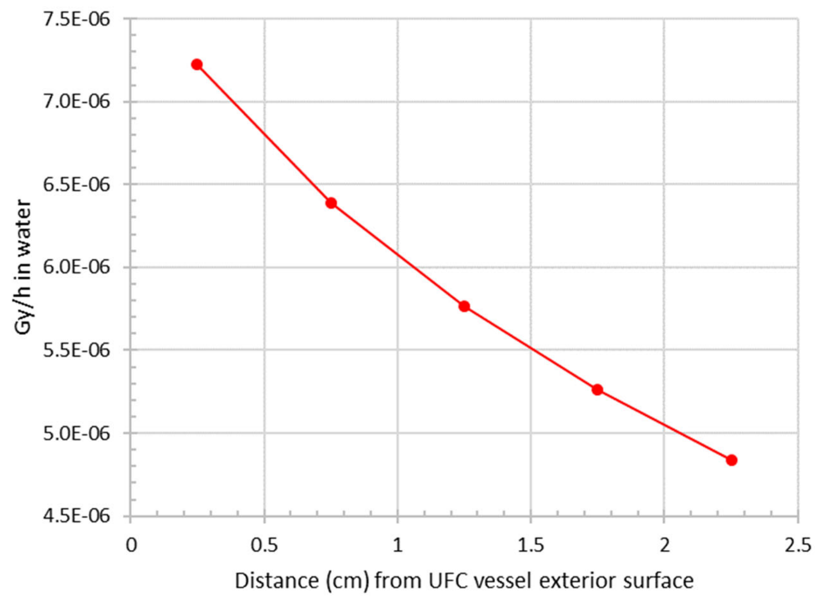


Figure 24: Neutron Dose Rates in Water Outside UFC (Single UFC)

Note: Values plotted above correspond to a UFC filled with bundles with 290 MWh/kgU burnup and 30 years decay time.

Table 22: Neutron Dose Rates in Water Outside UFC (Single UFC)

Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC	Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC
220 MWh/kgU	30	3.05E-06	290 MWh/kgU	30	7.23E-06
	100	1.88E-06		100	2.80E-06
	200	1.67E-06		200	2.31E-06
	300	1.58E-06		300	2.18E-06
	500	1.45E-06		500	1.99E-06
	1E+03	1.24E-06		1E+03	1.70E-06
	1E+04	5.34E-07		1E+04	7.63E-07
	1E+05	1.11E-07		1E+05	2.06E-07
	1E+06	2.89E-08		1E+06	4.69E-08
	1E+07	9.61E-09		1E+07	9.62E-09

4.5 Total Dose Rates

The total dose rates from the alpha, beta, gamma, and neutron emissions are listed in Table 23 (fuel-water interface), Table 24 (internal UFC-water interface), and Table 25 (external UFC-water interface).

Table 23: Total Dose Rates at Fuel-Water Interface (Single UFC)

Bundle Burnup	Decay Time (years)	Total Gy/h	Bundle Burnup	Decay Time (years)	Total Gy/h
220 MWh/kgU	30	3.93E+02	290 MWh/kgU	30	5.07E+02
	100	2.15E+02		100	2.79E+02
	200	1.54E+02		200	2.00E+02
	300	1.34E+02		300	1.72E+02
	500	1.08E+02		500	1.38E+02
	1E+03	7.30E+01		1E+03	9.10E+01
	1E+04	2.44E+01		1E+04	2.80E+01
	1E+05	1.31E+00		1E+05	1.40E+00
	1E+06	4.41E-01		1E+06	5.01E-01
	1E+07	2.81E-01		1E+07	2.83E-01

Table 24: Total Dose Rates at Internal UFC-Water Interface (Single UFC)

Bundle Burnup	Decay Time (years)	Water A (Gy/h)	Water B (Gy/h)
220 MWh/kgU	30	2.58E+01	1.50E+01
	100	5.04E+00	2.93E+00
	200	5.06E-01	2.93E-01
	300	5.31E-02	3.09E-02
	500	1.74E-03	1.00E-03
	1E+03	7.49E-04	4.22E-04
	1E+04	7.31E-04	4.29E-04
	1E+05	1.70E-03	1.01E-03
	1E+06	2.51E-03	1.49E-03
	1E+07	2.34E-03	1.39E-03
290 MWh/kgU	30	3.38E+01	1.96E+01
	100	6.58E+00	3.82E+00
	200	6.59E-01	3.82E-01
	300	6.91E-02	4.01E-02
	500	2.34E-03	1.33E-03
	1E+03	1.11E-03	6.27E-04
	1E+04	9.79E-04	5.68E-04
	1E+05	1.89E-03	1.12E-03
	1E+06	2.56E-03	1.52E-03
	1E+07	2.34E-03	1.39E-03
Notes: Water A = water inside UFC but outside of the fuel bundle envelope. Water B = water closest to the internal of the UFC (at 3 mm from the internal UFC surface).			

Table 25: Total Dose Rates at External UFC-Water Interface (Single UFC)

Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC	Bundle Burnup	Decay Time (years)	Maximum Gy/h outside of UFC
220 MWh/kgU	30	8.43E-01	290 MWh/kgU	30	1.11E+00
	100	1.64E-01		100	2.15E-01
	200	1.69E-02		200	2.20E-02
	300	1.92E-03		300	2.44E-03
	500	6.70E-05		500	8.80E-05
	1E+03	1.97E-05		1E+03	2.52E-05
	1E+04	2.83E-05		1E+04	3.40E-05
	1E+05	1.02E-04		1E+05	1.09E-04
	1E+06	1.67E-04		1E+06	1.70E-04
	1E+07	1.60E-04		1E+07	1.60E-04

5. DOSE RATES FROM SINGLE USED FUEL CONTAINER (MOIST AIR)

In this scenario, the UFC is filled with and surrounded by moist air with 70% relative humidity. Similar to the analysis presented in Section 4, the free space between the fuel element and the fuel sheath is also filled with the moist air. Dose rates were tabulated at the air-container interface outside of the UFC. The maximum dose rates at the external UFC interface are shown in Table 26. With the moist air filling the UFC instead of water, more gammas and neutrons escape the UFC resulting in approximately 15% increase in the dose rates at the external UFC interface.

Table 26: Maximum Gamma and Neutron Dose Rates in Moist Air outside UFC (Single UFC)

Burnup	Decay Time (years)	Gy/h			Ratio to Water Filled UFC
		Gamma	Neutron	Total	
220 MWh/kgU	30	9.70E-01	4.41E-06	9.70E-01	1.15
	100	1.89E-01	2.71E-06	1.89E-01	1.15
	200	1.94E-02	2.43E-06	1.94E-02	1.15
	300	2.20E-03	2.30E-06	2.20E-03	1.15
	500	8.01E-05	2.11E-06	8.22E-05	1.15
	1E+03	2.17E-05	1.82E-06	2.35E-05	1.18
	1E+04	3.20E-05	7.89E-07	3.28E-05	1.16
	1E+05	1.17E-04	1.65E-07	1.17E-04	1.15
	1E+06	1.92E-04	4.33E-08	1.92E-04	1.15
	1E+07	1.83E-04	1.49E-08	1.83E-04	1.15
290 MWh/kgU	30	1.28E+00	1.05E-05	1.28E+00	1.15
	100	2.47E-01	4.05E-06	2.47E-01	1.15
	200	2.53E-02	3.36E-06	2.53E-02	1.15
	300	2.82E-03	3.17E-06	2.82E-03	1.15
	500	9.96E-05	2.91E-06	1.03E-04	1.17
	1E+03	3.03E-05	2.51E-06	3.28E-05	1.18
	1E+04	3.94E-05	1.13E-06	4.06E-05	1.17
	1E+05	1.25E-04	3.06E-07	1.25E-04	1.15
	1E+06	1.95E-04	7.00E-08	1.95E-04	1.15
	1E+07	1.83E-04	1.49E-08	1.83E-04	1.15
Note: neutron dose rates include the contribution from the neutron-capture gammas.					

6. DOSE RATES IN PLACEMENT ROOM (SATURATED WITH WATER)

In this scenario, each UFC is placed inside a bentonite box. The bentonite boxes are stacked two-high in the crystalline rock placement room (see Appendix A.1.3 for the details on the placement room geometry). All UFCs in the calculation model are assumed to be breached, allowing water ingress into the space inside the UFCs. The fuel sheath integrity is also breached, allowing water ingress into the space or gap between the fuel element and the fuel sheath. The water condition is the same as that of the single fuel bundle in Section 3.

As in Section 3, dose rates are presented at the fuel-water interface and the UFC-water interfaces (internal and external).

6.1 Gamma Dose Rates

6.1.1 Fuel-Water Interface

The maximum gamma dose rates in water within the fuel bundle envelope for the placement room configuration are essentially the same as that of a single UFC presented in Section 4.3.1. Results shown in Table 27 indicate that the contribution from gamma sources from neighbouring UFCs is insignificant when compared to the contribution from gamma sources within the UFC.

Table 27: Maximum Gamma Dose Rates at Fuel-Water Interface (Placement Room)

Burnup	Decay Time (years)	Gy/h	Ratio to Single UFC*
220 MWh/kgU	30	5.14E+01	1.00
	100	1.01E+01	1.00
	200	1.02E+00	1.00
	300	1.10E-01	1.00
	500	4.98E-03	0.99
	1E+03	2.19E-03	1.00
	1E+04	1.51E-03	0.99
	1E+05	3.60E-03	1.01
	1E+06	5.52E-03	1.02
	1E+07	5.15E-03	1.01
290 MWh/kgU	30	6.76E+01	1.00
	100	1.32E+01	1.00
	200	1.33E+00	1.00
	300	1.43E-01	1.00
	500	6.77E-03	0.99
	1E+03	3.22E-03	1.00
	1E+04	2.05E-03	0.99
	1E+05	3.94E-03	1.00
	1E+06	5.65E-03	1.02
	1E+07	5.16E-03	1.01

Notes:
The ratio was obtained against the dose rate from Table 17.
The ratio value that is slightly less than 1.00 is an artifact of the statistical uncertainty (see Appendix B for the list of statistical uncertainties). The expected ratio is 1.0.

6.1.2 Internal UFC-Water Interface

The gamma dose rates in water within the UFC, but outside of the fuel bundle envelope, for the placement room configuration are essentially the same as that of a single UFC presented in Section 4.3.2. Results shown in Table 28 indicate that the contribution from gamma sources from neighbouring UFCs is insignificant when compared to the contribution from gamma sources within the UFC where the dose rates are calculated.

Table 28: Gamma Dose Rates in Water Inside UFC (Placement Room)

Bundle Burnup	Decay Time (years)	Gy/h		Ratio to Single UFC*	
		Water A	Water B	Water A	Water B
220 MWh/kgU	30	2.42E+01	1.40E+01	1.00	1.00
	100	4.75E+00	2.75E+00	1.00	1.00
	200	4.79E-01	2.78E-01	1.00	1.00
	300	5.06E-02	2.91E-02	1.00	0.99
	500	1.64E-03	9.41E-04	0.98	0.98
	1E+03	6.86E-04	3.87E-04	0.98	0.98
	1E+04	6.83E-04	3.96E-04	1.00	0.99
	1E+05	1.64E-03	9.72E-04	1.00	1.00
	1E+06	2.43E-03	1.44E-03	1.00	1.00
	1E+07	2.27E-03	1.35E-03	1.00	1.00
290 MWh/kgU	30	3.19E+01	1.85E+01	1.00	1.00
	100	6.23E+00	3.60E+00	1.00	1.00
	200	6.27E-01	3.64E-01	1.00	1.00
	300	6.60E-02	3.80E-02	1.00	0.99
	500	2.23E-03	1.27E-03	0.99	1.00
	1E+03	1.05E-03	5.90E-04	0.99	0.99
	1E+04	9.27E-04	5.36E-04	1.00	0.99
	1E+05	1.82E-03	1.08E-03	1.00	1.00
	1E+06	2.48E-03	1.47E-03	1.00	1.00
	1E+07	2.26E-03	1.34E-03	1.00	1.00

Notes:
* The ratio was obtained against the dose rate from Table 18.
Water A = water inside UFC but outside of fuel bundle envelope.
Water B = water closest to the internal of the UFC (at 3 mm from the internal UFC surface).

6.1.3 External UFC-Water Interface

The maximum gamma dose rates outside of the UFC (at the external UFC-water interface) are listed in Table 29. The dose rates are slightly lower than that from a single UFC submerged in water (Section 4.3.3) due to the different material outside of the UFC (saturated bentonite vs. water).

Table 29: Maximum Gamma Dose Rates Outside UFC (Placement Room)

Burnup	Decay Time (years)	Gy/h	Ratio to Single UFC*
220 MWh/kgU	30	7.97E-01	0.95
	100	1.55E-01	0.95
	200	1.59E-02	0.94
	300	1.76E-03	0.92
	500	6.57E-05	1.00
	1E+03	1.76E-05	0.95
	1E+04	2.58E-05	0.93
	1E+05	9.44E-05	0.93
	1E+06	1.56E-04	0.93
	1E+07	1.50E-04	0.94
290 MWh/kgU	30	1.05E+00	0.95
	100	2.03E-01	0.95
	200	2.07E-02	0.94
	300	2.26E-03	0.93
	500	8.09E-05	0.92
	1E+03	2.47E-05	0.98
	1E+04	3.19E-05	0.94
	1E+05	1.01E-04	0.93
	1E+06	1.58E-04	0.93
	1E+07	1.49E-04	0.93
Note: the ratio was obtained against the dose rate from Table 19.			

6.2 Neutron Dose Rates

Dose rates presented in this section includes the contributions from neutrons and neutron-capture gammas.

6.2.1 Fuel-Water Interface

The maximum neutron dose rates in water within the fuel bundle envelope for the placement room configuration are essentially the same as that of a single UFC presented in Section 4.4.1. Results shown in Table 30 indicate that the contribution from neutron sources from neighbouring UFCs is insignificant when compared to the contribution from neutron sources within the UFC where the dose rates are calculated.

Table 30: Maximum Neutron Dose Rates at Fuel-Water Interface (Placement Room)

Burnup	Decay Time (Years)	Gy/h	Ratio to Single UFC*
220 MWh/kgU	30	3.26E-05	1.00
	100	2.04E-05	1.00
	200	1.83E-05	1.00
	300	1.73E-05	1.00
	500	1.58E-05	0.99
	1E+03	1.36E-05	0.99
	1E+04	5.86E-06	0.99
	1E+05	1.20E-06	1.00
	1E+06	3.13E-07	1.00
	1E+07	1.15E-07	1.00
290 MWh/kgU	30	7.77E-05	1.00
	100	3.04E-05	1.00
	200	2.54E-05	0.99
	300	2.40E-05	0.99
	500	2.21E-05	1.00
	1E+03	1.89E-05	0.99
	1E+04	8.41E-06	0.99
	1E+05	2.22E-06	1.00
	1E+06	5.06E-07	1.00
	1E+07	1.15E-07	1.00

Notes:
The ratio was obtained against dose rates from Table 20.
The ratio value that is slightly less than 1.00 is an artifact of the statistical uncertainty (see Appendix B for the list of statistical uncertainties). The expected ratio is 1.0.

6.2.2 Internal UFC-Water Interface

Neutron dose rates in water within the UFC, but outside of the fuel bundle envelope, for the placement room configuration are marginally higher than the dose rates from a single UFC presented in Section 4.4.2. Results shown in Table 31 indicate that the contribution from neutron sources from neighbouring UFCs is small when compared to the contribution from neutron sources within the UFC where the dose rates are calculated.

Table 31: Neutron Dose Rates in Water Inside UFC (Placement Room)

Burnup	Decay Time (years)	Gy/h		Ratio to Single UFC*	
		Water A	Water B	Water A	Water B
220 MWh/kgU	30	1.38E-05	8.62E-06	1.01	1.03
	100	8.46E-06	5.28E-06	1.01	1.03
	200	7.55E-06	4.71E-06	1.01	1.03
	300	7.14E-06	4.45E-06	1.01	1.03
	500	6.53E-06	4.08E-06	1.01	1.03
	1E+03	5.60E-06	3.49E-06	1.01	1.03
	1E+04	2.42E-06	1.51E-06	1.01	1.03
	1E+05	5.09E-07	3.17E-07	1.01	1.03
	1E+06	1.32E-07	8.20E-08	1.01	1.03
	1E+07	4.40E-08	2.75E-08	1.01	1.03
290 MWh/kgU	30	3.28E-05	2.04E-05	1.01	1.03
	100	1.26E-05	7.88E-06	1.01	1.03
	200	1.04E-05	6.53E-06	1.01	1.03
	300	9.84E-06	6.14E-06	1.01	1.03
	500	9.00E-06	5.61E-06	1.01	1.03
	1E+03	7.72E-06	4.81E-06	1.01	1.03
	1E+04	3.46E-06	2.16E-06	1.01	1.03
	1E+05	9.42E-07	5.86E-07	1.01	1.03
	1E+06	2.14E-07	1.34E-07	1.01	1.03
	1E+07	4.40E-08	2.74E-08	1.01	1.03

Notes:
* The ratio was obtained against the dose rate from Table 21.
Water A = water inside UFC but outside of fuel bundle envelope.
Water B = water closest to the internal of the UFC (at 3 mm from the internal UFC surface).

6.2.3 External UFC-Water Interface

The maximum neutron dose rates in water outside of the UFC in the placement room configuration (Table 32) are approximately half of the dose rates from a single UFC presented in Section 4.4.3. This is due to the different materials outside of the UFC. For the placement room configuration, the external UFC dose rates are calculated in saturated bentonite with 2.02 g/cm^3 density. For the single UFC, the material is water with density of 0.9655 g/cm^3. The neutron absorption in water is higher than that in saturated bentonite.

Table 32: Maximum Neutron Dose Rates Outside UFC (Placement Room)

Burnup	Decay Time (Years)	Gy/h	Ratio to Single UFC*
220 MWh/kgU	30	1.50E-06	0.49
	100	9.21E-07	0.49
	200	8.25E-07	0.49
	300	7.79E-07	0.49
	500	7.16E-07	0.50
	1E+03	6.15E-07	0.50
	1E+04	2.67E-07	0.50
	1E+05	5.62E-08	0.50
	1E+06	1.46E-08	0.50
	1E+07	4.95E-09	0.52
290 MWh/kgU	30	3.57E-06	0.49
	100	1.38E-06	0.49
	200	1.14E-06	0.49
	300	1.07E-06	0.49
	500	9.87E-07	0.50
	1E+03	8.49E-07	0.50
	1E+04	3.82E-07	0.50
	1E+05	1.04E-07	0.50
	1E+06	2.37E-08	0.51
	1E+07	4.96E-09	0.52

Note: the ratio was obtained against the dose rate from Table 22.
For the placement room configuration, the material outside of the UFC is saturated bentonite. For the single UFC configuration, the material outside of the UFC is water.

6.3 Total Dose Rates

The maximum dose rates in water at the UFC-water interface in the post-closure placement room are listed in Table 33. Contributions from gamma and neutrons (including neutron-capture gammas) are accounted for; however, the dose rates at both UFC-water interfaces are dominated by the dose rates from gamma sources. Dose rates at the internal UFC-water interface are essentially the same as that of a single UFC submerged in water. The dose rate at the external UFC-water interface is lower in the placement room configuration due to the different materials outside of the UFC (saturated bentonite vs. water).

Table 33: Maximum Dose Rates at UFC Interface (Placement Room)

Burnup	Decay Time (Years)	Gy/h		Ratio to single UFC	
		Internal UFC Interface	External UFC Interface	Internal UFC Interface	External UFC Interface
220 MWh/kgU	30	1.40E+01	7.97E-01	1.00	0.95
	100	2.75E+00	1.55E-01	1.00	0.95
	200	2.78E-01	1.59E-02	1.00	0.94
	300	2.91E-02	1.76E-03	1.00	0.92
	500	9.45E-04	6.64E-05	1.00	0.99
	1E+03	3.90E-04	1.82E-05	1.00	0.93
	1E+04	3.97E-04	2.60E-05	1.00	0.92
	1E+05	9.73E-04	9.45E-05	1.00	0.93
	1E+06	1.45E-03	1.56E-04	1.00	0.93
	1E+07	1.35E-03	1.50E-04	1.00	0.94
290 MWh/kgU	30	1.85E+01	1.05E+00	1.00	0.95
	100	3.60E+00	2.03E-01	1.00	0.95
	200	3.64E-01	2.07E-02	1.00	0.94
	300	3.80E-02	2.26E-03	1.00	0.93
	500	1.28E-03	8.19E-05	1.00	0.91
	1E+03	5.95E-04	2.56E-05	1.00	0.95
	1E+04	5.38E-04	3.23E-05	1.00	0.93
	1E+05	1.08E-03	1.02E-04	1.00	0.93
	1E+06	1.47E-03	1.58E-04	1.00	0.93
	1E+07	1.34E-03	1.49E-04	1.00	0.93

Notes: For the placement room configuration, the material outside of the UFC is saturated bentonite. For the single UFC configuration, the material outside of the UFC is water.

7. DOSE RATES IN PLACEMENT ROOM (SATURATED WITH MOIST AIR)

In this scenario, the placement room is saturated with moist air with 70% relative humidity. Similar to the analysis presented in Section 5, the free space inside the UFC and between the fuel element and the fuel sheath are also filled with the moist air. Dose rates were tabulated at the air-container interface outside of the UFC. The maximum gamma and neutron dose rates at the external UFC interface are tabulated in Table 34. The moist air environment results in higher dose rates at the external UFC interface due to more gammas and neutrons escaping the UFC.

Table 34: Maximum Dose Rates Outside UFC (Placement Room Saturated with Moist Air)

Burnup	Decay Time (years)	Gy/h			Ratio to Placement Room Saturated with Water
		Gamma	Neutron	Total	
220 MWh/kgU	30	1.13E+00	2.71E-06	1.13E+00	1.42
	100	2.21E-01	1.66E-06	2.21E-01	1.42
	200	2.26E-02	1.48E-06	2.26E-02	1.42
	300	2.49E-03	1.41E-06	2.49E-03	1.41
	500	8.92E-05	1.30E-06	9.05E-05	1.36
	1E+03	2.54E-05	1.13E-06	2.65E-05	1.45
	1E+04	3.60E-05	4.89E-07	3.65E-05	1.40
	1E+05	1.26E-04	1.04E-07	1.26E-04	1.34
	1E+06	2.07E-04	2.73E-08	2.07E-04	1.32
	1E+07	1.97E-04	9.47E-09	1.97E-04	1.32
290 MWh/kgU	30	1.49E+00	6.46E-06	1.49E+00	1.42
	100	2.89E-01	2.48E-06	2.89E-01	1.42
	200	2.94E-02	2.06E-06	2.94E-02	1.42
	300	3.21E-03	1.95E-06	3.21E-03	1.42
	500	1.11E-04	1.79E-06	1.13E-04	1.37
	1E+03	3.55E-05	1.56E-06	3.71E-05	1.45
	1E+04	4.50E-05	7.03E-07	4.57E-05	1.42
	1E+05	1.36E-04	1.92E-07	1.36E-04	1.34
	1E+06	2.09E-04	4.41E-08	2.09E-04	1.32
	1E+07	1.96E-04	9.46E-09	1.96E-04	1.32
Note: neutron dose rates include the contribution from the neutron-capture gammas.					

8. SENSITIVITY CASES

In addition to the nominal cases presented in Sections 3 through 7, a number of sensitivity calculations were carried out. The list of sensitivity cases is given in Table 35. Except for case number 6, all sensitivity cases were performed on used fuel bundle with 290 MWh/kgU burnup and 30 years decay time (i.e., the bundle with the highest radiation source terms). For sensitivity case 6, dose rates are generated for all burnup and decay values.

Table 35: List of Sensitivity Cases

Sensitivity case no.	Geometry configuration	Change vs. nominal case	Impact being analysed	Result
1	Single fuel bundle configuration (Section 3).	Absence of fuel sheath	Maximum dose rate within fuel bundle envelope.	Section 8.1.1
2		Within-element radiation source distribution (skin effect)		Section 8.2
3		Low temperature water		Section 8.4.1
4	Single UFC configuration (Section 4)	Absence of fuel sheath	Maximum dose rate at internal UFC-water interface.	Section 8.1.2
5		Variation in water density		Section 8.4.2
6		Intact UFC		Section 8.3.1.3
		Intact UFC	Dose rates inside UFC	Section 0
		UFC filled with moist air	Dose rates inside UFC	Section 8.3.1.2
7	Single UFC configuration (Section 4)	Low temperature water	Maximum dose rate at external UFC-water interface	Section 8.4.2
8	Single UFC configuration submerged in moist air (Section 5)	Intact UFC	Maximum dose rate at external UFC-water interface	Section 8.3.2
9	Placement room configuration (Section 6)	Low temperature water	Maximum dose rate at internal UFC-water interface.	Section 8.4.3
10		Sedimentary rock		Section 8.5
11		Sedimentary rock	Maximum dose rate at external UFC-water interface	Section 8.5
12		Intact UFC		Section 8.3.3
13		Low temperature water		Section 8.4.3
14	Placement room configuration with moist air (Section 7)	Sedimentary rock	Maximum dose rate at external UFC-water interface	Section 8.5

8.1 Absence of Fuel Sheath

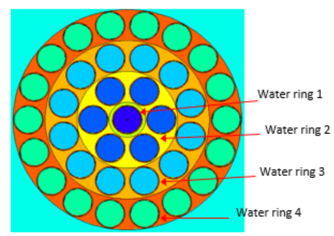
In this scenario, the space occupied by zircaloy was replaced with water. The sensitivity was performed on single fuel bundle and single UFC configurations.

8.1.1 Single Fuel Bundle – Absence of Fuel Sheath

In the nominal case (Section 3), the maximum dose rate occurs in water occupying the gap between the fuel element and the fuel sheath. The principal source of dose rate in that location is the alpha energy deposition. Therefore, the impact of replacing zircaloy fuel sheath with water on the maximum dose rate is negligible. The maximum dose rate listed in Table 14 apply to both with and without fuel sheath configurations.

The impact of the fuel sheath removal is greater in the interstitial water between the fuel elements in the bundle. When the fuel sheath is removed, the beta energy deposition increases by a factor of four – see Table 36. The increase in gamma dose rate is about 4%.

Table 36: Dose Rates Within Fuel Bundle Envelope: Sensitivity Case 1



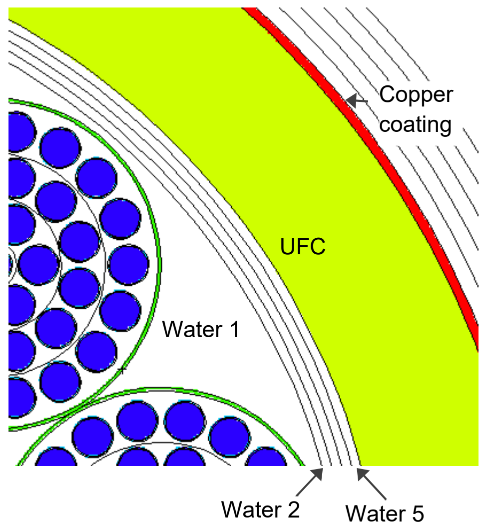
Water region	Gy/h, nominal case				
	Beta	Gamma	Neutron	Total	
Water ring 1	1.41E+01	5.40E+01	3.36E-05	6.81E+01	
Water ring 2	1.18E+01	5.47E+01	3.29E-05	6.65E+01	
Water ring 3	1.25E+01	5.62E+01	3.24E-05	6.87E+01	
Water ring 4	1.17E+01	4.59E+01	2.86E-05	5.77E+01	
Water region	**Gy/h, sensitivity case**				
	Beta	**Gamma**	**Neutron**	**Total**	**Ratio to nominal**
Water ring 1	5.98E+01	5.63E+01	3.25E-05	1.16E+02	1.7
Water ring 2	4.80E+01	5.68E+01	3.20E-05	1.05E+02	1.6
Water ring 3	5.10E+01	5.83E+01	3.16E-05	1.09E+02	1.6
Water ring 4	5.02E+01	4.76E+01	2.79E-05	9.78E+01	1.7
Notes: Dose rates shown above include contributions from electrons, bremsstrahlung gammas, gammas, neutrons, and neutron-capture gammas. The values correspond to a fuel bundle with 290 MWh/kgU burnup and 30 years decay time.					

8.1.2 Single UFC – Absence of Fuel Sheath

The dose rates in water inside the UFC are listed in Table 37. The gamma dose rates increase by about 4%, and the neutron dose rates increase by 30% to 50%, with the fuel sheath removed. However, since neutron dose rates are insignificant compared to the gamma dose rates, the overall dose rates in water inside the UFC are dominated by gamma and increase only by 4% when the zircaloy components of the fuel bundle are replaced with water.

Table 37: Dose Rates inside UFC: Sensitivity Case 4

Water Region	Gy/h, nominal case		
	Gamma	Neutron	Total
Water 1	3.18E+01	3.26E-05	3.18E+01
Water 2	2.12E+01	2.17E-05	2.12E+01
Water 3	2.02E+01	2.09E-05	2.02E+01
Water 4	1.93E+01	2.02E-05	1.93E+01
Water 5	1.85E+01	1.98E-05	1.85E+01



Water Region	Gy/h, sensitivity case			
	Gamma	Neutron	Total	Ratio to nominal
Water 1	3.32E+01	3.16E-05	3.32E+01	1.04
Water 2	2.21E+01	2.12E-05	2.21E+01	1.04
Water 3	2.11E+01	2.03E-05	2.11E+01	1.04
Water 4	2.01E+01	1.97E-05	2.01E+01	1.04
Water 5	1.93E+01	1.93E-05	1.93E+01	1.04

8.2 Skin Effect

In the nominal case, the source term distribution within bundles was accounted for (i.e., rings of elements). However, within each element, the source term distribution was assumed to be uniform. In reality, there is a sharp peak in the power density and burnup at the radial periphery of the fuel element (commonly called the “skin effect”). The skin effect would result in higher radiation source emissions at the radial periphery of the fuel element.

For the sensitivity case study, the profile of radiation source emission within elements is set to the profile shown in Figure 25. The profile is similar to the heat generation rate in Figure 18 of Tayal and Gacesa 2015.

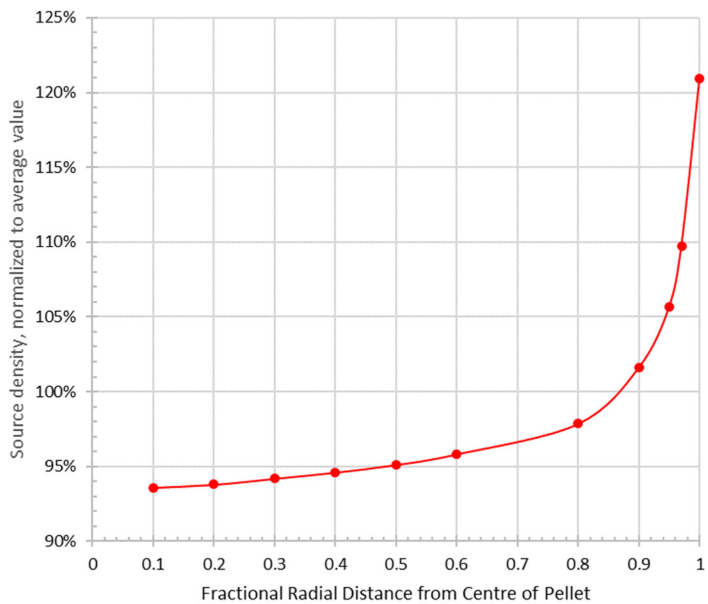


Figure 25: Radiation Source Profile for Skin Effect Sensitivity Case

The impact of the skin effect is more pronounced in the alpha and beta dose rate calculations due to the limited range of these particles. For alpha dose rate calculations, only alpha particles that are generated within the 20 μm from the fuel-water interface contribute to the dose rates in water. For a fuel element of 0.61 cm radius with a uniform alpha source distribution, 99.3% of alpha particles generated in the fuel element do not reach the fuel-water interface. Based on the profile shown in Figure 25, there are 21% more alpha particles generated within the 20 μm outermost region of the fuel element compared to that of uniform distribution. Thus, when the skin effect is accounted for, the dose rates listed in Table 10 would increase by 21%.

For beta dose rate calculations, the maximum dose rate at the fuel-water interface inside the fuel sheath increases by approximately 19%, from 216 Gy/h (Figure 12) to 257 Gy/h when the skin-effect source distribution is applied. The maximum dose rates in water outside of fuel sheaths increases by 17%, from 14.1 Gy/h (Figure 12) to 16.5 Gy/h.

For gamma and neutron dose rate calculations, the dose rates at the fuel-water interface would increase by a maximum of 2% when the within-element gamma or neutron source distribution follows the profile in Figure 25 instead of a uniform distribution.

Discussions above are summarized in Table 38.

Table 38: Maximum Dose Rates at Fuel-Water Interface (Single Fuel Bundle): Sensitivity Case 2

Radiation Source	Without Skin Effect	With Skin Effect	Notes
Alpha	2.23E+02	2.70E+02	21% increase
Beta	2.13E+02	2.54E+02	19% increase
Gamma	4.92E+01	4.99E+01	2% increase
Neutron	3.09E-05	3.15E-05	2% increase
Total	4.86E+02	5.74E+02	18% increase
Note: values are for used fuel bundle with 290 MWh/kgU burnup at 30 years decay time.			

8.3 Intact UFC

Unless otherwise noted, for this scenario, the UFC is intact and the free space inside the UFC is filled with dry air instead of water.

8.3.1 Single Intact UFC Submerged in Water

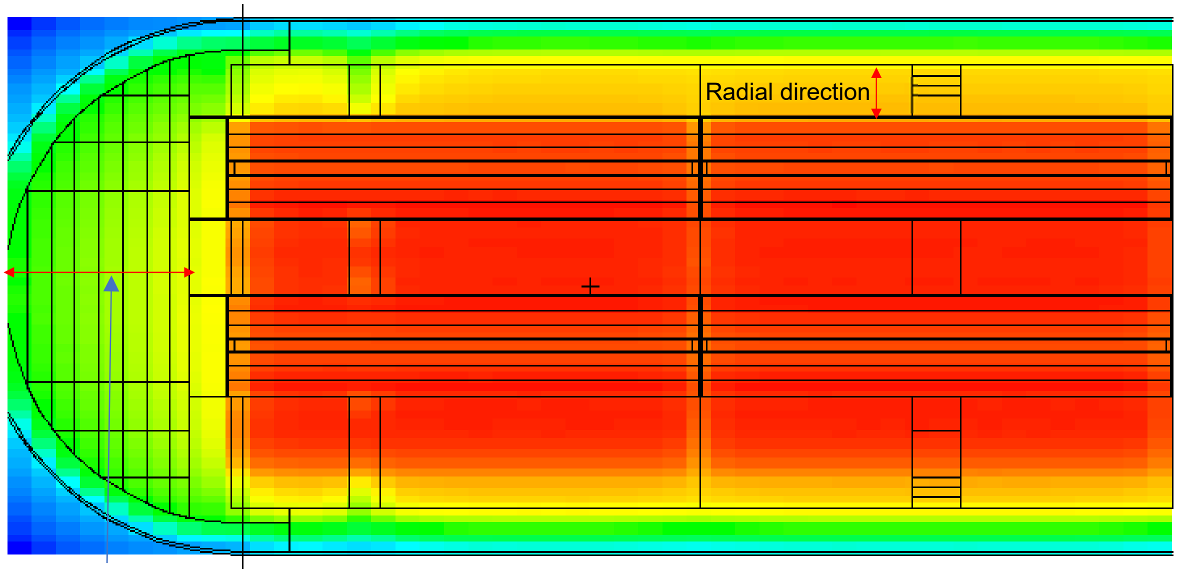
This sensitivity case involves an intact UFC submerged in water. For the internal UFC dose rate calculations, two cases were completed: one with the UFC filled with dry air (Section 0) and another one with the UFC filled with moist air (Section 8.3.1.2). The external dose rate calculations (Section 8.3.1.3) were performed with the UFC filled with dry air.

8.3.1.1 Internal UFC (Dry Air)

Dose rates inside the UFC were calculated at different distances from the interior surface of the UFC. Dose rates at the vicinity of the UFC interior surfaces are dominated by the gamma radiation. The general dose rate profile and the description on where dose rates are tallied are shown in Figure 26. The calculated dose rates are tabulated in Table 39.

8.3.1.2 Internal UFC (Moist Air)

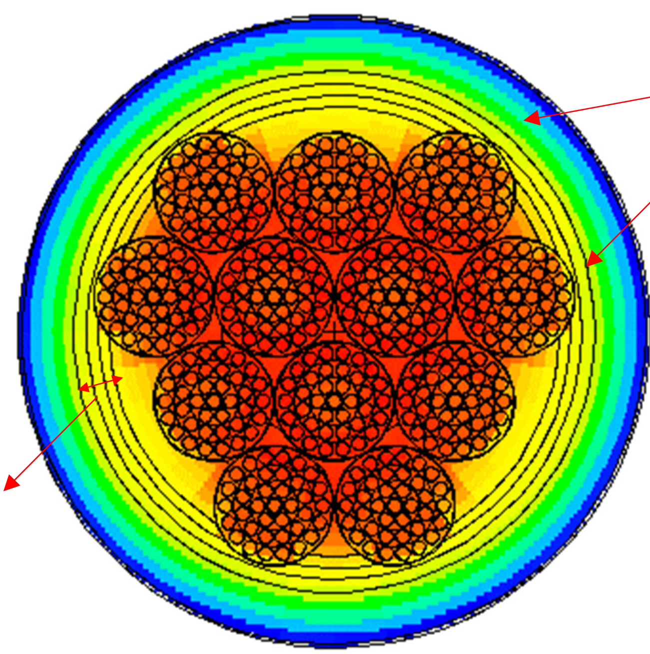
This case is identical to the one in Section 0, except for the cavity inside the UFC is now filled with moist air instead of dry air. The calculated dose rates are tabulated in Table 40. The absorbed dose rates in the moist air are approximately five percent higher than the dose rates in dry air (Section 0).



Range of dose rates in axial direction

Top figure: side view

Dose rates in axial direction are given for eight distances from the UFC interior surface: at 1.2, 3.65, 6.15, 8.65, 11.15, 13.65, 16.15, 18.4 cm from the surface.



UFC vessel

UFC vessel interior surface

Range of dose rates in radial direction

Bottom figure: section view

Dose rates in radial direction are given for three distances from the UFC interior surface: at 0.5875, 1.675, 1.175 cm from the surface.

Figure 26: Illustration of Dose Rate Profile Inside an Intact UFC

Table 39: Dose Rates (Gy/h) inside UFC Filled with Dry Air

220 MWh/kgU, 30 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.23E+01	3.81E-06	4.80E-02	2.24E+01
	1.675	2.08E+01	3.66E-06	4.80E-02	2.08E+01
	0.5875	1.92E+01	3.54E-06	4.80E-02	1.93E+01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.17E+01	1.87E-06	8.23E-01	1.25E+01
	16.15	1.02E+01	1.65E-06	8.23E-01	1.10E+01
	13.65	8.69E+00	1.46E-06	8.23E-01	9.51E+00
	11.15	7.49E+00	1.32E-06	8.23E-01	8.31E+00
	8.65	6.50E+00	1.21E-06	8.23E-01	7.32E+00
	6.15	5.70E+00	1.12E-06	8.23E-01	6.53E+00
	3.65	5.08E+00	1.06E-06	8.23E-01	5.90E+00
	1.2	4.73E+00	1.04E-06	8.23E-01	5.55E+00
220 MWh/kgU, 100 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.37E+00	2.33E-06	6.77E-03	4.38E+00
	1.675	4.07E+00	2.24E-06	6.77E-03	4.08E+00
	0.5875	3.77E+00	2.17E-06	6.77E-03	3.77E+00
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.30E+00	1.15E-06	1.49E-01	2.44E+00
	16.15	1.99E+00	1.02E-06	1.49E-01	2.14E+00
	13.65	1.70E+00	9.02E-07	1.49E-01	1.85E+00
	11.15	1.47E+00	8.13E-07	1.49E-01	1.62E+00
	8.65	1.28E+00	7.44E-07	1.49E-01	1.42E+00
	6.15	1.12E+00	6.91E-07	1.49E-01	1.27E+00
	3.65	9.97E-01	6.55E-07	1.49E-01	1.15E+00
	1.2	9.28E-01	6.39E-07	1.49E-01	1.08E+00
220 MWh/kgU, 200 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.41E-01	2.08E-06	4.54E-04	4.42E-01
	1.675	4.11E-01	2.00E-06	4.54E-04	4.11E-01
	0.5875	3.80E-01	1.94E-06	4.54E-04	3.80E-01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.32E-01	1.03E-06	1.26E-02	2.44E-01
	16.15	2.01E-01	9.08E-07	1.26E-02	2.14E-01
	13.65	1.72E-01	8.05E-07	1.26E-02	1.85E-01
	11.15	1.48E-01	7.26E-07	1.26E-02	1.61E-01
	8.65	1.29E-01	6.64E-07	1.26E-02	1.41E-01
	6.15	1.13E-01	6.17E-07	1.26E-02	1.25E-01
	3.65	1.01E-01	5.85E-07	1.26E-02	1.13E-01
	1.2	9.37E-02	5.71E-07	1.26E-02	1.06E-01
220 MWh/kgU, 300 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.65E-02	1.97E-06	4.41E-05	4.65E-02
	1.675	4.33E-02	1.89E-06	4.41E-05	4.34E-02
	0.5875	4.01E-02	1.83E-06	4.41E-05	4.01E-02

Axial	Distance from UFC interior surface, hemispherical head (cm)	Gamma	Neutron	Beta	Total
	18.4	2.45E-02	9.71E-07	1.31E-03	2.58E-02
	16.15	2.12E-02	8.58E-07	1.31E-03	2.25E-02
	13.65	1.81E-02	7.62E-07	1.31E-03	1.94E-02
	11.15	1.56E-02	6.86E-07	1.31E-03	1.69E-02
	8.65	1.36E-02	6.28E-07	1.31E-03	1.49E-02
	6.15	1.19E-02	5.83E-07	1.31E-03	1.32E-02
	3.65	1.06E-02	5.52E-07	1.31E-03	1.19E-02
	1.2	9.84E-03	5.41E-07	1.31E-03	1.11E-02
220 MWh/kgU, 500 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.49E-03	1.80E-06	7.83E-07	1.49E-03
	1.675	1.39E-03	1.73E-06	7.83E-07	1.39E-03
	0.5875	1.29E-03	1.68E-06	7.83E-07	1.29E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	7.19E-04	8.87E-07	3.10E-05	7.50E-04
	16.15	6.23E-04	7.86E-07	3.10E-05	6.55E-04
	13.65	5.31E-04	6.97E-07	3.10E-05	5.63E-04
	11.15	4.58E-04	6.29E-07	3.10E-05	4.90E-04
	8.65	3.97E-04	5.76E-07	3.10E-05	4.28E-04
	6.15	3.46E-04	5.35E-07	3.10E-05	3.77E-04
	3.65	3.09E-04	5.08E-07	3.10E-05	3.40E-04
	1.2	2.88E-04	4.96E-07	3.10E-05	3.20E-04
220 MWh/kgU, 1E3 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.24E-04	1.54E-06	8.49E-07	6.26E-04
	1.675	5.82E-04	1.48E-06	8.49E-07	5.85E-04
	0.5875	5.40E-04	1.43E-06	8.49E-07	5.43E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.82E-04	7.59E-07	1.97E-05	3.02E-04
	16.15	2.45E-04	6.73E-07	1.97E-05	2.66E-04
	13.65	2.11E-04	5.97E-07	1.97E-05	2.31E-04
	11.15	1.83E-04	5.39E-07	1.97E-05	2.03E-04
	8.65	1.59E-04	4.93E-07	1.97E-05	1.80E-04
	6.15	1.40E-04	4.59E-07	1.97E-05	1.60E-04
	3.65	1.26E-04	4.35E-07	1.97E-05	1.46E-04
	1.2	1.17E-04	4.25E-07	1.97E-05	1.38E-04
220 MWh/kgU, 1E4 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.27E-04	6.66E-07	2.47E-07	6.28E-04
	1.675	5.84E-04	6.40E-07	2.47E-07	5.85E-04
	0.5875	5.41E-04	6.20E-07	2.47E-07	5.42E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.26E-04	3.28E-07	2.20E-05	3.48E-04
	16.15	2.82E-04	2.91E-07	2.20E-05	3.04E-04
	13.65	2.41E-04	2.58E-07	2.20E-05	2.63E-04
	11.15	2.08E-04	2.33E-07	2.20E-05	2.30E-04
	8.65	1.80E-04	2.13E-07	2.20E-05	2.02E-04
	6.15	1.58E-04	1.98E-07	2.20E-05	1.80E-04
	3.65	1.40E-04	1.88E-07	2.20E-05	1.62E-04

	1.2	1.30E-04	1.84E-07	2.20E-05	1.52E-04
220 MWh/kgU, 1E5 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.51E-03	1.40E-07	1.73E-06	1.51E-03
	1.675	1.40E-03	1.35E-07	1.73E-06	1.40E-03
	0.5875	1.29E-03	1.30E-07	1.73E-06	1.29E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	8.49E-04	6.87E-08	2.99E-05	8.79E-04
	16.15	7.28E-04	6.07E-08	2.99E-05	7.58E-04
	13.65	6.17E-04	5.40E-08	2.99E-05	6.47E-04
	11.15	5.27E-04	4.87E-08	2.99E-05	5.57E-04
	8.65	4.55E-04	4.46E-08	2.99E-05	4.85E-04
	6.15	3.97E-04	4.15E-08	2.99E-05	4.27E-04
	3.65	3.51E-04	3.95E-08	2.99E-05	3.81E-04
	1.2	3.24E-04	3.85E-08	2.99E-05	3.54E-04
220 MWh/kgU, 1E6 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.23E-03	3.63E-08	4.25E-06	2.23E-03
	1.675	2.06E-03	3.49E-08	4.25E-06	2.07E-03
	0.5875	1.90E-03	3.38E-08	4.25E-06	1.91E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.28E-03	1.79E-08	3.71E-05	1.31E-03
	16.15	1.09E-03	1.58E-08	3.71E-05	1.13E-03
	13.65	9.23E-04	1.41E-08	3.71E-05	9.61E-04
	11.15	7.88E-04	1.27E-08	3.71E-05	8.25E-04
	8.65	6.78E-04	1.17E-08	3.71E-05	7.15E-04
	6.15	5.90E-04	1.09E-08	3.71E-05	6.27E-04
	3.65	5.21E-04	1.03E-08	3.71E-05	5.58E-04
	1.2	4.80E-04	1.01E-08	3.71E-05	5.17E-04
220 MWh/kgU, 1E7 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.07E-03	1.21E-08	2.65E-06	2.08E-03
	1.675	1.92E-03	1.16E-08	2.65E-06	1.93E-03
	0.5875	1.77E-03	1.13E-08	2.65E-06	1.77E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.20E-03	6.01E-09	3.17E-05	1.23E-03
	16.15	1.02E-03	5.34E-09	3.17E-05	1.06E-03
	13.65	8.66E-04	4.75E-09	3.17E-05	8.97E-04
	11.15	7.38E-04	4.30E-09	3.17E-05	7.70E-04
	8.65	6.35E-04	3.95E-09	3.17E-05	6.67E-04
	6.15	5.52E-04	3.67E-09	3.17E-05	5.84E-04
	3.65	4.87E-04	3.49E-09	3.17E-05	5.19E-04
	1.2	4.49E-04	3.42E-09	3.17E-05	4.81E-04
290 MWh/kgU, 030 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.93E+01	9.05E-06	3.35E-02	2.94E+01
	1.675	2.73E+01	8.70E-06	3.35E-02	2.73E+01
	0.5875	2.53E+01	8.41E-06	3.35E-02	2.53E+01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.54E+01	4.44E-06	9.63E-01	1.64E+01

	16.15	1.34E+01	3.92E-06	9.63E-01	1.43E+01
	13.65	1.15E+01	3.48E-06	9.63E-01	1.24E+01
	11.15	9.87E+00	3.13E-06	9.63E-01	1.08E+01
	8.65	8.57E+00	2.86E-06	9.63E-01	9.53E+00
	6.15	7.52E+00	2.65E-06	9.63E-01	8.48E+00
	3.65	6.70E+00	2.52E-06	9.63E-01	7.66E+00
	1.2	6.24E+00	2.45E-06	9.63E-01	7.20E+00
290 MWh/kgU, 100 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	5.74E+00	3.49E-06	5.58E-03	5.74E+00
	1.675	5.34E+00	3.35E-06	5.58E-03	5.34E+00
	0.5875	4.94E+00	3.24E-06	5.58E-03	4.95E+00
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.02E+00	1.72E-06	1.63E-01	3.18E+00
	16.15	2.62E+00	1.52E-06	1.63E-01	2.78E+00
	13.65	2.24E+00	1.35E-06	1.63E-01	2.40E+00
	11.15	1.93E+00	1.21E-06	1.63E-01	2.09E+00
	8.65	1.68E+00	1.11E-06	1.63E-01	1.84E+00
	6.15	1.47E+00	1.03E-06	1.63E-01	1.63E+00
	3.65	1.31E+00	9.79E-07	1.63E-01	1.47E+00
	1.2	1.22E+00	9.55E-07	1.63E-01	1.38E+00
290 MWh/kgU, 200 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	5.78E-01	2.88E-06	5.64E-04	5.78E-01
	1.675	5.38E-01	2.77E-06	5.64E-04	5.38E-01
	0.5875	4.97E-01	2.68E-06	5.64E-04	4.98E-01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.04E-01	1.42E-06	1.54E-02	3.19E-01
	16.15	2.64E-01	1.26E-06	1.54E-02	2.79E-01
	13.65	2.25E-01	1.11E-06	1.54E-02	2.41E-01
	11.15	1.94E-01	1.01E-06	1.54E-02	2.10E-01
	8.65	1.69E-01	9.20E-07	1.54E-02	1.84E-01
	6.15	1.48E-01	8.55E-07	1.54E-02	1.63E-01
	3.65	1.32E-01	8.11E-07	1.54E-02	1.47E-01
	1.2	1.23E-01	7.92E-07	1.54E-02	1.38E-01
290 MWh/kgU, 300 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.07E-02	2.71E-06	5.44E-05	6.07E-02
	1.675	5.65E-02	2.61E-06	5.44E-05	5.66E-02
	0.5875	5.23E-02	2.53E-06	5.44E-05	5.24E-02
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.19E-02	1.34E-06	1.56E-03	3.35E-02
	16.15	2.76E-02	1.19E-06	1.56E-03	2.92E-02
	13.65	2.37E-02	1.05E-06	1.56E-03	2.52E-02
	11.15	2.04E-02	9.47E-07	1.56E-03	2.20E-02
	8.65	1.77E-02	8.67E-07	1.56E-03	1.93E-02
	6.15	1.55E-02	8.06E-07	1.56E-03	1.71E-02
	3.65	1.38E-02	7.65E-07	1.56E-03	1.54E-02
	1.2	1.28E-02	7.46E-07	1.56E-03	1.44E-02
290 MWh/kgU, 500 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**

	1.175	2.03E-03	2.48E-06	1.54E-07	2.04E-03
	1.675	1.89E-03	2.38E-06	1.54E-07	1.90E-03
	0.5875	1.76E-03	2.31E-06	1.54E-07	1.76E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	9.59E-04	1.22E-06	3.57E-05	9.96E-04
	16.15	8.33E-04	1.08E-06	3.57E-05	8.70E-04
	13.65	7.13E-04	9.61E-07	3.57E-05	7.50E-04
	11.15	6.16E-04	8.67E-07	3.57E-05	6.53E-04
	8.65	5.36E-04	7.93E-07	3.57E-05	5.72E-04
	6.15	4.68E-04	7.37E-07	3.57E-05	5.04E-04
	3.65	4.18E-04	6.97E-07	3.57E-05	4.54E-04
	1.2	3.91E-04	6.81E-07	3.57E-05	4.28E-04
290 MWh/kgU, 1E3 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	9.56E-04	2.13E-06	2.35E-07	9.59E-04
	1.675	8.92E-04	2.04E-06	2.35E-07	8.95E-04
	0.5875	8.28E-04	1.98E-06	2.35E-07	8.30E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	4.21E-04	1.05E-06	2.26E-05	4.44E-04
	16.15	3.67E-04	9.28E-07	2.26E-05	3.91E-04
	13.65	3.16E-04	8.24E-07	2.26E-05	3.40E-04
	11.15	2.75E-04	7.44E-07	2.26E-05	2.98E-04
	8.65	2.39E-04	6.81E-07	2.26E-05	2.62E-04
	6.15	2.09E-04	6.32E-07	2.26E-05	2.33E-04
	3.65	1.89E-04	6.00E-07	2.26E-05	2.12E-04
	1.2	1.75E-04	5.85E-07	2.26E-05	1.98E-04
290 MWh/kgU, 1E4 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	8.51E-04	9.52E-07	7.26E-07	8.53E-04
	1.675	7.93E-04	9.16E-07	7.26E-07	7.95E-04
	0.5875	7.35E-04	8.87E-07	7.26E-07	7.37E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	4.33E-04	4.70E-07	1.93E-05	4.53E-04
	16.15	3.75E-04	4.16E-07	1.93E-05	3.95E-04
	13.65	3.21E-04	3.69E-07	1.93E-05	3.41E-04
	11.15	2.77E-04	3.33E-07	1.93E-05	2.97E-04
	8.65	2.41E-04	3.05E-07	1.93E-05	2.60E-04
	6.15	2.11E-04	2.84E-07	1.93E-05	2.31E-04
	3.65	1.88E-04	2.69E-07	1.93E-05	2.07E-04
	1.2	1.74E-04	2.63E-07	1.93E-05	1.94E-04
290 MWh/kgU, 1E5 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.67E-03	2.59E-07	1.11E-06	1.67E-03
	1.675	1.55E-03	2.49E-07	1.11E-06	1.55E-03
	0.5875	1.43E-03	2.41E-07	1.11E-06	1.43E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	9.36E-04	1.27E-07	3.01E-05	9.66E-04
	16.15	8.03E-04	1.12E-07	3.01E-05	8.33E-04
	13.65	6.81E-04	9.98E-08	3.01E-05	7.11E-04
	11.15	5.83E-04	9.01E-08	3.01E-05	6.13E-04

	8.65	5.03E-04	8.27E-08	3.01E-05	5.33E-04
	6.15	4.38E-04	7.70E-08	3.01E-05	4.69E-04
	3.65	3.88E-04	7.30E-08	3.01E-05	4.18E-04
	1.2	3.58E-04	7.13E-08	3.01E-05	3.88E-04
290 MWh/kgU, 1E6 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.27E-03	5.90E-08	4.29E-06	2.28E-03
	1.675	2.11E-03	5.68E-08	4.29E-06	2.11E-03
	0.5875	1.94E-03	5.49E-08	4.29E-06	1.94E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.30E-03	2.90E-08	3.28E-05	1.33E-03
	16.15	1.11E-03	2.56E-08	3.28E-05	1.14E-03
	13.65	9.41E-04	2.28E-08	3.28E-05	9.73E-04
	11.15	8.03E-04	2.06E-08	3.28E-05	8.35E-04
	8.65	6.91E-04	1.89E-08	3.28E-05	7.24E-04
	6.15	6.01E-04	1.75E-08	3.28E-05	6.34E-04
	3.65	5.31E-04	1.67E-08	3.28E-05	5.64E-04
	1.2	4.89E-04	1.63E-08	3.28E-05	5.22E-04
290 MWh/kgU, 1E7 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.07E-03	1.21E-08	3.13E-06	2.07E-03
	1.675	1.92E-03	1.16E-08	3.13E-06	1.92E-03
	0.5875	1.77E-03	1.13E-08	3.13E-06	1.77E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.20E-03	6.00E-09	3.08E-05	1.23E-03
	16.15	1.02E-03	5.33E-09	3.08E-05	1.05E-03
	13.65	8.64E-04	4.75E-09	3.08E-05	8.95E-04
	11.15	7.37E-04	4.29E-09	3.08E-05	7.68E-04
	8.65	6.34E-04	3.94E-09	3.08E-05	6.65E-04
	6.15	5.51E-04	3.67E-09	3.08E-05	5.82E-04
	3.65	4.86E-04	3.49E-09	3.08E-05	5.17E-04
	1.2	4.48E-04	3.41E-09	3.08E-05	4.79E-04

Table 40: Dose Rates (Gy/h) inside UFC Filled with Moist Air

220 MWh/kgU, 30 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.33E+01	1.70E-05	2.67E-02	2.34E+01
	1.675	2.17E+01	1.64E-05	2.67E-02	2.17E+01
	0.5875	2.01E+01	1.60E-05	2.67E-02	2.01E+01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.23E+01	7.58E-06	8.57E-01	1.31E+01
	16.15	1.06E+01	6.71E-06	8.57E-01	1.15E+01
	13.65	9.09E+00	5.95E-06	8.57E-01	9.95E+00
	11.15	7.83E+00	5.38E-06	8.57E-01	8.69E+00
	8.65	6.80E+00	4.94E-06	8.57E-01	7.66E+00
	6.15	5.97E+00	4.63E-06	8.57E-01	6.83E+00
	3.65	5.32E+00	4.45E-06	8.57E-01	6.18E+00
	1.2	4.95E+00	4.39E-06	8.57E-01	5.81E+00
220 MWh/kgU, 100 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.58E+00	1.04E-05	4.94E-03	4.58E+00
	1.675	4.26E+00	1.01E-05	4.94E-03	4.27E+00
	0.5875	3.94E+00	9.84E-06	4.94E-03	3.95E+00
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.40E+00	4.69E-06	1.57E-01	2.56E+00
	16.15	2.08E+00	4.14E-06	1.57E-01	2.24E+00
	13.65	1.78E+00	3.68E-06	1.57E-01	1.94E+00
	11.15	1.54E+00	3.33E-06	1.57E-01	1.69E+00
	8.65	1.33E+00	3.06E-06	1.57E-01	1.49E+00
	6.15	1.17E+00	2.86E-06	1.57E-01	1.33E+00
	3.65	1.04E+00	2.75E-06	1.57E-01	1.20E+00
	1.2	9.72E-01	2.72E-06	1.57E-01	1.13E+00
220 MWh/kgU, 200 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.62E-01	9.35E-06	6.65E-04	4.63E-01
	1.675	4.30E-01	9.02E-06	6.65E-04	4.30E-01
	0.5875	3.98E-01	8.80E-06	6.65E-04	3.98E-01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.43E-01	4.20E-06	1.41E-02	2.57E-01
	16.15	2.10E-01	3.71E-06	1.41E-02	2.24E-01
	13.65	1.80E-01	3.30E-06	1.41E-02	1.94E-01
	11.15	1.55E-01	2.98E-06	1.41E-02	1.69E-01
	8.65	1.35E-01	2.74E-06	1.41E-02	1.49E-01
	6.15	1.18E-01	2.57E-06	1.41E-02	1.32E-01
	3.65	1.05E-01	2.46E-06	1.41E-02	1.19E-01
	1.2	9.80E-02	2.43E-06	1.41E-02	1.12E-01
220 MWh/kgU, 300 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	4.86E-02	8.84E-06	4.52E-05	4.87E-02
	1.675	4.53E-02	8.55E-06	4.52E-05	4.54E-02
	0.5875	4.19E-02	8.33E-06	4.52E-05	4.20E-02
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**

	18.4	2.56E-02	3.97E-06	1.46E-03	2.71E-02
	16.15	2.22E-02	3.51E-06	1.46E-03	2.36E-02
	13.65	1.90E-02	3.12E-06	1.46E-03	2.04E-02
	11.15	1.64E-02	2.82E-06	1.46E-03	1.78E-02
	8.65	1.42E-02	2.59E-06	1.46E-03	1.57E-02
	6.15	1.25E-02	2.43E-06	1.46E-03	1.39E-02
	3.65	1.11E-02	2.33E-06	1.46E-03	1.26E-02
	1.2	1.03E-02	2.31E-06	1.46E-03	1.18E-02
220 MWh/kgU, 500 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.56E-03	8.13E-06	1.03E-06	1.57E-03
	1.675	1.45E-03	7.85E-06	1.03E-06	1.46E-03
	0.5875	1.35E-03	7.65E-06	1.03E-06	1.35E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	7.52E-04	3.64E-06	3.59E-05	7.91E-04
	16.15	6.52E-04	3.22E-06	3.59E-05	6.91E-04
	13.65	5.56E-04	2.86E-06	3.59E-05	5.95E-04
	11.15	4.79E-04	2.59E-06	3.59E-05	5.18E-04
	8.65	4.15E-04	2.38E-06	3.59E-05	4.54E-04
	6.15	3.62E-04	2.23E-06	3.59E-05	4.00E-04
	3.65	3.23E-04	2.14E-06	3.59E-05	3.61E-04
	1.2	3.02E-04	2.12E-06	3.59E-05	3.40E-04
220 MWh/kgU, 1E3 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.52E-04	7.00E-06	4.43E-07	6.59E-04
	1.675	6.09E-04	6.76E-06	4.43E-07	6.16E-04
	0.5875	5.65E-04	6.59E-06	4.43E-07	5.72E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	2.94E-04	3.14E-06	2.78E-05	3.25E-04
	16.15	2.57E-04	2.77E-06	2.78E-05	2.87E-04
	13.65	2.21E-04	2.46E-06	2.78E-05	2.51E-04
	11.15	1.92E-04	2.23E-06	2.78E-05	2.22E-04
	8.65	1.67E-04	2.05E-06	2.78E-05	1.97E-04
	6.15	1.47E-04	1.92E-06	2.78E-05	1.76E-04
	3.65	1.32E-04	1.84E-06	2.78E-05	1.62E-04
	1.2	1.23E-04	1.82E-06	2.78E-05	1.52E-04
220 MWh/kgU, 1E4 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.56E-04	3.04E-06	7.11E-07	6.59E-04
	1.675	6.11E-04	2.93E-06	7.11E-07	6.15E-04
	0.5875	5.66E-04	2.86E-06	7.11E-07	5.70E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.41E-04	1.36E-06	2.57E-05	3.68E-04
	16.15	2.95E-04	1.20E-06	2.57E-05	3.22E-04
	13.65	2.52E-04	1.07E-06	2.57E-05	2.79E-04
	11.15	2.17E-04	9.67E-07	2.57E-05	2.44E-04
	8.65	1.88E-04	8.90E-07	2.57E-05	2.15E-04
	6.15	1.65E-04	8.34E-07	2.57E-05	1.92E-04
	3.65	1.47E-04	8.00E-07	2.57E-05	1.73E-04
	1.2	1.36E-04	7.90E-07	2.57E-05	1.63E-04
220 MWh/kgU, 1E5 years decay					

Radial	Distance from UFC interior surface (cm)	Gamma	Neutron	Beta	Total
	1.175	1.58E-03	6.38E-07	1.90E-06	1.58E-03
	1.675	1.46E-03	6.16E-07	1.90E-06	1.46E-03
	0.5875	1.35E-03	6.00E-07	1.90E-06	1.35E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	8.88E-04	2.85E-07	3.48E-05	9.23E-04
	16.15	7.61E-04	2.51E-07	3.48E-05	7.96E-04
	13.65	6.45E-04	2.23E-07	3.48E-05	6.80E-04
	11.15	5.52E-04	2.02E-07	3.48E-05	5.87E-04
	8.65	4.76E-04	1.86E-07	3.48E-05	5.11E-04
	6.15	4.15E-04	1.74E-07	3.48E-05	4.50E-04
	3.65	3.67E-04	1.67E-07	3.48E-05	4.02E-04
	1.2	3.39E-04	1.66E-07	3.48E-05	3.74E-04
220 MWh/kgU, 1E6 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.33E-03	1.67E-07	4.26E-06	2.33E-03
	1.675	2.16E-03	1.61E-07	4.26E-06	2.16E-03
	0.5875	1.99E-03	1.57E-07	4.26E-06	1.99E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.34E-03	7.45E-08	3.73E-05	1.37E-03
	16.15	1.14E-03	6.59E-08	3.73E-05	1.18E-03
	13.65	9.66E-04	5.86E-08	3.73E-05	1.00E-03
	11.15	8.24E-04	5.30E-08	3.73E-05	8.62E-04
	8.65	7.09E-04	4.88E-08	3.73E-05	7.47E-04
	6.15	6.18E-04	4.58E-08	3.73E-05	6.55E-04
	3.65	5.45E-04	4.40E-08	3.73E-05	5.83E-04
	1.2	5.03E-04	4.34E-08	3.73E-05	5.40E-04
220 MWh/kgU, 1E7 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.17E-03	5.73E-08	2.25E-06	2.17E-03
	1.675	2.01E-03	5.53E-08	2.25E-06	2.01E-03
	0.5875	1.85E-03	5.40E-08	2.25E-06	1.85E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.25E-03	2.58E-08	4.21E-05	1.30E-03
	16.15	1.07E-03	2.28E-08	4.21E-05	1.11E-03
	13.65	9.06E-04	2.02E-08	4.21E-05	9.48E-04
	11.15	7.72E-04	1.83E-08	4.21E-05	8.14E-04
	8.65	6.64E-04	1.69E-08	4.21E-05	7.06E-04
	6.15	5.78E-04	1.58E-08	4.21E-05	6.20E-04
	3.65	5.10E-04	1.52E-08	4.21E-05	5.52E-04
	1.2	4.70E-04	1.50E-08	4.21E-05	5.12E-04
290 MWh/kgU, 030 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	3.07E+01	4.03E-05	4.88E-02	3.07E+01
	1.675	2.86E+01	3.89E-05	4.88E-02	2.86E+01
	0.5875	2.64E+01	3.79E-05	4.88E-02	2.65E+01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.62E+01	1.80E-05	1.11E+00	1.73E+01
	16.15	1.40E+01	1.59E-05	1.11E+00	1.51E+01
	13.65	1.20E+01	1.41E-05	1.11E+00	1.31E+01

	11.15	1.03E+01	1.28E-05	1.11E+00	1.14E+01
	8.65	8.97E+00	1.17E-05	1.11E+00	1.01E+01
	6.15	7.87E+00	1.10E-05	1.11E+00	8.98E+00
	3.65	7.01E+00	1.05E-05	1.11E+00	8.12E+00
	1.2	6.53E+00	1.04E-05	1.11E+00	7.64E+00
290 MWh/kgU, 100 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.00E+00	1.56E-05	9.33E-03	6.01E+00
	1.675	5.59E+00	1.51E-05	9.33E-03	5.60E+00
	0.5875	5.17E+00	1.47E-05	9.33E-03	5.18E+00
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.16E+00	7.00E-06	2.01E-01	3.36E+00
	16.15	2.74E+00	6.19E-06	2.01E-01	2.94E+00
	13.65	2.34E+00	5.49E-06	2.01E-01	2.54E+00
	11.15	2.02E+00	4.97E-06	2.01E-01	2.22E+00
	8.65	1.75E+00	4.57E-06	2.01E-01	1.95E+00
	6.15	1.54E+00	4.28E-06	2.01E-01	1.74E+00
	3.65	1.37E+00	4.11E-06	2.01E-01	1.57E+00
	1.2	1.28E+00	4.05E-06	2.01E-01	1.48E+00
290 MWh/kgU, 200 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.04E-01	1.29E-05	7.28E-04	6.05E-01
	1.675	5.62E-01	1.25E-05	7.28E-04	5.63E-01
	0.5875	5.20E-01	1.22E-05	7.28E-04	5.21E-01
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.18E-01	5.81E-06	1.94E-02	3.37E-01
	16.15	2.76E-01	5.13E-06	1.94E-02	2.95E-01
	13.65	2.36E-01	4.56E-06	1.94E-02	2.55E-01
	11.15	2.03E-01	4.12E-06	1.94E-02	2.23E-01
	8.65	1.77E-01	3.79E-06	1.94E-02	1.96E-01
	6.15	1.55E-01	3.56E-06	1.94E-02	1.74E-01
	3.65	1.38E-01	3.41E-06	1.94E-02	1.57E-01
	1.2	1.29E-01	3.37E-06	1.94E-02	1.48E-01
290 MWh/kgU, 300 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	6.35E-02	1.22E-05	5.43E-05	6.35E-02
	1.675	5.91E-02	1.18E-05	5.43E-05	5.92E-02
	0.5875	5.47E-02	1.15E-05	5.43E-05	5.48E-02
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	3.34E-02	5.48E-06	1.60E-03	3.50E-02
	16.15	2.89E-02	4.85E-06	1.60E-03	3.05E-02
	13.65	2.48E-02	4.31E-06	1.60E-03	2.64E-02
	11.15	2.14E-02	3.89E-06	1.60E-03	2.30E-02
	8.65	1.85E-02	3.58E-06	1.60E-03	2.01E-02
	6.15	1.63E-02	3.35E-06	1.60E-03	1.79E-02
	3.65	1.45E-02	3.22E-06	1.60E-03	1.61E-02
	1.2	1.35E-02	3.19E-06	1.60E-03	1.51E-02
290 MWh/kgU, 500 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.12E-03	1.12E-05	5.18E-07	2.14E-03
	1.675	1.98E-03	1.08E-05	5.18E-07	1.99E-03

	0.5875	1.84E-03	1.06E-05	5.18E-07	1.85E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.00E-03	5.02E-06	4.62E-05	1.05E-03
	16.15	8.72E-04	4.44E-06	4.62E-05	9.22E-04
	13.65	7.46E-04	3.95E-06	4.62E-05	7.96E-04
	11.15	6.45E-04	3.57E-06	4.62E-05	6.95E-04
	8.65	5.61E-04	3.28E-06	4.62E-05	6.11E-04
	6.15	4.90E-04	3.07E-06	4.62E-05	5.39E-04
	3.65	4.37E-04	2.95E-06	4.62E-05	4.86E-04
	1.2	4.10E-04	2.91E-06	4.62E-05	4.59E-04
290 MWh/kgU, 1E3 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.00E-03	9.66E-06	2.82E-07	1.01E-03
	1.675	9.32E-04	9.33E-06	2.82E-07	9.42E-04
	0.5875	8.66E-04	9.10E-06	2.82E-07	8.75E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	4.40E-04	4.33E-06	2.03E-05	4.64E-04
	16.15	3.84E-04	3.83E-06	2.03E-05	4.08E-04
	13.65	3.31E-04	3.40E-06	2.03E-05	3.55E-04
	11.15	2.88E-04	3.08E-06	2.03E-05	3.11E-04
	8.65	2.50E-04	2.83E-06	2.03E-05	2.73E-04
	6.15	2.19E-04	2.65E-06	2.03E-05	2.42E-04
	3.65	1.97E-04	2.54E-06	2.03E-05	2.20E-04
	1.2	1.83E-04	2.52E-06	2.03E-05	2.06E-04
290 MWh/kgU, 1E4 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	8.90E-04	4.35E-06	4.37E-07	8.95E-04
	1.675	8.30E-04	4.20E-06	4.37E-07	8.35E-04
	0.5875	7.69E-04	4.10E-06	4.37E-07	7.74E-04
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	4.53E-04	1.95E-06	2.71E-05	4.82E-04
	16.15	3.92E-04	1.72E-06	2.71E-05	4.21E-04
	13.65	3.36E-04	1.53E-06	2.71E-05	3.65E-04
	11.15	2.90E-04	1.38E-06	2.71E-05	3.18E-04
	8.65	2.52E-04	1.27E-06	2.71E-05	2.80E-04
	6.15	2.21E-04	1.19E-06	2.71E-05	2.49E-04
	3.65	1.97E-04	1.15E-06	2.71E-05	2.25E-04
	1.2	1.83E-04	1.13E-06	2.71E-05	2.11E-04
290 MWh/kgU, 1E5 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	1.75E-03	1.18E-06	9.45E-07	1.75E-03
	1.675	1.62E-03	1.14E-06	9.45E-07	1.63E-03
	0.5875	1.50E-03	1.11E-06	9.45E-07	1.50E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	9.79E-04	5.26E-07	3.01E-05	1.01E-03
	16.15	8.40E-04	4.65E-07	3.01E-05	8.71E-04
	13.65	7.13E-04	4.13E-07	3.01E-05	7.43E-04
	11.15	6.10E-04	3.74E-07	3.01E-05	6.40E-04
	8.65	5.26E-04	3.44E-07	3.01E-05	5.56E-04
	6.15	4.59E-04	3.23E-07	3.01E-05	4.89E-04

	3.65	4.06E-04	3.09E-07	3.01E-05	4.36E-04
	1.2	3.75E-04	3.06E-07	3.01E-05	4.05E-04
290 MWh/kgU, 1E6 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.38E-03	2.70E-07	2.90E-06	2.38E-03
	1.675	2.20E-03	2.61E-07	2.90E-06	2.21E-03
	0.5875	2.03E-03	2.54E-07	2.90E-06	2.03E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.36E-03	1.20E-07	3.83E-05	1.40E-03
	16.15	1.16E-03	1.06E-07	3.83E-05	1.20E-03
	13.65	9.84E-04	9.45E-08	3.83E-05	1.02E-03
	11.15	8.40E-04	8.55E-08	3.83E-05	8.78E-04
	8.65	7.23E-04	7.87E-08	3.83E-05	7.61E-04
	6.15	6.29E-04	7.38E-08	3.83E-05	6.68E-04
	3.65	5.55E-04	7.09E-08	3.83E-05	5.94E-04
	1.2	5.12E-04	7.01E-08	3.83E-05	5.50E-04
290 MWh/kgU, 1E7 years decay					
Radial	**Distance from UFC interior surface (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	1.175	2.17E-03	5.73E-08	2.54E-06	2.17E-03
	1.675	2.01E-03	5.52E-08	2.54E-06	2.01E-03
	0.5875	1.85E-03	5.39E-08	2.54E-06	1.85E-03
Axial	**Distance from UFC interior surface, hemispherical head (cm)**	**Gamma**	**Neutron**	**Beta**	**Total**
	18.4	1.25E-03	2.57E-08	3.84E-05	1.29E-03
	16.15	1.07E-03	2.28E-08	3.84E-05	1.11E-03
	13.65	9.04E-04	2.02E-08	3.84E-05	9.43E-04
	11.15	7.71E-04	1.83E-08	3.84E-05	8.09E-04
	8.65	6.63E-04	1.69E-08	3.84E-05	7.02E-04
	6.15	5.77E-04	1.58E-08	3.84E-05	6.15E-04
	3.65	5.09E-04	1.52E-08	3.84E-05	5.48E-04
	1.2	4.69E-04	1.50E-08	3.84E-05	5.07E-04

8.3.1.3 External UFC-Water Interface

Dose rates outside of the UFC were calculated and compared to those in Sections 4.3.3 (gamma) and 4.4.3 (neutron). Results in Figure 27 show that the dose rates on the external UCF-water interface increase by approximately 45% when the UFC is intact (i.e., filled with air, not water). The higher dose rates are due to the less attenuation inside the UFC cavity.

For the water-filled UFC case (Section 4), the maximum dose rate at the external surface of the UFC hemispherical head is significantly lower than the maximum dose rate at the external of the cylindrical body since the gamma and neutron radiation have to go through more water in axial direction towards the hemispherical head compared to the attenuation in the radial direction. For the intact UFC case, there is no water to attenuate the radiation from the bundles and the steel thickness in the hemispherical head is less than in the cylindrical body. The gamma flux profile in and around the UFC (Figure 28) show that the maximum dose rate at the hemispherical head of the UFC would be similar to that of the cylindrical body. Results in Table 41 confirm that the maximum dose rate occurs at the radial direction.

As mentioned previously, unless otherwise noted, all calculations were based on the reference used fuel bundle (regular 37-element fuel bundle design). This fuel bundle design represents the majority of the used CANDU fuel inventory to be stored in the DGR. As demonstrated in

Appendix C, the 28-element fuel bundle design, can potentially yield higher dose rates at the external UFC-water interface. As such, if the subsequent analysis utilizing the dose rate information is related to a single or limiting UFC, the use of the calculated dose rates for the 28-element fuel bundle design is recommended to be used instead of the regular 37-element fuel bundle design. However, if the analysis involves a general storage configuration, the dose rates associated with the reference fuel bundle (regular 37-element fuel bundle design) can be applied based on the bundle abundance.

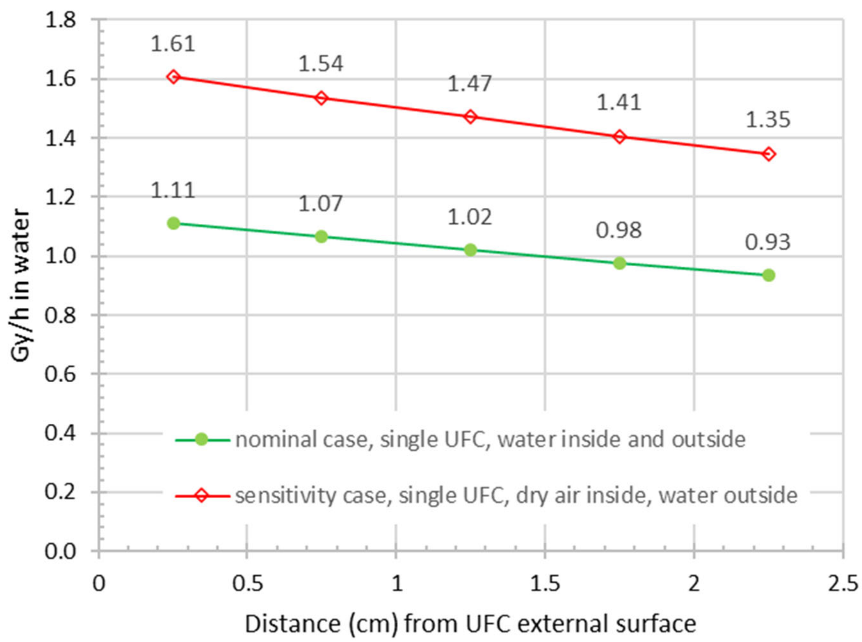


Figure 27: Dose Rates in Water Outside UFC (Sensitivity Case 6)

Notes: Dose rates shown above include contributions from gammas, neutrons, and neutron-capture gammas. Plotted values correspond to a UFC filled with used fuel bundles with 290 MWh/kgU and 30 years decay time.

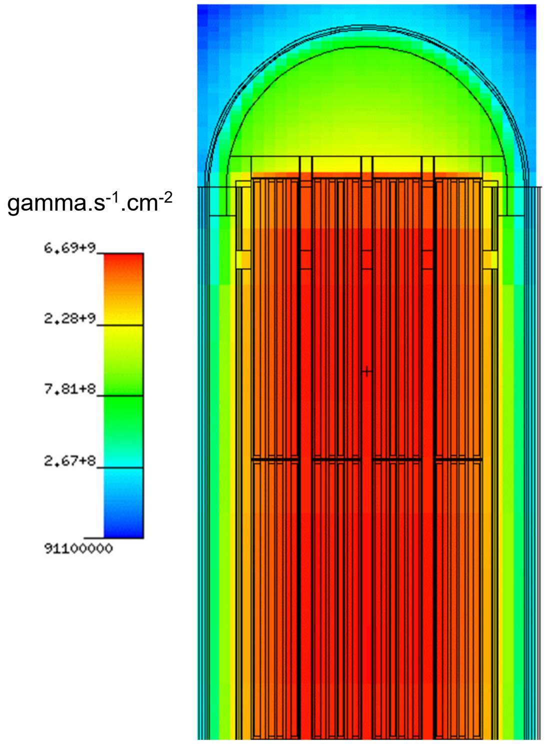


Figure 28: Gamma Flux Profile in and around a UFC (Sensitivity Case 6)

Notes: The profile corresponds to a UFC filled fuel bundles with 290 MWh/kgU and 30 years decay time

Table 41: Maximum Dose Rates at External UFC-Water Interface: Sensitivity Case 6

Burnup	Decay Time (years)	Maximum Gy/h at External UFC-Water Interface	
		Radial (UFC Cylindrical Body)	Axial (UFC Hemispherical Head)
220 MWh/kgU	30	1.21E+00	1.14E+00
	100	2.37E-01	2.23E-01
	200	2.43E-02	2.26E-02
	300	2.70E-03	2.42E-03
	500	1.03E-04	8.12E-05
	1E+03	3.26E-05	2.77E-05
	1E+04	4.13E-05	3.40E-05
	1E+05	1.38E-04	1.04E-04
	1E+06	2.26E-04	1.64E-04
	1E+07	2.15E-04	1.55E-04
290 MWh/kgU	30	1.60E+00	1.50E+00
	100	3.10E-01	2.94E-01
	200	3.17E-02	2.96E-02
	300	3.48E-03	3.19E-03
	500	1.29E-04	1.04E-04
	1E+03	4.57E-05	3.93E-05
	1E+04	5.19E-05	4.32E-05
	1E+05	1.49E-04	1.13E-04
	1E+06	2.29E-04	1.67E-04
	1E+07	2.15E-04	1.55E-04

8.3.2 Single Intact UFC Surrounded by Moist Air

The intact UFC (filled with dry air) is surrounded by moist air with 70% relative humidity. Dose rates outside of the UFC were calculated and compared against the results in Section 5 (UFC filled with moist air). Results shown in Figure 29 show nearly identical dose rates outside of the UFC.

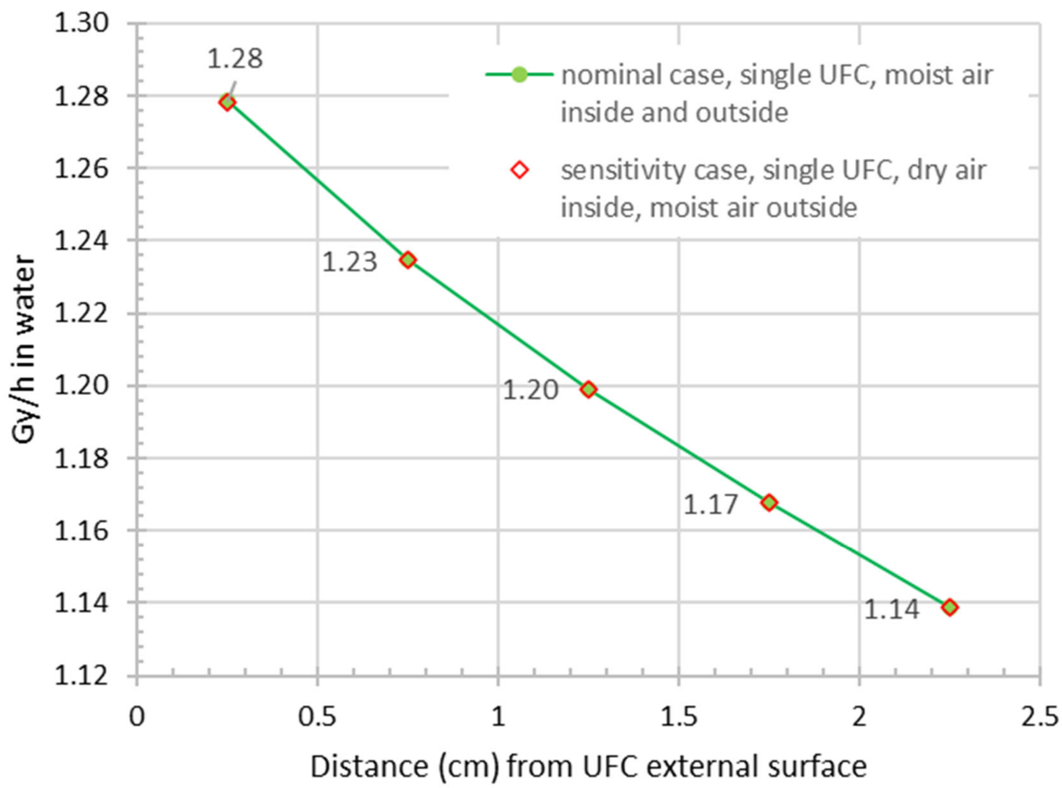


Figure 29: Dose Rates in Moist Air outside UFC (Sensitivity Case 8)

Notes: Dose rates shown above include contributions from gammas, neutrons, and neutron-capture gammas. Plotted values correspond to a UFC filled with used fuel bundles with 290 MWh/kgU and 30 years decay time.

8.3.3 Intact UFC in Placement Room

The intact UFCs are placed in bentonite boxes and arranged in a crystalline rock placement room. The room is saturated with water, but unlike the scenario in Section 6, the UFCs have not failed. Similar to the sensitivity case result for the single UFC (Section 8.3.1), the dose rate increases by approximately 45%– see Figure 30. This increase is due to the air in the container, which attenuates radiation less effectively than water and thus leads to more energy deposition at the external UFC-water interface.

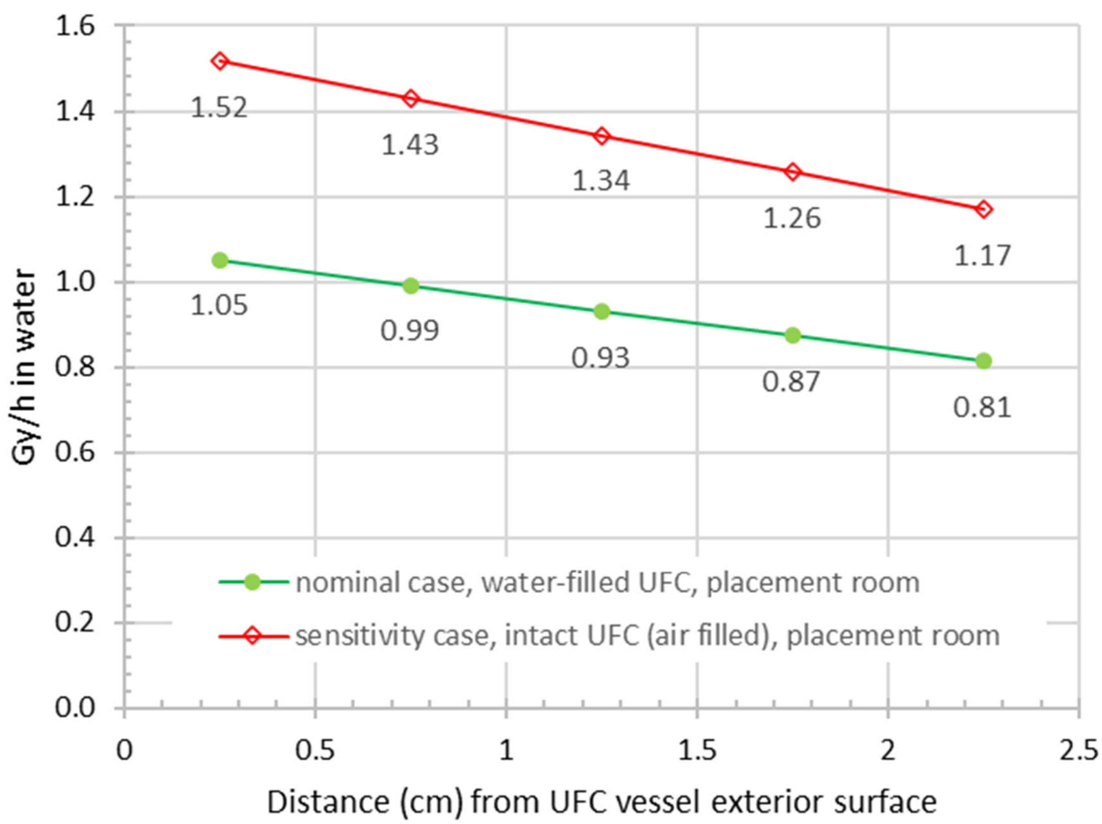


Figure 30: Dose Rates in Water Outside UFC (Sensitivity Case 12)

Notes: Dose rates shown above include contributions from gammas, neutrons, and neutron-capture gammas. Plotted values correspond to a placement room containing UFCs filled with used fuel bundles with 290 MWh/kgU and 30 years decay time.

8.4 Low Temperature Water

The nominal cases apply the water density (0.9655 g/cm^3) at the highest container surface temperature (93°C, based on Guo 2018). For the sensitivity case, the lowest temperature (10°C), which corresponds to 1.002 g/cm^3 water density, was applied. The sensitivity cases were applied to three geometry configurations: single fuel bundle, single UFC, and placement room.

8.4.1 Single Fuel Bundle in Low Temperature Water

The impact of increased water density on the maximum dose rate at the fuel-water interface was assessed for the single fuel bundle configuration. The maximum dose rates within the fuel bundle envelope, which occurs in water in the gap between the fuel element and the fuel sheath, are listed in Table 42. The higher water density results in marginally higher dose rates because more beta, gamma, and neutron energy is absorbed in the higher density water

Table 42: Maximum Dose Rates Within Fuel Bundle Envelope: Sensitivity Case 3

	Gy/h		Ratio to nominal
Radiation source	**Nominal case**	**Sensitivity case**	
Alpha	2.23E+02	2.21E+02	0.99
Beta	2.13E+02	2.19E+02	1.03
Gamma	5.70E+01	5.92E+01	1.04
Neutron	3.46E-05	3.58E-05	1.04
All sources	4.94E+02	4.99E+02	1.01

8.4.2 Single UFC in Low Temperature Water

The impact of increased water density on the maximum dose rates at UFC interface was assessed for the single UFC configuration. The dose rates are plotted in Figure 31 (internal UFC) and Figure 32 (external UFC). The higher water density results in slightly lower dose rates (around one percent lower) at the internal and external surfaces of the UFC because more gammas and neutrons are attenuated in the water that surrounds the fuel bundles (see Section 8.4.1).

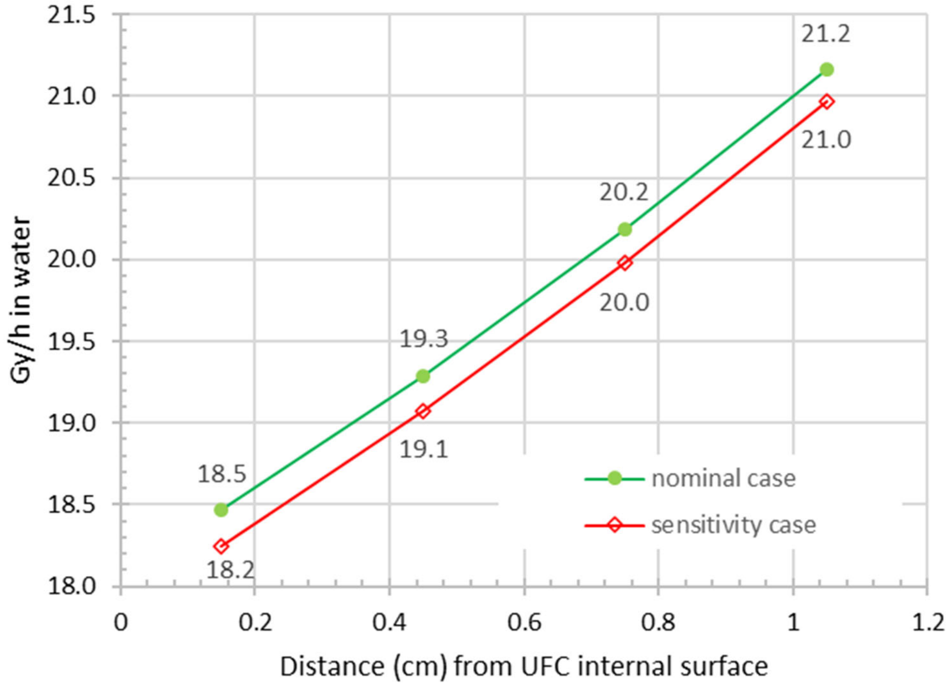


Figure 31: Dose Rates Inside UFC (Sensitivity Case 5)

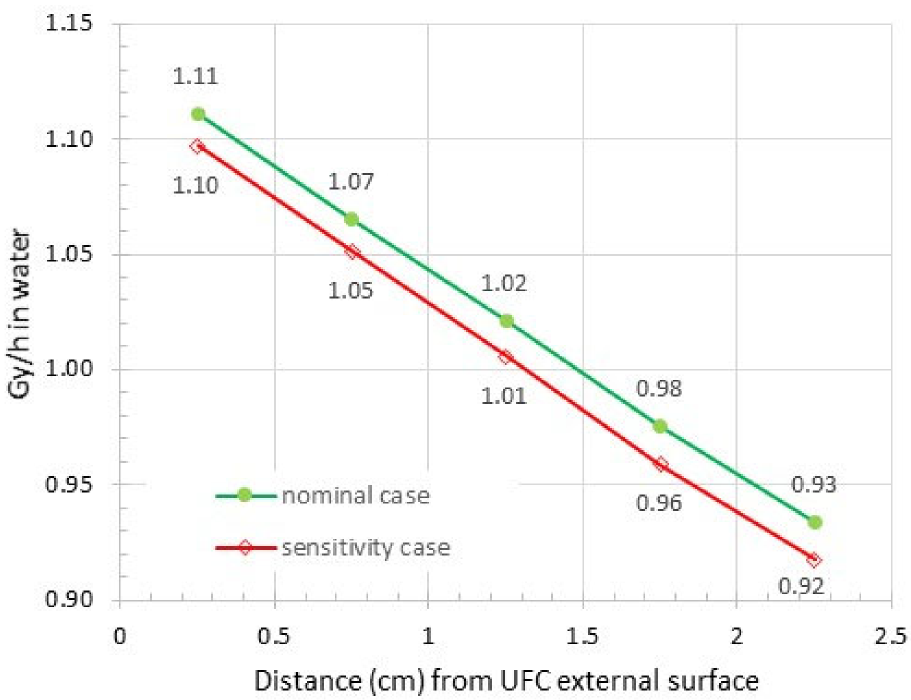


Figure 32: Dose Rates Outside UFC (Sensitivity Case 7)

8.4.3 Placement Room in Low Temperature Water

The impact of increased water density on the maximum dose rates at UFC interface in post-closure placement room configuration was assessed. The dose rates are plotted in Figure 33 (internal UFC) and Figure 34 (external UFC). The higher water density results in a small decrease (about 1%) in dose rates at the internal and external surfaces of the UFC since slightly more gammas and neutrons are attenuated in water surrounding the fuel bundles.

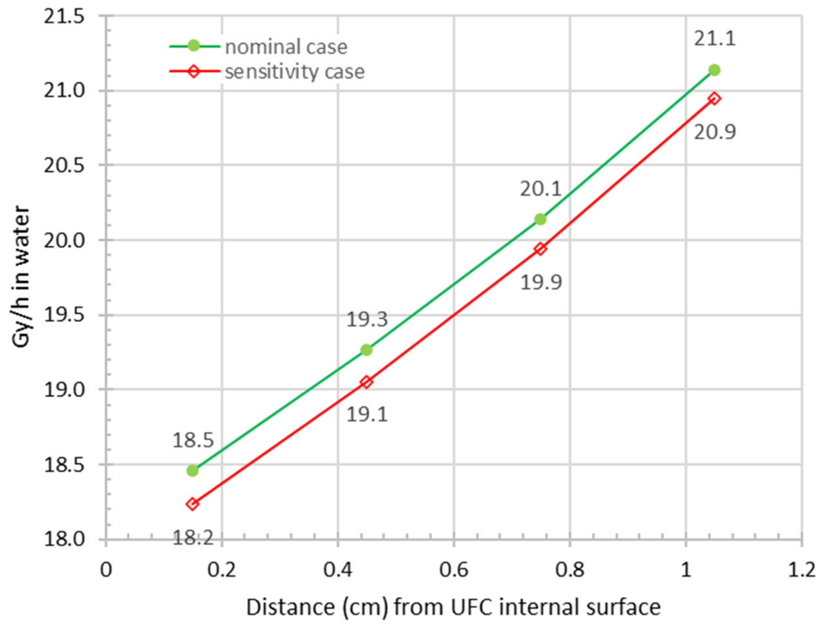


Figure 33: Dose Rates Inside UFC (Sensitivity Case 9)

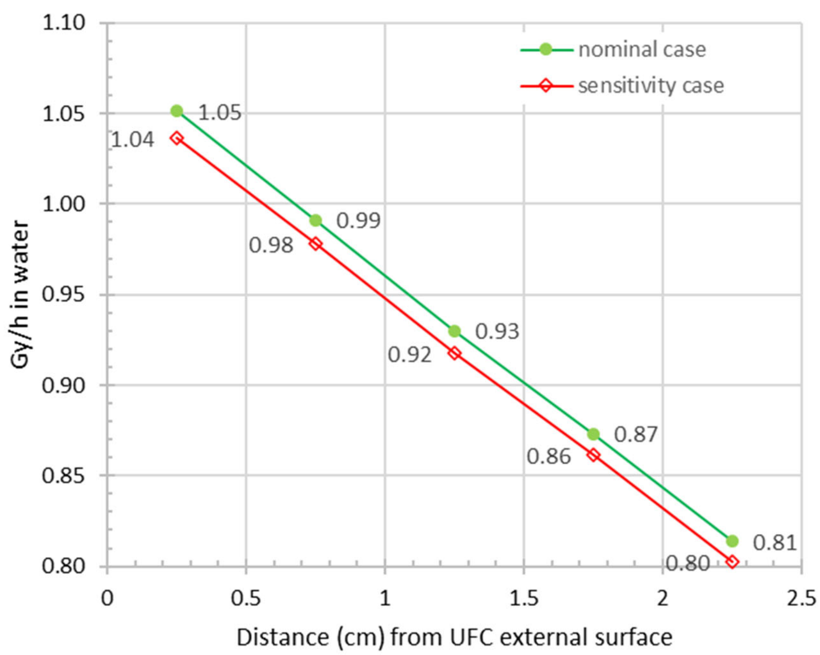


Figure 34: Dose Rates Inside UFC (Sensitivity Case 13)

8.5 Placement Room Sedimentary Rock

In this scenario, the placement room is located within a sedimentary rock instead of a crystalline rock. The UFC spacing in the sedimentary rock placement room is farther apart (70 cm) than that in crystalline rock placement room (50 cm) – see Figure A-6.

For the case where the placement room is saturated with water, the difference in UFC spacing and surrounding rock material does not change the observed dose rate at the external UFC – water interface (as shown in Figure 35). Similar results are observed for the case where the placement room is saturated with moist air(as shown in Figure 36).

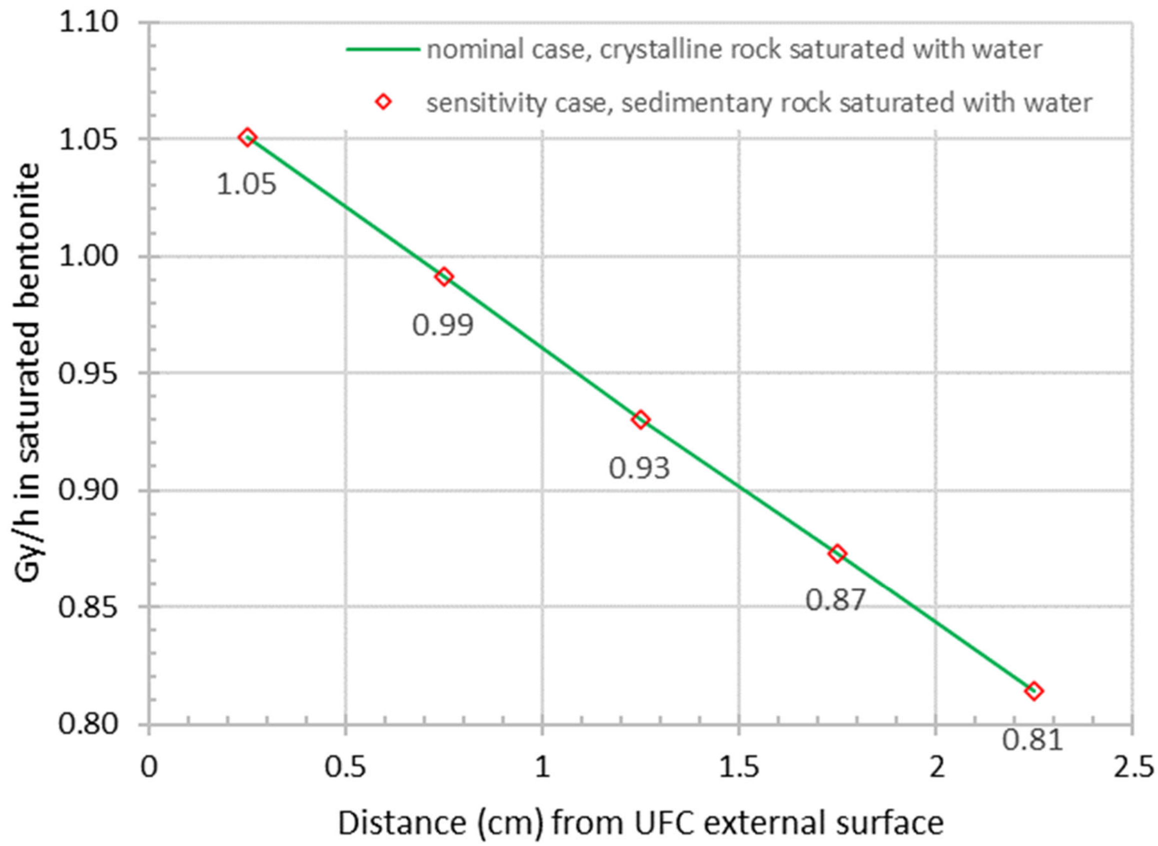


Figure 35: Dose Rates Outside UFC (Sensitivity Case 11)

Notes: Dose rates shown above include contributions from gammas, neutrons, and neutron-capture gammas. Plotted values correspond to a placement room containing UFCs filled with used fuel bundles with 290 MWh/kgU and 30 years decay time.

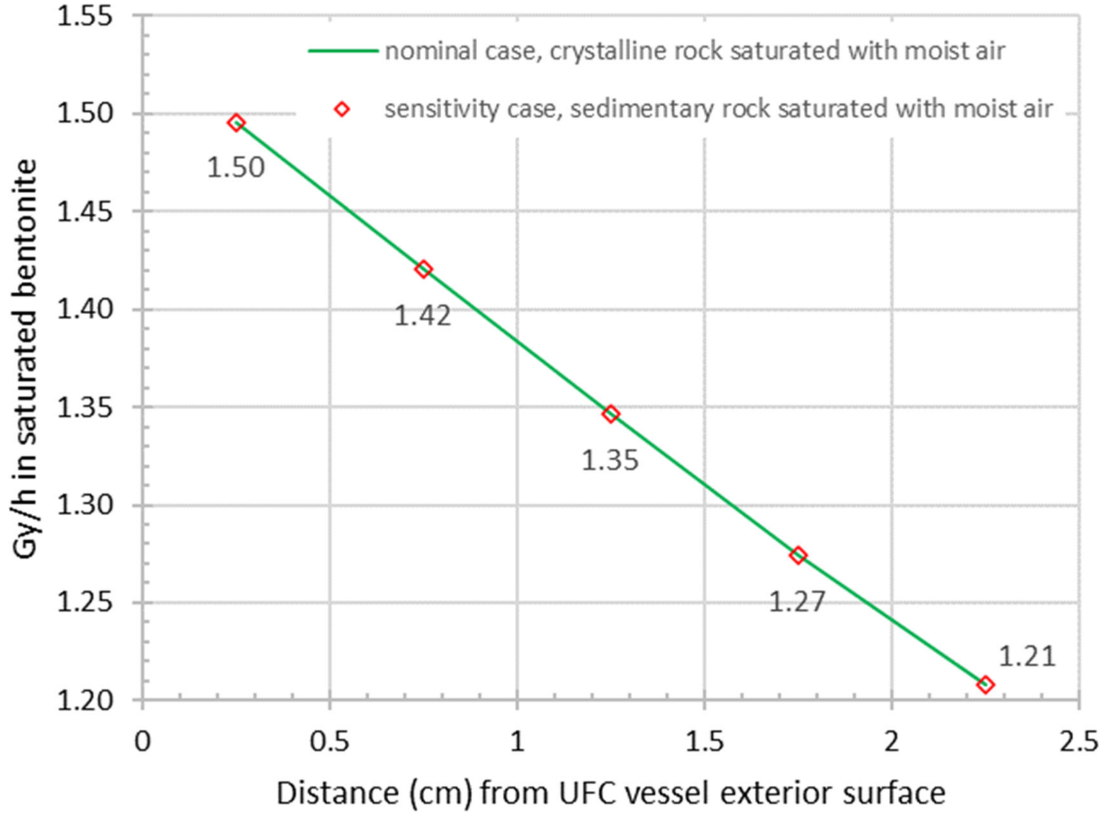


Figure 36: Dose Rates Outside UFC (Sensitivity Case 14)

Notes: Dose rates shown above include contributions from gammas, neutrons, and neutron-capture gammas. Plotted values correspond to a placement room containing UFCs filled with used fuel bundles with 290 MWh/kgU and 30 years decay time.

9. RESULT UNCERTAINTY

Contributors to uncertainties in calculated dose rates 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. As such, the radiation source term uncertainty would dominate the uncertainty of the calculated dose rates.
- Statistical uncertainty in the MCNP simulations.
  For the current analysis, the statistical uncertainties of the quoted dose rates are typically less than or around one percent. Appendix B lists the statistical uncertainties of the dose rates.
- 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. Modeling simplifications, if any, related to the modeling of UFC and placement room are also expected to have a minimal impact on the calculated dose rates.
- Uncertainty in radiation source term 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.
- Modeling assumptions (Section 2.4).
  The impact of variations in parameters for which the values were assumed in the reference case calculations has been assessed as sensitivity cases (Section 8).

Based on information above, the suggested result uncertainties for each of the fuel configurations are given in Table 43.

Table 43: Suggested Uncertainties for Dose Rates for Use in Radiolysis Assessment

Dose Rate Type	Dose Rate Value	Uncertainty of Dose Rate Value
Maximum dose rates in water in contact with fuel elements	Table 27 The maximum dose rate in a UFC configuration is essentially the same as in a placement room configuration.	98% to 142% of the calculated values Contributors: ± 20% from radiation source term uncertainty ± 0.2% MCNP statistical uncertainty + 18% from not modeling the skin effect for alpha and beta dose rates
Maximum dose rates in water at fuel bundle envelope	Table 24 (Water A) The maximum dose rate in a UFC configuration is essentially the same as in a placement room configuration.	± 20% Contributors: ± 20% from radiation source term uncertainty ± 1% MCNP statistical uncertainty
Maximum dose rates at internal UFC-water interface	Table 24 (Water B) The maximum dose rate in a UFC configuration is	± 20% Contributors: ± 20% from radiation source term uncertainty ± 1% MCNP statistical uncertainty

Dose Rate Type	Dose Rate Value	Uncertainty of Dose Rate Value
	essentially the same as in a placement room configuration.	
Maximum dose rates at external UFC-water interface	Table 41 The dose rates in a placement room configuration are less than the values in Table 41.	± 20% Contributors: ± 20% from radiation source term uncertainty ± 1% MCNP statistical uncertainty

The two burnup values used in the dose rate calculations, i.e., 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, are conservative if applied to the average used CANDU fuel.

10. COMPARISON AGAINST RESULTS FROM PREVIOUS WORK

Dose rate estimates from the current analysis are lower than those from previous work (Garisto et al. 2009). A comparison of alpha and beta dose rates is shown in Table 44. Dose rate estimates previous work are found to be higher. Alpha dose rate profiles (Figure 37) are similar, but the beta dose rate profiles (Figure 38) are different between decay times of 300 and 500 years.

The beta dose rate profile from the current work follows the beta source intensity shown in Figure 5 and the drop in the beta source average energy listed in Table 4. The sharp drop is due to a change in the principal beta sources (i.e., initially Cs-137 and Sr-90, then Tc-99 and Np-239). On the other hand, the profile from the previous work does not exhibit the same drop. The cause of the difference is unknown, but the current calculations are considered a more accurate reflection of the physical processes at play. The previous work primarily relied on hand calculation estimates using alpha and beta stopping power and estimated alpha and beta energies. In contrast, the current analysis uses more precise estimates of the alpha and beta source terms and more accurate particle transport calculations for specific geometry configurations, including explicit tracking of bremsstrahlung photons.

Gamma dose rates are compared in Table 45 and Figure 39. Results from the current analysis are higher than the previous work, except at 1E+04 years decay time. The difference can be attributed to the different used fuel container designs and changes in source terms calculations (Heckman and Edward 2020). At 1E+04 years decay time, the photon source intensity and photon emission from Heckman and Edward 2020 differ by -39.6% and -25%, respectively, when compared to the values from the reference used for Garisto et al. 2009 analysis. The difference at 1E+04 years decay time is larger than differences observed at any other decay time between 30 and 1E+07 years. This is a likely explanation for why the gamma dose rate from the current analysis is 20% lower than from the previous work.

Neutron dose rates were ignored in the previous work.

Table 44: Comparison against Previous Work: Alpha and Beta Dose Rates

Decay time (years)	Gy/a from previous work		Gy/a from current analysis	
	Alpha	Beta	Alpha	Beta
30	1.89E+06	2.20E+06	1.45E+06	1.55E+06
100	2.00E+06	4.04E+05	1.50E+06	2.90E+05
200	1.77E+06	3.96E+04	1.31E+06	2.68E+04
300	1.58E+06	6.66E+03	1.17E+06	2.60E+03
500	1.30E+06	2.69E+03	9.49E+05	1.03E+02
1E+03	9.03E+05	1.53E+03	6.39E+05	7.81E+01
1E+04	3.21E+05	3.78E+02	2.14E+05	1.01E+02
1E+05	1.80E+04	1.68E+02	1.13E+04	1.04E+02
1E+06	6.24E+03	1.49E+02	3.72E+03	9.46E+01
1E+07	4.19E+03	1.15E+02	2.32E+03	9.12E+01

Notes: Values from previous work were taken from Table 6.1 of Garisto et al. 2009. The listed dose rates are for fuel bundle with 220 MWh/kgU burnup.

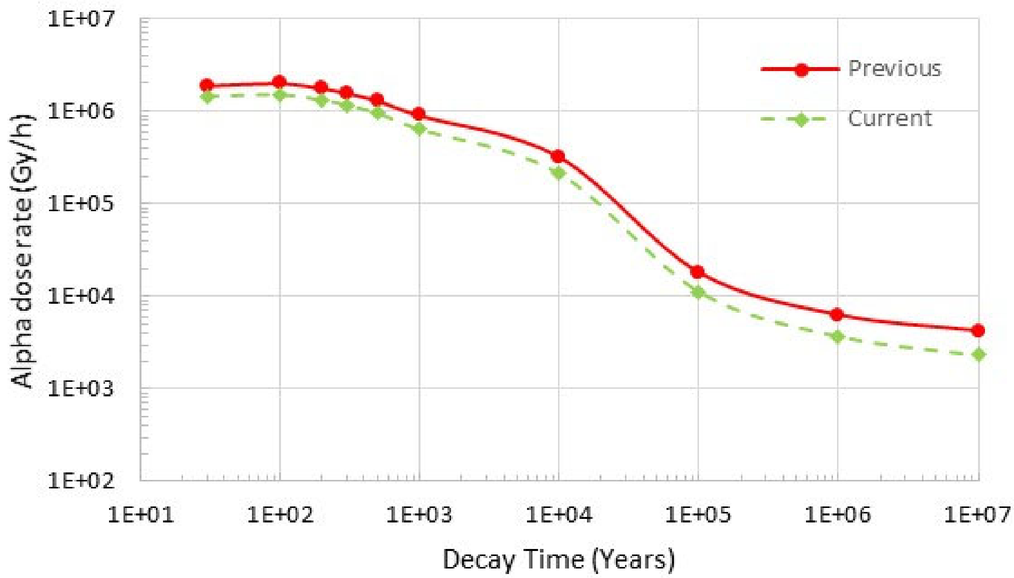


Figure 37: Comparison against Previous Work: Alpha Dose Rates

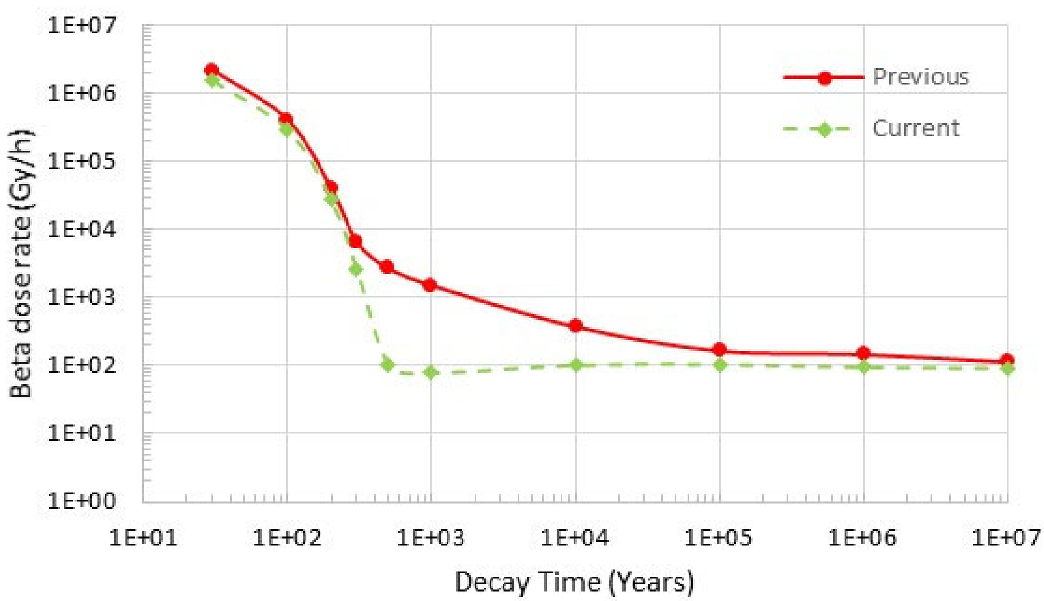


Figure 38: Comparison against Previous Work: Beta Dose Rates

Table 45: Comparison against Previous Work: Gamma Dose Rates

Decay time (years)	Previous work	Current analysis	Ratio
10	7.11E+05	n/a	n/a
50	2.20E+05	n/a	n/a
30	n/a	4.51E+05	n/a
100	5.87E+04	8.84E+04	1.51
200	6.80E+03	8.97E+03	1.32
300	n/a	9.64E+02	n/a
500	2.28E+01	4.62E+01	2.03
1E+03	1.55E+01	1.94E+01	1.25
1E+04	1.65E+01	1.33E+01	0.81
1E+05	2.84E+01	3.12E+01	1.10
1E+06	3.84E+01	4.73E+01	1.23
1E+07	3.58E+01	4.46E+01	1.25

Notes:
Values from previous work were taken from Table 6.2 of Garisto et al. 2009.
The listed dose rates are for fuel bundle with 220 MWh/kgU burnup.
Ratio is defined as the ratio of current to previous value.
Values from previous work correspond to the average dose rate in water for IV-324-HEX container.
Values from current analysis correspond to the maximum dose rate in water for single UFC configuration. The UFC design is shown in Figure A-3.

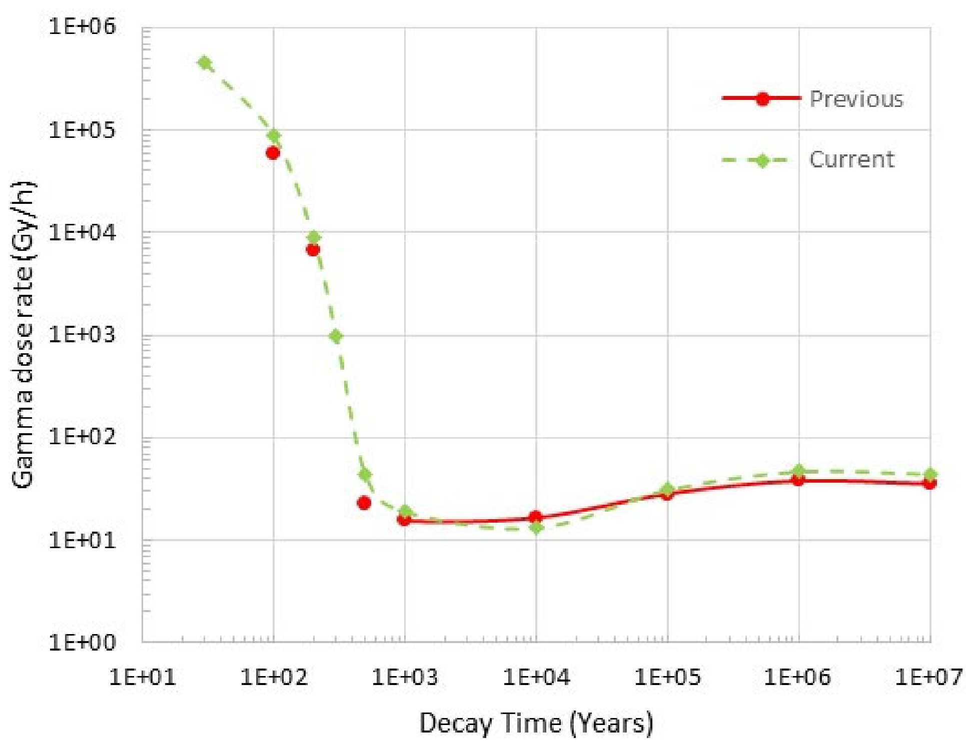


Figure 39: Comparison against Previous Work: Gamma Dose Rates

11. SUMMARY

Dose rates in water or moist air surrounding used CANDU fuel bundles and fuel containers from alpha, beta, gamma, and neutron sources in the fuel have been calculated using the state-of-the art computational codes. The work documented in the current report provides the following improvement over previous work:

- More accurate estimates alpha and beta radiation source terms.
- Use of updated radionuclide estimates based on the latest fuel irradiation/decay calculations (Heckman and Edward 2020).
- Representative source term distribution within fuel bundles. Different source intensity and spectra are applied to each of the four fuel rings (centre, inner, intermediate, and outer) based on the latest fuel irradiation/decay calculations (Heckman and Edward 2020).
- More accurate alpha and beta dose rate calculations using MCNP transport code.
- Use of the latest fuel container and placement design configuration.
- Additional dose rate calculations for higher bundle burnup (290 MWh/kgU).
- Additional dose rate calculations for 28-element fuel bundle design

Configurations involving single fuel bundle, single UFC, and multiple UFCs in a placement room configuration were analysed. Dose rates in water inside the fuel bundle envelope and at the internal and external UFC surfaces were tabulated.

Table 43 summarizes the analysis, pointing to appropriate dose rate tables within the report to estimate radiolysis at various locations (fuel surface, fuel bundle envelope, internal UFC surface, external UFC surface) and showing associated uncertainties.

The bounding configurations are as follows. All quoted dose rates were associated with used CANDU fuel with 290 MWh/kgU burnup with 30 years decay time.

- Fuel bundle envelope
    - Bounding configuration: fuel bundle submerged in water with fuel sheath failure resulting in water filling the gap between the fuel element and the fuel sheath.
    - Maximum dose rate in water inside the fuel sheath: 573 Gy/h when the skin effect source distribution is considered.
- Internal UFC interface
    - Bounding configuration: water-filled UFC submerged in water. The UFC and fuel sheath are breached resulting in water ingress into the UFC and the space between the fuel element and fuel sheath.
    - Maximum dose rate in water inside the UFC: 32 Gy/h.
- External UFC interface
    - Bounding configuration: intact UFC (air-filled) submerged in water.
    - Maximum dose rate in water at the external UFC interface: 1.6 Gy/h.

The bounding dose rates above are applicable to both reference fuel bundle and 28-element fuel bundle design.

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APPENDIX A : DETAILS ON CALCULATION MODELS

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: Reference 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

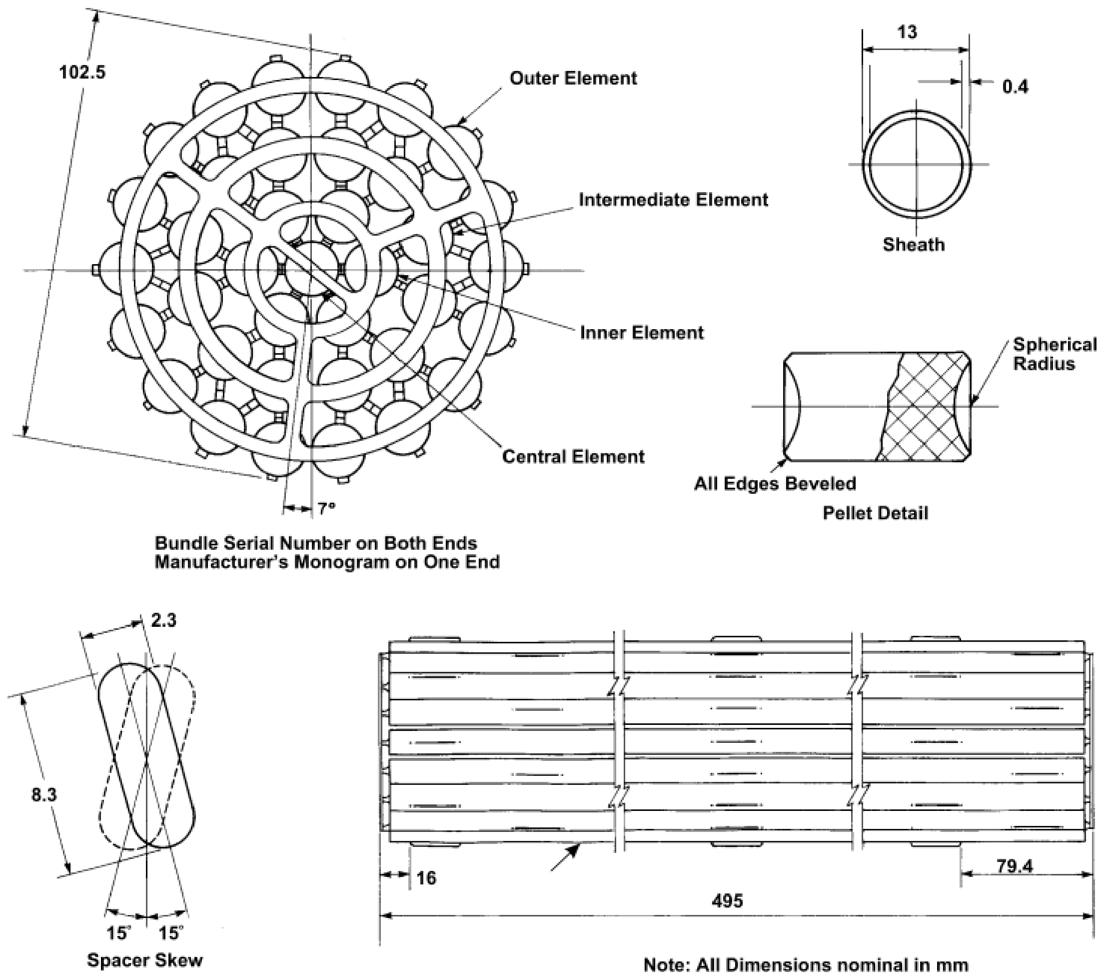

Figure A-1: Standard 37R Fuel Bundle Design

Source: Figure 2 of Heckman and Edwards 2020.

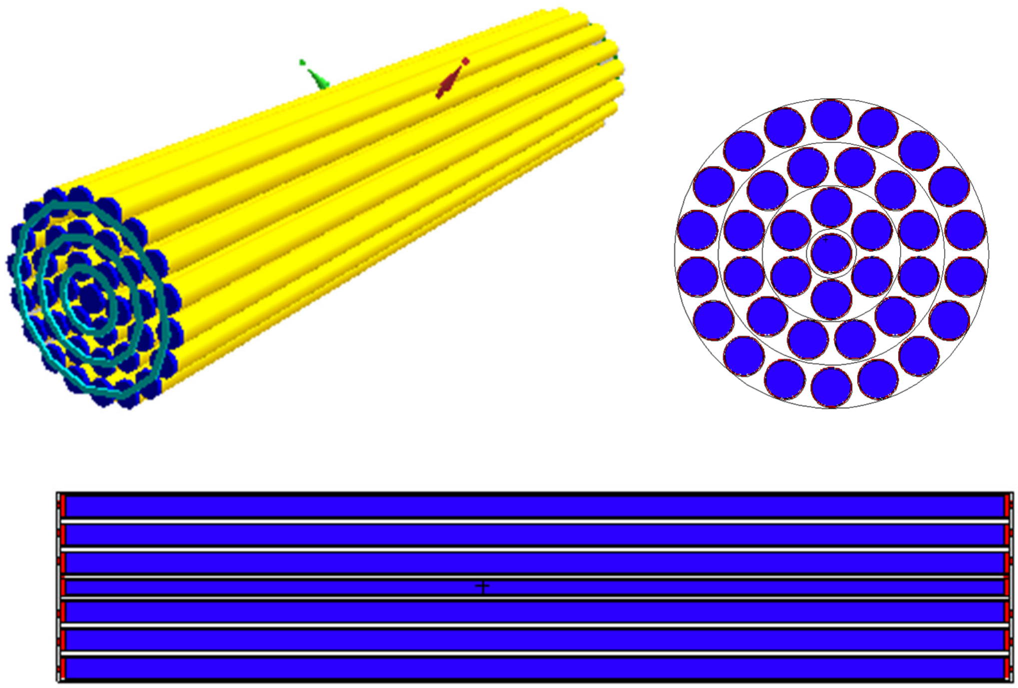

Figure A-2: MCNP Representation of Reference Used Fuel Bundle

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 standard 37R used fuel bundles. 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.

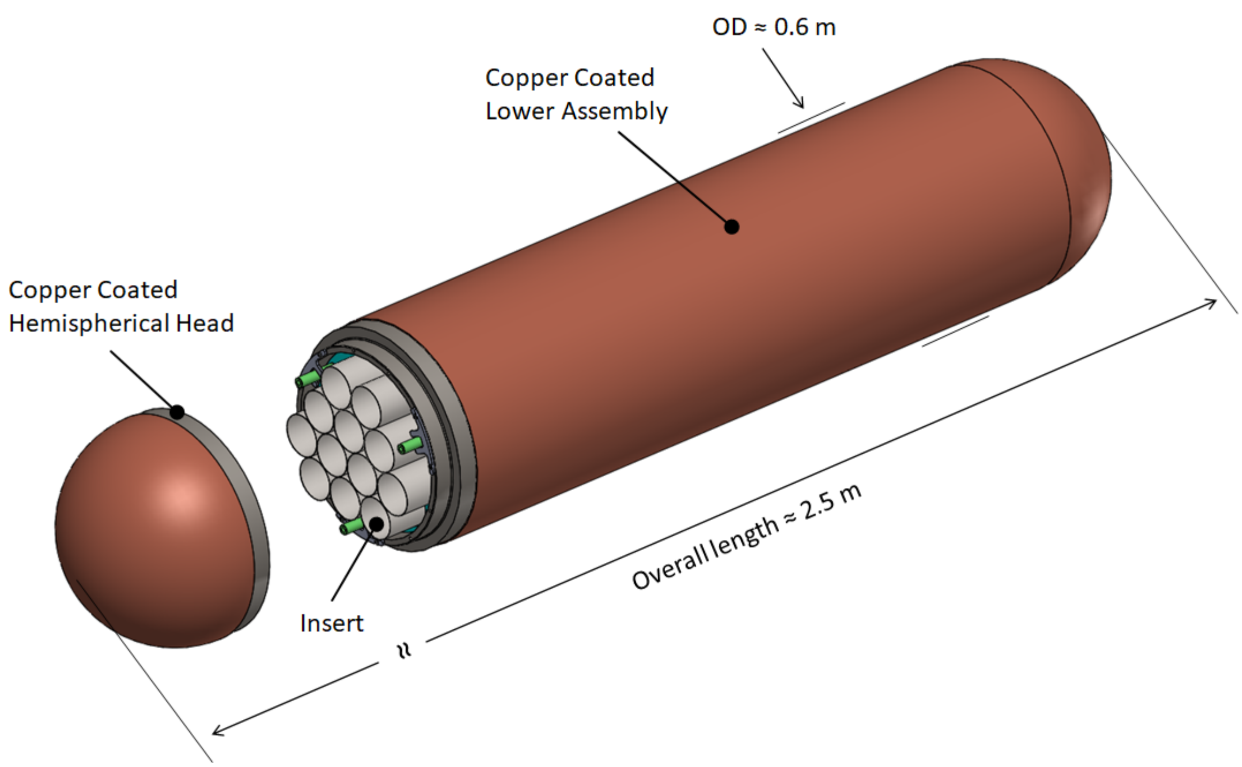

Figure A-3: Used Fuel Container

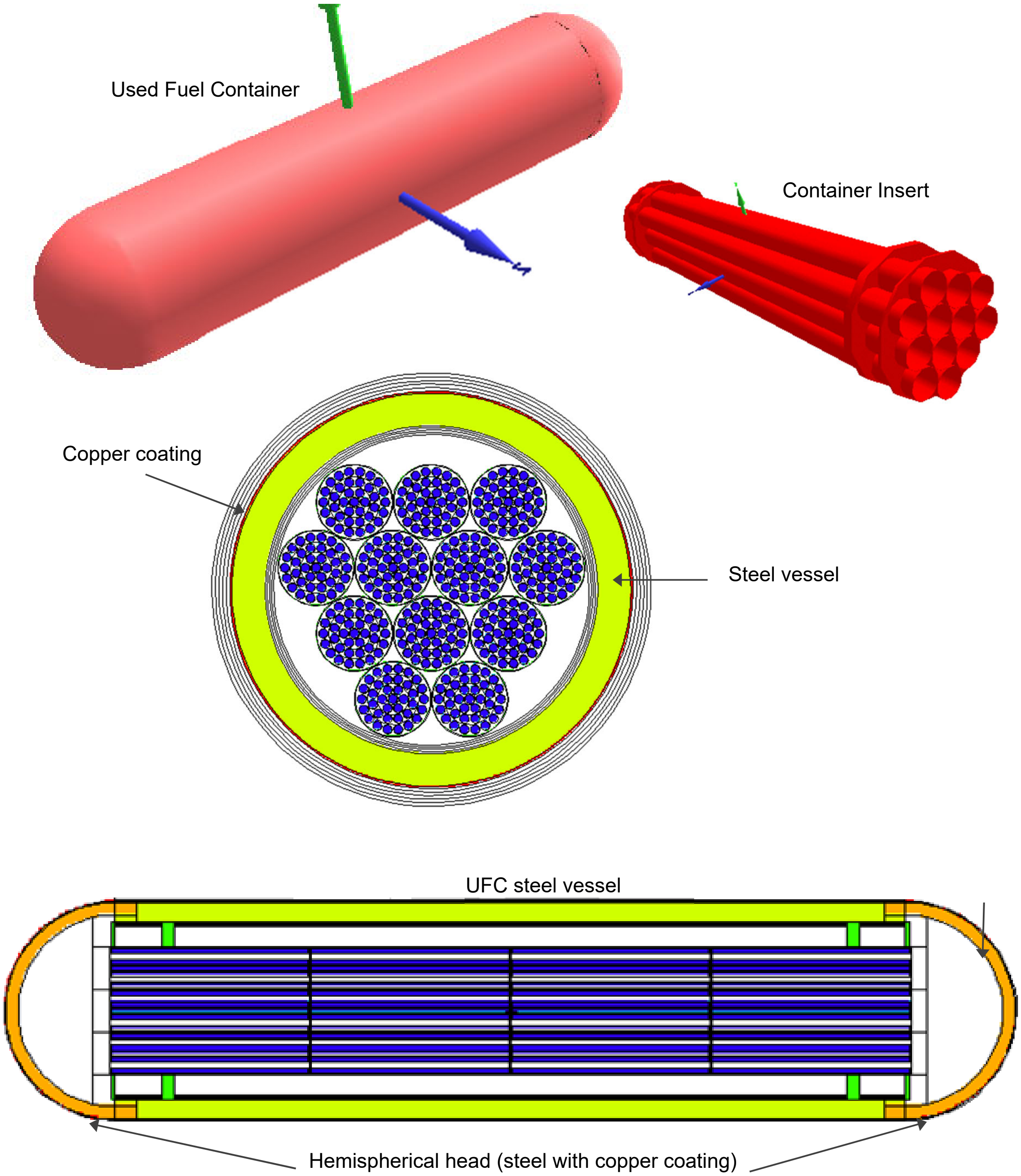

Figure A-4: MCNP Representation of Used Fuel Container and Container Insert

A.1.3 Placement Room

Each UFC will be encased in a highly compacted clay buffer box during placement in the repository. Bentonite clay is a natural material formed from volcanic ash. Bentonite is proven to be a powerful barrier to water flow. It swells when exposed to water, which makes it an excellent sealing material. Bentonite is also very stable. The buffer box dimensions (width x height x length) are 1m x 1m x 2.8 m. The geometry of the UFC storage in the placement room is shown in Figure A-6. The MCNP representation of used fuel bundles in the placement room is shown in Figure A-7.

When the placement room has achieved water saturation, bentonite in buffer boxes, spacer blocks and back fill will be homogenized and geometry details of the buffer boxes, spacer blocks, and back fill materials are no longer of importance. UFC spacing as shown in Figure A-6 still applies for the water-saturated placement room configuration.

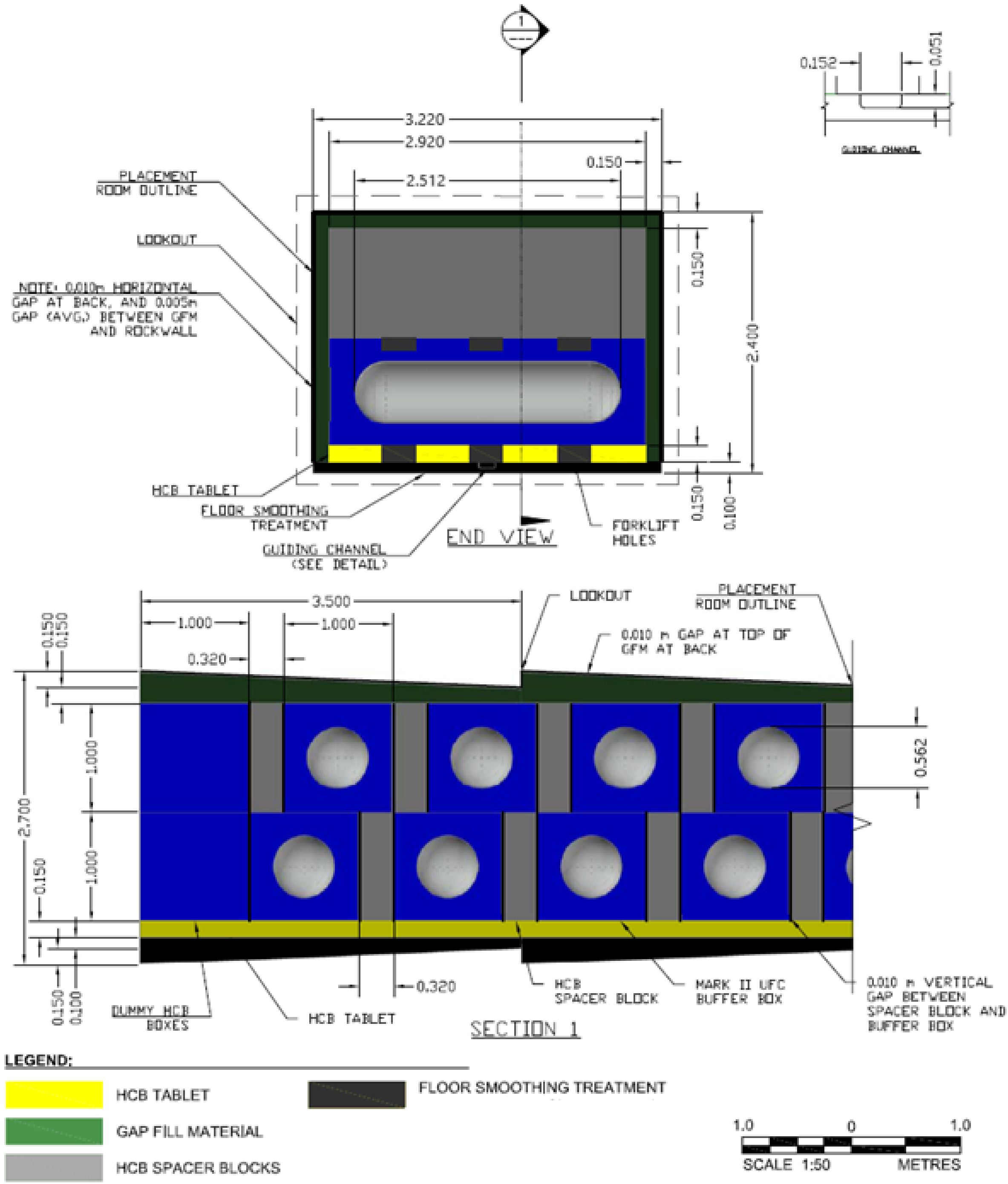

Figure A-5: Placement Room – Crystalline Rock

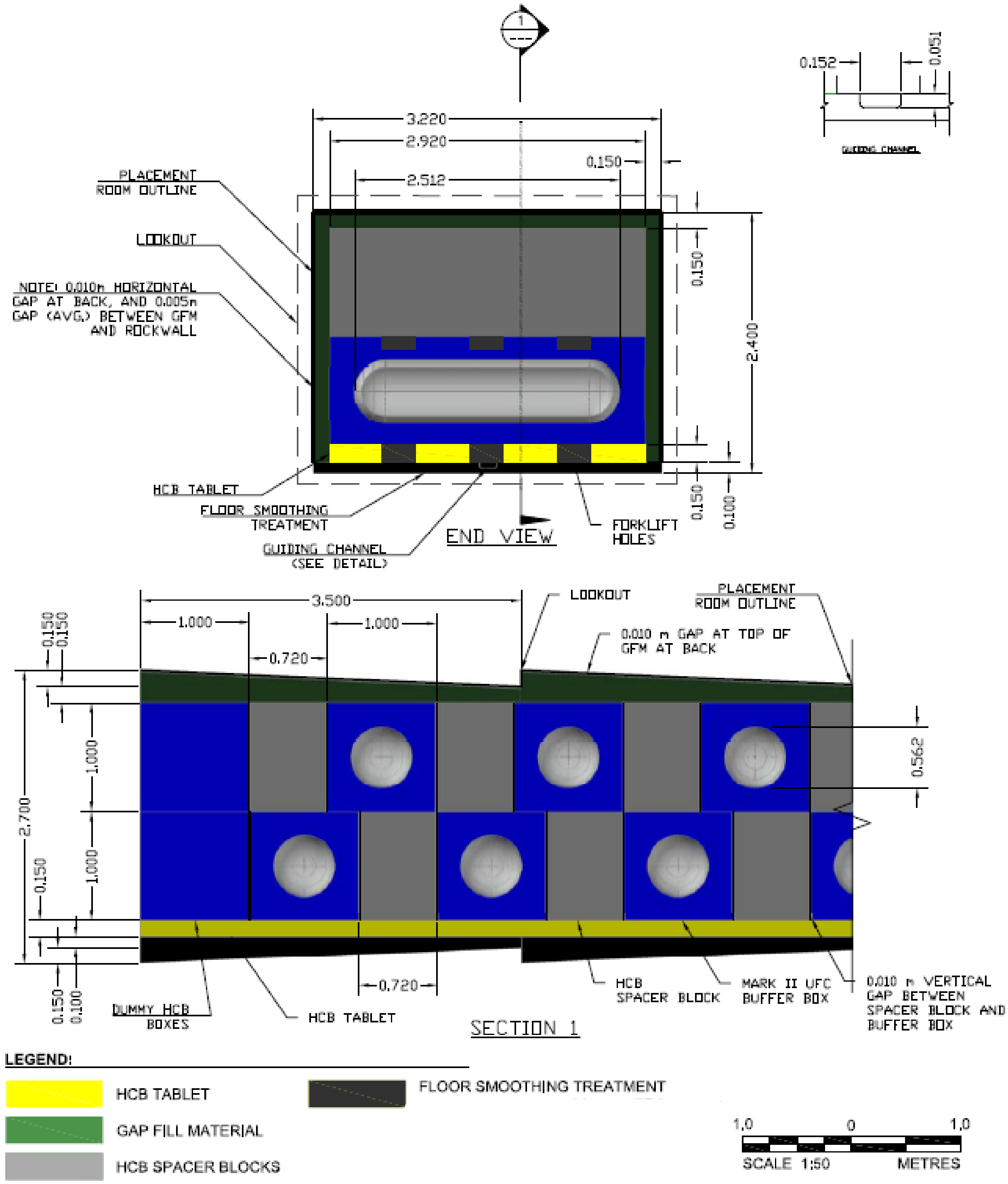

Figure A-6: Placement Room – Sedimentary Rock

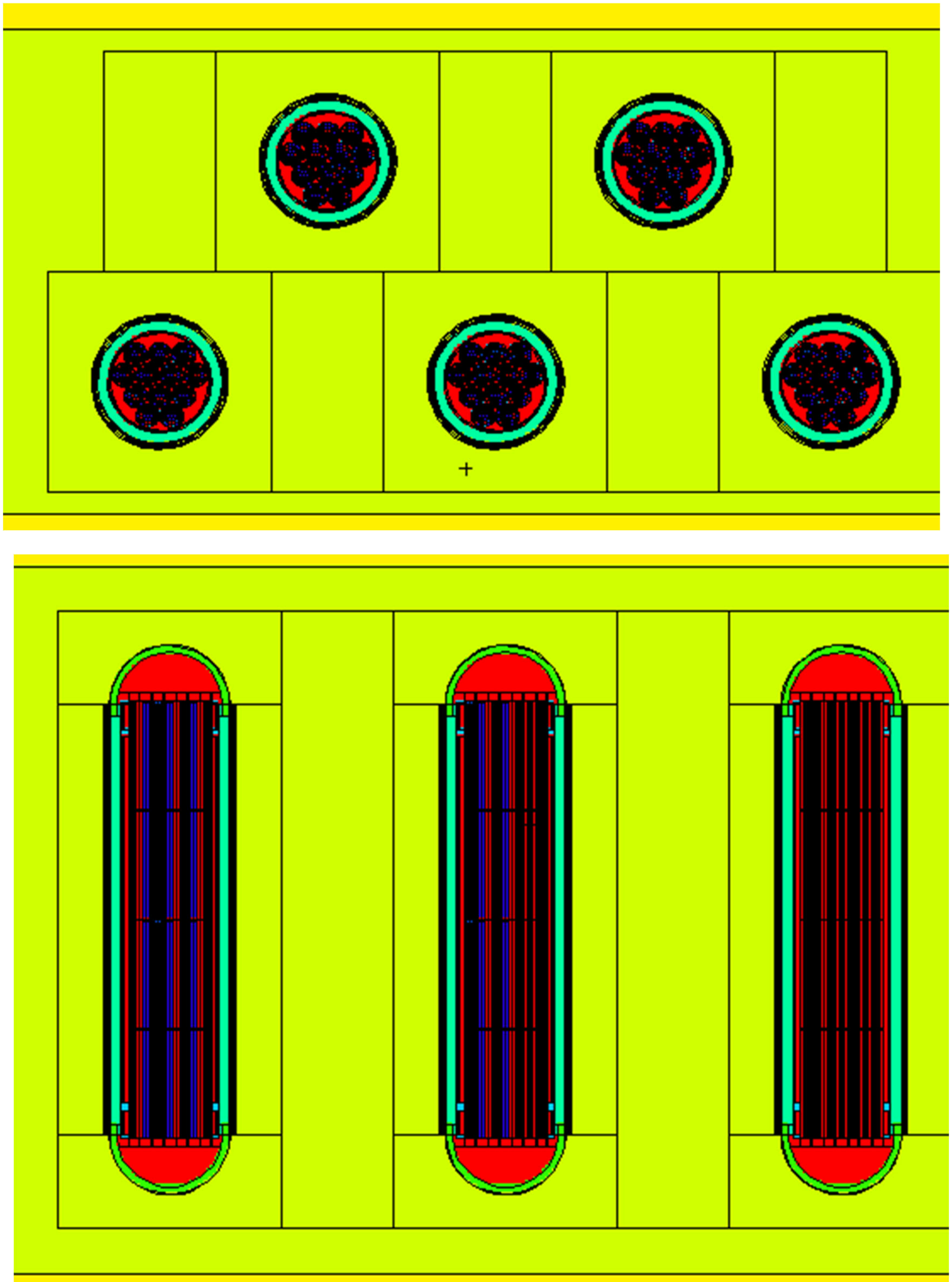

Figure A-7: MCNP Representation of Used Fuel Bundles in Placement Room

A.2 PARTICLE TRANSPORT CALCULATION OPTIONS

A.2.1 Alpha Transport

The following options (specified via PHYS:a particle physics option card) were applied to the alpha particle transport simulations:

- Choice of Coulomb scattering model: use of FermiLab angular deflection model with Vavilov straggling,
- Treatment of nuclear interactions: process all interactions, and
- Treatment of nuclear elastic scattering by Parel/Liu/Striganov model.

A.2.2 Beta (Electron) Transport

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

- Generate bremsstrahlung photons,
- Photon production by electrons is turned on,
- Perform full bremsstrahlung tabular angular distribution,
- Use sampled value straggling method to compute electron energy loss at each collision,
- Production of bremsstrahlung photons created along electron substeps: produce the analog number of bremsstrahlung photons. Radiative energy loss uses the bremsstrahlung energy of the first sampled photon,
- Sampling of electron-induced X-rays produced along electron substeps: produce the analog number of electron-induced X-rays,
- Creation of knock-on electrons produced in electron interactions: produce the analog number of knock-on electrons,
- Generation of photon-induced secondary electrons: Produce the analog number of photon-induced secondary electrons,
- Bremsstrahlung production on each electron substep: analog bremsstrahlung production,
- Coulomb scattering model: select the standard Goudsmit-Sunderson angular deflection method, and
- Start of single-event transport: apply 0.001 MeV energy above which MCNP transport electrons by the condensed-history algorithms and below which the single-event method is used.

A.2.3 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.2.4 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 : STATISTICAL UNCERTAINTIES

This Appendix provides statistical uncertainties of the dose rate values listed in the main body.

Table Number and Table Title	Statistical Uncertainties
Table 10: Maximum Alpha Dose Rates at Fuel-Water Interface	0.2% for all values
Table 11: Maximum Beta Dose Rates at Fuel Water Interface	See Table B-1
Table 12: Maximum Gamma Dose Rate at Fuel-Water Interface (Single Fuel Bundle)	See Table B-2
Table 13: Maximum Neutron Dose Rate at Fuel-Water Interface (Single Fuel Bundle)	0.2% for all values, except at 1E7 years (0.1%)
Table 14: Maximum Dose Rates at Fuel-Water Interface (Single Fuel Bundle)	Can be derived from uncertainties of the contributors in Table 10, Table 11, Table 12, and Table 13.
Table 15: Maximum Beta Dose Rates at Fuel-Water Interface (Single UFC)	See Table B-3
Table 16: Beta Dose Rates in Water Inside UFC (Single UFC)	See Table B-4
Table 17: Maximum Gamma Dose Rate at Fuel-Water Interface (Single UFC)	See Table B-5
Table 18: Gamma Dose Rates in Water Inside UFC (Single UFC)	See Table B-6
Table 19: Gamma Dose Rates in Water Outside UFC (Single UFC)	See Table B-7
Table 20: Maximum Neutron Dose Rate at Fuel-Water Interface (Single UFC)	See Table B-8
Table 21: Neutron Dose Rates in Water Inside UFC (Single UFC)	0.04% for water A 0.08% for water B
Table 22: Neutron Dose Rates in Water Outside UFC (Single UFC)	0.1% for all values
Table 23: Total Dose Rates at Fuel-Water Interface (Single UFC)	Can be derived from uncertainties of contributors in Table 10, Table 15, Table 17, and Table 20
Table 24: Total Dose Rates at Internal UFC-Water Interface (Single UFC)	Can be derived from uncertainties of contributors in Table 16, Table 18, and Table 21
Table 25: Total Dose Rates at External UFC-Water Interface (Single UFC)	Can be derived from uncertainties of contributors in Table 19 and Table 22
Table 26: Maximum Gamma and Neutron Dose Rates in Moist Air outside UFC (Single UFC)	See Table B-9
Table 27: Maximum Gamma Dose Rates at Fuel-Water Interface (Placement Room)	See Table B-10
Table 28: Gamma Dose Rates in Water Inside UFC (Placement Room)	See Table B-11
Table 29: Maximum Gamma Dose Rates Outside UFC (Placement Room)	See Table B-12
Table 30: Maximum Neutron Dose Rates at Fuel-Water Interface (Placement Room)	See Table B-13
Table 31: Neutron Dose Rates in Water Inside UFC (Placement Room)	0.1% for all values
Table 32: Maximum Neutron Dose Rates Outside UFC (Placement Room)	0.2% for all values

Table Number and Table Title	Statistical Uncertainties
Table 33: Maximum Dose Rates at UFC Interface (Placement Room)	Can be derived from uncertainties of contributors in Table 28 and Table 31
Table 34: Maximum Dose Rates Outside UFC (Placement Room Saturated with Moist Air)	Can be derived from uncertainties of contributors in Table 29 and Table 32
Table 41: Maximum Dose Rates at External UFC-Water Interface: Sensitivity Case 6	Table B-14

Table B-1: Statistical Uncertainties in Dose Rates Shown in Table 11

Bundle Burnup	Decay Time (Years)	Electron	Bremsstrahlung Photons
220 MWh/kgU	30	0.4%	0.4%
	100	0.3%	0.3%
	200	0.3%	0.3%
	300	0.4%	0.4%
	500	0.6%	0.8%
	1E+03	0.6%	0.9%
	1E+04	0.5%	0.8%
	1E+05	0.5%	0.6%
	1E+06	0.4%	0.5%
	1E+07	0.4%	0.4%
290 MWh/kgU	30	0.4%	0.4%
	100	0.3%	0.3%
	200	0.4%	0.4%
	300	0.4%	0.4%
	500	0.6%	0.9%
	1E+03	0.6%	1.1%
	1E+04	0.5%	1.0%
	1E+05	0.5%	0.7%
	1E+06	0.4%	0.5%
	1E+07	1.4%	1.2%

Table B-2: Statistical Uncertainties in Dose Rates Shown in Table 12

Bundle Burnup	Decay Time (Years)	Uncertainty	Bundle Burnup	Decay Time (Years)	Uncertainty
220 MWh/kgU	30	0.1%	290 MWh/kgU	30	0.1%
	100	0.1%		100	0.1%
	200	0.2%		200	0.2%
	300	0.4%		300	0.4%
	500	1.1%		500	1.5%
	1E+03	1.0%		1E+03	1.0%
	1E+04	0.7%		1E+04	0.6%
	1E+05	0.7%		1E+05	0.7%
	1E+06	0.6%		1E+06	0.6%
	1E+07	0.5%		1E+07	0.5%

Table B-3: Statistical Uncertainties in Dose Rates Shown in Table 15

Bundle Burnup	Decay Time (Years)	Uncertainty	Bundle Burnup	Decay Time (Years)	Uncertainty
220 MWh/kgU	30	1.0%	290 MWh/kgU	30	1.0%
	100	0.9%		100	0.9%
	200	1.0%		200	1.0%
	300	1.0%		300	1.0%
	500	1.7%		500	1.8%
	1E+03	1.8%		1E+03	1.8%
	1E+04	1.3%		1E+04	1.4%
	1E+05	1.3%		1E+05	1.4%
	1E+06	1.1%		1E+06	1.1%
	1E+07	3.8%		1E+07	3.8%

Table B-4: Statistical Uncertainties in Dose Rates Shown in Table 16

Bundle Burnup	Decay Time (years)	Uncertainty Water A	Uncertainty Water B
220 MWh/kgU	30	0.6%	1.9%
	100	0.5%	1.7%
	200	0.6%	1.8%
	300	0.6%	2.0%
	500	1.3%	4.4%
	1E+03	1.4%	4.5%
	1E+04	1.3%	4.4%
	1E+05	1.0%	3.8%
	1E+06	0.7%	2.4%
	1E+07	0.6%	2.0%
290 MWh/kgU	30	0.6%	1.8%
	100	0.6%	1.8%
	200	0.6%	1.8%
	300	0.6%	2.0%
	500	1.5%	4.8%
	1E+03	1.7%	5.4%
	1E+04	1.5%	4.7%
	1E+05	1.2%	4.1%
	1E+06	0.8%	2.5%
	1E+07	0.6%	2.0%

Table B-5: Statistical Uncertainties in Dose Rates Shown in Table 17

Bundle Burnup	Decay Time (Years)	Uncertainty
220 MWh/kgU	30	0.2%
	100	0.2%
	200	0.3%
	300	0.5%
	500	0.3%
	1E+03	0.3%
	1E+04	1.0%
	1E+05	0.3%
	1E+06	0.9%
	1E+07	0.7%
290 MWh/kgU	30	0.2%
	100	0.2%
	200	0.3%
	300	0.5%
	500	0.3%
	1E+03	0.3%
	1E+04	0.8%
	1E+05	0.4%
	1E+06	0.9%
	1E+07	0.7%

Table B-6: Statistical Uncertainties in Dose Rates Shown in Table 18

Bundle Burnup	Decay Time (years)	Uncertainty Water A	Uncertainty Water B
220 MWh/kgU	30	0.1%	0.1%
	100	0.1%	0.2%
	200	0.1%	0.2%
	300	0.3%	0.5%
	500	0.3%	0.4%
	1E+03	0.3%	0.4%
	1E+04	0.5%	0.7%
	1E+05	0.2%	0.2%
	1E+06	0.1%	0.2%
	1E+07	0.1%	0.2%
290 MWh/kgU	30	0.1%	0.1%
	100	0.1%	0.2%
	200	0.1%	0.2%
	300	0.3%	0.5%
	500	0.3%	0.4%
	1E+03	0.2%	0.3%
	1E+04	0.5%	0.7%
	1E+05	0.2%	0.2%
	1E+06	0.2%	0.2%
	1E+07	0.1%	0.2%

Table B-7: Statistical Uncertainties in Dose Rates Shown in Table 19

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.4%	290 MWh/kgU	30	0.4%
	100	0.4%		100	0.4%
	200	0.6%		200	0.6%
	300	1.3%		300	1.4%
	500	1.2%		500	1.3%
	1E+03	1.4%		1E+03	1.3%
	1E+04	2.2%		1E+04	2.2%
	1E+05	0.5%		1E+05	0.5%
	1E+06	0.4%		1E+06	0.4%
	1E+07	0.3%		1E+07	0.3%

Table B-8: Statistical Uncertainties in Dose Rates Shown in Table 20

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.1%	290 MWh/kgU	30	0.1%
	100	0.3%		100	0.3%
	200	0.3%		200	0.3%
	300	0.3%		300	0.3%
	500	0.3%		500	0.3%
	1.E+03	0.3%		1.E+03	0.3%
	1.E+04	0.3%		1.E+04	0.3%
	1.E+05	0.1%		1.E+05	0.1%
	1.E+06	0.1%		1.E+06	0.1%
	1.E+07	0.3%		1.E+07	0.3%

Table B-9: Statistical Uncertainties in Dose Rates Shown in Table 26

Bundle Burnup	Decay Time (years)	Uncertainty Gamma	Uncertainty Neutron
220 MWh/kgU	30	0.3%	0.04%
	100	0.4%	0.04%
	200	0.5%	0.04%
	300	1.1%	0.04%
	500	1.0%	0.04%
	1E+03	1.1%	0.04%
	1E+04	1.8%	0.04%
	1E+05	0.4%	0.04%
	1E+06	0.4%	0.04%
	1E+07	0.3%	0.04%
290 MWh/kgU	30	0.3%	0.04%
	100	0.3%	0.04%
	200	0.5%	0.04%
	300	1.1%	0.04%
	500	1.0%	0.04%
	1E+03	1.0%	0.04%
	1E+04	1.8%	0.04%
	1E+05	0.5%	0.04%
	1E+06	0.4%	0.04%
	1E+07	0.3%	0.04%

Table B-10: Statistical Uncertainties in Dose Rates Shown in Table 27

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.2%	290 MWh/kgU	30	0.2%
	100	0.2%		100	0.2%
	200	0.3%		200	0.3%
	300	0.6%		300	0.6%
	500	0.2%		500	0.2%
	1.E+03	0.2%		1.E+03	0.2%
	1.E+04	0.3%		1.E+04	0.3%
	1.E+05	1.2%		1.E+05	1.3%
	1.E+06	1.1%		1.E+06	1.1%
	1.E+07	0.8%		1.E+07	0.9%

Table B-11: Statistical Uncertainties in Dose Rates Shown in Table 28

Bundle Burnup	Decay Time (years)	Uncertainty Water A	Uncertainty Water B
220 MWh/kgU	30	0.1%	0.2%
	100	0.1%	0.2%
	200	0.2%	0.3%
	300	0.4%	0.6%
	500	0.2%	0.3%
	1E+03	0.2%	0.3%
	1E+04	0.2%	0.3%
	1E+05	0.2%	0.3%
	1E+06	0.2%	0.2%
	1E+07	0.1%	0.2%
290 MWh/kgU	30	0.1%	0.2%
	100	0.1%	0.2%
	200	0.2%	0.3%
	300	0.4%	0.6%
	500	0.2%	0.3%
	1E+03	0.2%	0.2%
	1E+04	0.2%	0.3%
	1E+05	0.2%	0.3%
	1E+06	0.2%	0.3%
	1E+07	0.1%	0.2%

Table B-12: Statistical Uncertainties in Dose Rates Shown in Table 29

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.5%	290 MWh/kgU	30	0.5%
	100	0.5%		100	0.5%
	200	0.7%		200	0.7%
	300	0.5%		300	0.5%
	500	0.8%		500	0.9%
	1E+03	0.9%		1E+03	0.9%
	1E+04	0.8%		1E+04	0.9%
	1E+05	0.6%		1E+05	0.7%
	1E+06	0.5%		1E+06	0.5%
	1E+07	0.4%		1E+07	0.4%

Table B-13: Statistical Uncertainties in Dose Rates Shown in Table 30

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.2%	290 MWh/kgU	30	0.2%
	100	0.2%		100	0.2%
	200	0.5%		200	0.2%
	300	0.2%		300	0.4%
	500	0.5%		500	0.5%
	1E+03	0.2%		1E+03	0.5%
	1E+04	0.5%		1E+04	0.5%
	1E+05	0.2%		1E+05	0.2%
	1E+06	0.2%		1E+06	0.2%
	1E+07	0.5%		1E+07	0.5%

Table B-14: Statistical Uncertainties in Dose Rates Shown in Table 41

Bundle Burnup	Decay Time (years)	Uncertainty	Bundle Burnup	Decay Time (years)	Uncertainty
220 MWh/kgU	30	0.06%	290 MWh/kgU	30	0.06%
	100	0.06%		100	0.06%
	200	0.09%		200	0.09%
	300	0.20%		300	0.20%
	500	0.31%		500	0.32%
	1E+03	0.33%		1E+03	0.33%
	1E+04	0.32%		1E+04	0.31%
	1E+05	0.08%		1E+05	0.08%
	1E+06	0.06%		1E+06	0.07%
	1E+07	0.05%		1E+07	0.05%

APPENDIX C : DOSE RATE CALCULATIONS USING 28-ELEMENT FUEL BUNDLE DESIGN

C.1 INTRODUCTION

As stated in Section 2.1, the reference bundle type was selected to as the regular 37-element fuel bundle design, which is the most abundant fuel type in the current inventory of used CANDU fuel. Screening studies (Heckman and Edward 2020) indicated that the differences in the source terms are small (approximately 5%). Based on the radiation source comparisons (see Appendix C.3), the 28-element fuel bundle yields higher radiation source terms per bundle compared to the reference bundle. The higher source terms do not necessarily translate to higher absorbed dose rates in water due to the differences in the fuel element configuration.

To confirm the expectation that the dose rates calculated using the reference fuel bundle is limiting versus that of 28-element fuel bundle design, two fuel configurations have been analysed and documented in this Appendix:

- Configuration 1: single used fuel bundle submerged in water.
  This configuration is comparable to that of Section 3.
- Configuration 2: single intact UFC submerged in water.
  This configuration is comparable to that of Section 8.3.1.

The methodology for the different particle transport calculations for the assessing the 28-element fuel bundle is identical to that of the reference fuel bundle, see Section 2.3.

The Appendix is structured as follows:

- Appendix C.2 describes the geometry representation of the 28-element fuel bundle.
- Appendix C.3 describes the radiation sources associated with the 28-element fuel bundle.
- Appendix C.4 describes the results of the single used fuel bundle submerged in water configuration.
  With respect to the fuel-water interface dose rates (i.e., closest to the fuel bundle), the reference fuel bundle is bounding, except at 10 million years decay time. The dose rates at the fuel-water interface are dominated by the short-range particles (alphas and betas). The alpha and beta emissions per unit mass is slightly lower (by one or two percent) in the 28-element fuel bundle.

- Appendix C.5 describes the results of the single intact UFC submerged in water configuration.
  With respect to the dose rates outside of a UFC, the reference fuel bundle is not bounding. Dose rates at the exterior UFC-water interface are dominated by gammas. Since the gamma emissions per bundle or in outer ring fuel elements are higher in the 28-element fuel bundle versus the reference fuel bundle, more gammas reach the exterior of the UFC and deposit their energy at the UFC-water interface.

C.2 GEOMETRY REPRESENTATION

Input parameters and geometry pertaining to the 28-element used fuel bundle are listed in Table C-1 and Figure C-1.

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

For the configuration where the 28-element used fuel bundles are placed in a UFC, the geometry representation of the UFC is identical to that of the 37-element fuel bundle (see Appendix A.1.2), except for the 48 used fuel bundles. The fuel bundles are replaced with the 28-element fuel bundle design.

Table C-1: 28-Element Fuel Bundle Parameters

Parameter	Value
Number of fuel elements per bundle	28
UO_2 pellet diameter	14.3 mm
Sheath outer diameter Sheath thickness Sheath inner diameter	15.2 mm 0.4 mm 14.4 mm
Gap between UO_2 pellet and sheath	0.05 mm
Stack length	481 mm
U loading	20.1 kg
UO_2 loading	22.8 kg
UO_2 density	10.54 g/cm^3
Bundle length	497 mm
End plate diameter End plate thickness	88.4 mm 1.6 mm
Material for sheath, end plates, end caps	Zircaloy-4

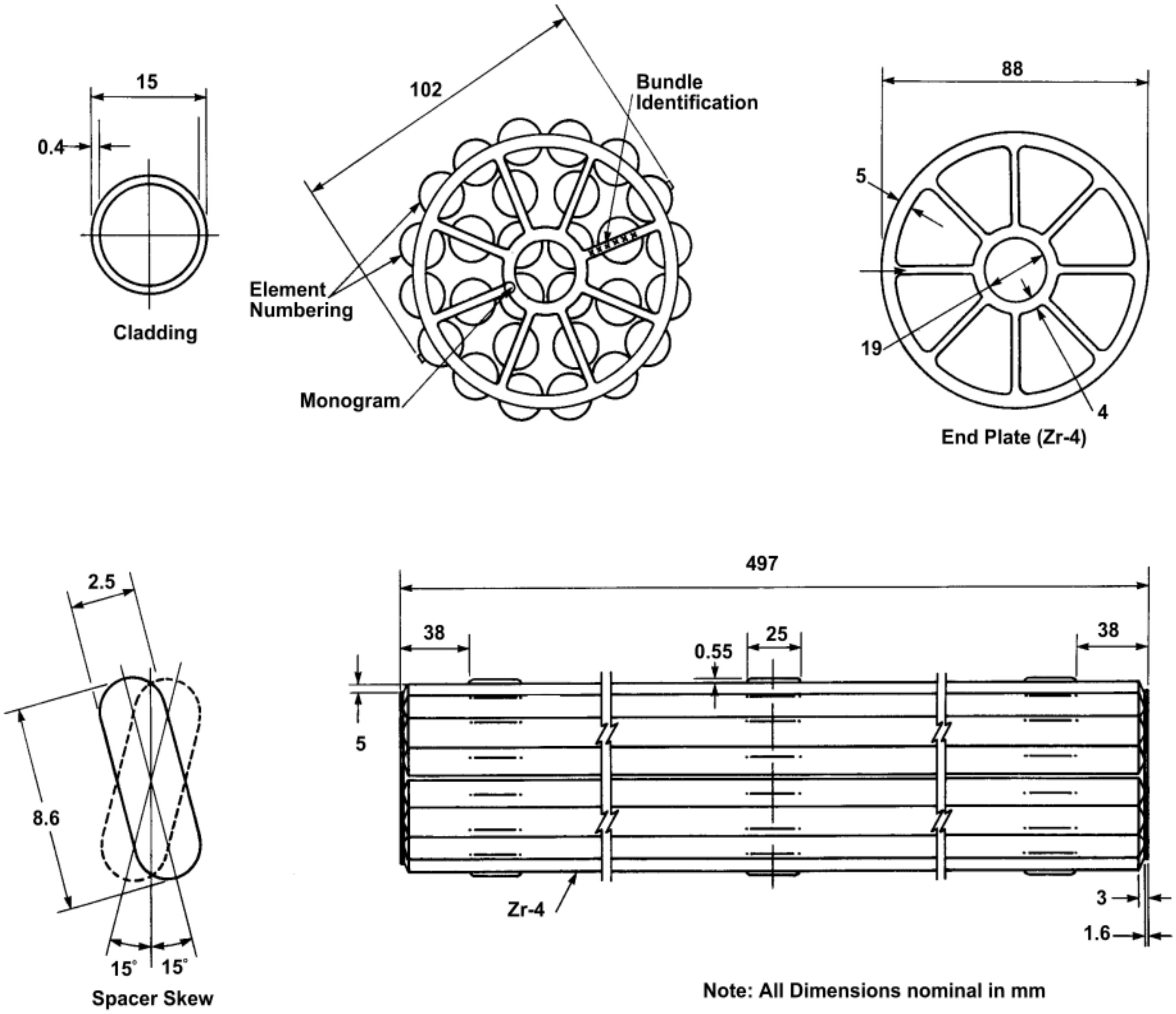

Figure C-1: 28-Element Fuel Bundle Design

Source: Figure 1 of Heckman and Edwards 2020.

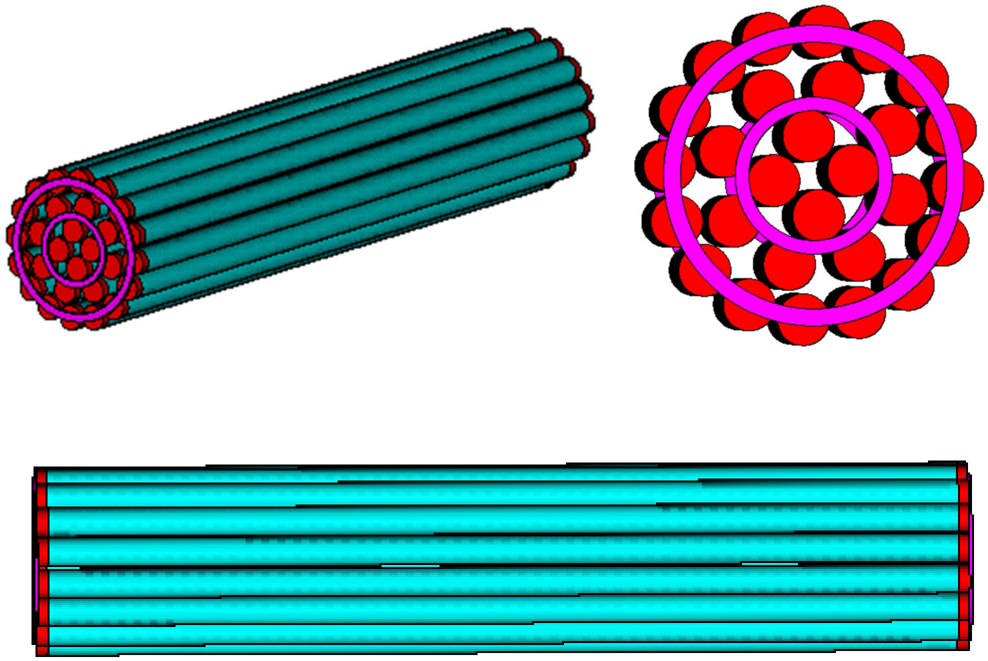

Figure C-2: MCNP Representation of 28-Element Used Fuel Bundle

C.3 RADIATION SOURCES

C.3.1 Alpha Emissions

Similar to the reference fuel bundle (see Section 2.2.1), the alpha emission spectra for the 28-element fuel bundle design were generated for each fuel ring (see Figure C-3 for the fuel ring designation) using the radionuclide inventory estimated in Heckman and Edward 2020. Alpha source terms were calculated using the ORIGEN module from the SCALE code package, which uses an alpha decay library based on the ENDF/B-VII.1 nuclear data file. Alpha particles were modelled with discrete energies and the source spectrum was generated by straightforward binning into the desired energy group structure.

The alpha source intensity and average energy in each ring at different burnup and decay times are listed in Table C-2 and Table e-3, respectively. Comparisons against the reference fuel bundle alpha emissions are also shown in Table e-3.

Similar to the approach for the reference fuel bundle (see Section 2.2.1), the alpha transport calculations were performed for sources emitted from the outer ring fuel element only since it emits more alphas compared to other fuel elements. Since the alpha particle range is short (less than 50 μm in water), the alpha particles emitted from a fuel element will not reach locations beyond its own fuel sheath. The dose rates obtained for the outer ring fuel element would bound the alpha dose rates from other fuel elements. The outer ring fuel element in a 28-element fuel bundle emits more alphas per element compared to that of the regular fuel bundle.

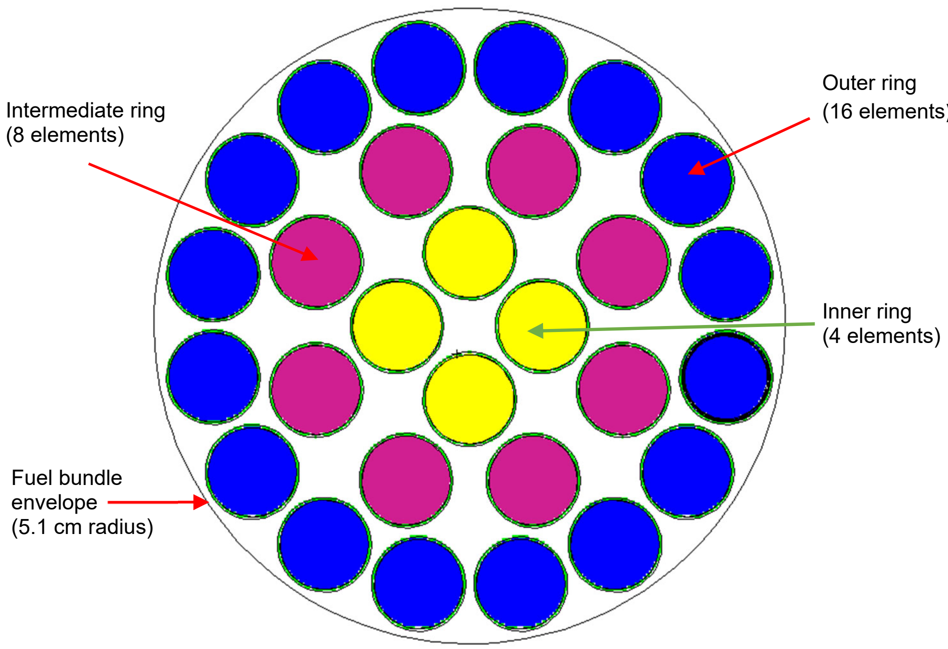

Figure C-3: Ring-Wise Designation in 28-Element Fuel Bundle Model

Table C-2: Alpha Source Intensity (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Alpha/s Source Intensity per Fuel Element: Inner Ring	Intermediate Ring	Outer Ring	Ratio to Reference Fuel Bundle (see Note 1)
220 MWh/kgU	30	2.74E+10	3.05E+10	3.93E+10	1.05
	100	2.83E+10	3.16E+10	4.09E+10	1.05
	200	2.50E+10	2.79E+10	3.59E+10	1.05
	300	2.25E+10	2.50E+10	3.21E+10	1.05
	1E+03	1.37E+10	1.49E+10	1.82E+10	1.05
	1E+04	5.44E+09	5.72E+09	6.44E+09	1.05
	1E+05	3.22E+08	3.28E+08	3.53E+08	1.05
	1E+06	1.01E+08	1.04E+08	1.14E+08	1.05
	1E+07	7.58E+07	7.60E+07	7.66E+07	1.05
290 MWh/kgU	30	3.87E+10	4.26E+10	5.31E+10	1.05
	100	3.97E+10	4.35E+10	5.39E+10	1.05
	200	3.48E+10	3.81E+10	4.69E+10	1.05
	300	3.10E+10	3.39E+10	4.15E+10	1.05
	1E+03	1.80E+10	1.93E+10	2.28E+10	1.05
	1E+04	6.51E+09	6.77E+09	7.43E+09	1.05
	1E+05	3.53E+08	3.57E+08	3.82E+08	1.06
	1E+06	1.13E+08	1.17E+08	1.29E+08	1.05
	1E+07	7.66E+07	7.68E+07	7.74E+07	1.05

Notes
General: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.
Note 1: Values shown are the ratio of alpha/s in a 28-element fuel bundle to that that of the reference fuel bundle (see Table 2).

Table e-3: Alpha Source Average Energy (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Alpha Particles		
		Inner Ring	Intermediate Ring	Outer Ring
220 MWh/kgU	30	5.38	5.39	5.40
	100	5.37	5.38	5.39
	200	5.35	5.35	5.37
	300	5.33	5.34	5.35
	1E+03	5.24	5.25	5.26
	1E+04	5.15	5.15	5.15
	1E+05	5.20	5.20	5.19
	1E+06	5.48	5.49	5.52
	1E+07	5.35	5.35	5.35
290 MWh/kgU	30	5.40	5.40	5.41
	100	5.38	5.39	5.39
	200	5.36	5.36	5.37
	300	5.34	5.35	5.36
	1E+03	5.26	5.26	5.27
	1E+04	5.15	5.15	5.15
	1E+05	5.20	5.19	5.18
	1E+06	5.52	5.52	5.54
	1E+07	5.35	5.34	5.34
Note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.				

C.3.2 Beta Emissions

Similar to the reference fuel bundle (see Section 2.2.2), the beta emission spectra were generated for each fuel ring using the radionuclide inventory estimated in Heckman and Edward 2020. The beta emission energy spectra were determined using the ORIGEN module from the SCALE code package. The beta source intensity and average energy in each ring at different burnup and decay times are listed in Table C-4 and Table C-5, respectively. Comparisons against the reference fuel bundle beta emissions are also shown in Table C-4.

Table C-4: Beta Source Intensity (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Beta/s Source Intensity per Fuel Element			Ratio to Reference Fuel Bundle (see Note 1)
		Inner Ring	Intermediate Ring	Outer Ring	
220 MWh/kgU	30	8.62E+11	9.54E+11	1.17E+12	1.05
	100	1.43E+11	1.57E+11	1.90E+11	1.05
	200	1.36E+10	1.49E+10	1.79E+10	1.05
	300	1.58E+09	1.72E+09	2.05E+09	1.06
	1E+03	1.52E+08	1.69E+08	2.17E+08	1.07
	1E+04	1.39E+08	1.52E+08	1.87E+08	1.06
	1E+05	1.28E+08	1.36E+08	1.58E+08	1.05
	1E+06	7.82E+07	8.09E+07	8.82E+07	1.05
	1E+07	5.46E+07	5.47E+07	5.49E+07	1.05
290 MWh/kgU	30	1.13E+12	1.23E+12	1.49E+12	1.05
	100	1.83E+11	2.00E+11	2.38E+11	1.05
	200	1.73E+10	1.89E+10	2.25E+10	1.05
	300	2.01E+09	2.19E+09	2.61E+09	1.06
	1E+03	2.12E+08	2.40E+08	3.22E+08	1.07
	1E+04	1.84E+08	2.03E+08	2.56E+08	1.06

Bundle Burnup	Decay Time (Years)	Beta/s Source Intensity per Fuel Element Inner Ring	Intermediate Ring	Outer Ring	Ratio to Reference Fuel Bundle (see Note 1)
	1E+05	1.57E+08	1.67E+08	1.94E+08	1.05
	1E+06	8.75E+07	9.07E+07	9.93E+07	1.05
	1E+07	5.49E+07	5.50E+07	5.52E+07	1.05
General note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements. Note 1: values shown are the ratio of beta/s in a 28-element fuel bundle to that that of the reference fuel bundle (see Table 3).					

Table C-5: Beta Source Average Energy (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Beta Particles		
		Inner Ring	Intermediate Ring	Outer Ring
220 MWh/kgU	30	0.35	0.35	0.34
	100	0.40	0.40	0.39
	200	0.39	0.39	0.38
	300	0.33	0.33	0.32
	1E+03	0.13	0.13	0.12
	1E+04	0.14	0.14	0.13
	1E+05	0.18	0.18	0.16
	1E+06	0.29	0.28	0.27
	1E+07	0.35	0.35	0.35
290 MWh/kgU	30	0.34	0.34	0.33
	100	0.39	0.39	0.38
	200	0.38	0.38	0.38
	300	0.32	0.32	0.32
	1E+03	0.12	0.12	0.11
	1E+04	0.13	0.13	0.12
	1E+05	0.17	0.16	0.15
	1E+06	0.27	0.27	0.26
	1E+07	0.35	0.35	0.35
Note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.				

C.3.3 Gamma Emissions

Ring-wise gamma emission spectra from Heckman and Edward 2020 were applied. The gamma source intensity and average energy in each ring at different burnup and decay times are listed in Table C-6 and Table C-7, respectively. The gamma emissions were binned into 20 energy groups.

The principal gamma sources are:

- Ba-137m for 30 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 C-6: Gamma Source Intensity (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Gamma/s Source Intensity per Fuel Element			Ratio to Reference Fuel Bundle	
		Inner Ring	Intermediate Ring	Outer Ring	Whole Bundle	Outer Ring
220 MWh/kgU	30	5.22E+11	5.76E+11	7.02E+11	1.05	1.20
	100	1.10E+11	1.22E+11	1.50E+11	1.05	1.20
	200	1.96E+10	2.20E+10	2.86E+10	1.05	1.19
	300	9.57E+09	1.09E+10	1.48E+10	1.05	1.19
	1E+03	3.29E+09	3.75E+09	5.06E+09	1.05	1.19
	1E+04	3.47E+08	3.81E+08	4.67E+08	1.05	1.20
	1E+05	6.36E+07	6.69E+07	7.63E+07	1.05	1.21
	1E+06	6.50E+07	6.78E+07	7.60E+07	1.05	1.21
	1E+07	4.31E+07	4.32E+07	4.34E+07	1.05	1.23
290 MWh/kgU	30	6.79E+11	7.45E+11	8.98E+11	1.05	1.20
	100	1.45E+11	1.59E+11	1.94E+11	1.05	1.20
	200	2.72E+10	3.01E+10	3.78E+10	1.05	1.20
	300	1.39E+10	1.55E+10	2.00E+10	1.05	1.19
	1E+03	4.79E+09	5.35E+09	6.87E+09	1.05	1.19
	1E+04	4.73E+08	5.18E+08	6.30E+08	1.05	1.20
	1E+05	7.63E+07	8.02E+07	9.15E+07	1.05	1.21
	1E+06	7.51E+07	7.84E+07	8.78E+07	1.05	1.21
	1E+07	4.34E+07	4.35E+07	4.38E+07	1.05	1.23

Notes:

- See Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.
- Ratio to reference fuel bundle:
  - Whole bundle values include the contribution from 37 fuel elements (reference fuel bundle, see Table 5).
  - Outer ring values include the contribution from 16 fuel elements (for 28-element fuel bundle) or 18 fuel elements (for reference fuel bundle).

Table C-7: Gamma Source Average Energy (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Gamma Particles		
		Inner Ring	Intermediate Ring	Outer Ring
220 MWh/kgU	30	0.39	0.39	0.39
	100	0.36	0.36	0.36
	200	0.22	0.22	0.21
	300	0.08	0.08	0.07
	1E+03	0.04	0.04	0.04
	1E+04	0.04	0.04	0.04
	1E+05	0.23	0.22	0.21
	1E+06	0.30	0.29	0.27
	1E+07	0.39	0.39	0.39
290 MWh/kgU	30	0.39	0.39	0.40
	100	0.36	0.37	0.37
	200	0.21	0.21	0.21
	300	0.07	0.07	0.07
	1E+03	0.04	0.04	0.04
	1E+04	0.04	0.04	0.05

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Gamma Particles		
		Inner Ring	Intermediate Ring	Outer Ring
	1E+05	0.22	0.21	0.21
	1E+06	0.27	0.26	0.25
	1E+07	0.39	0.39	0.39
Note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.				

C.3.4 Neutron Emissions

Ring-wise neutron emission spectra from Heckman and Edward 2020 were applied. The neutron source intensity and average energy in each ring at different burnup and decay times are listed in Table 7 and Table 8 respectively. Neutron emissions were binned into 46 energy groups. The neutron emission spectra at different decay times are shown in Figure 8. The neutron source intensity at different burnup and decay times is shown in Figure 9.

Table C-8: Neutron Source Intensity (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Neutron/s Source Intensity per Fuel Element			Ratio to Reference Fuel Bundle (see Note 1)
		Inner Ring	Intermediate Ring	Outer Ring	
220 MWh/kgU	30	1.89E+03	2.32E+03	3.80E+03	1.05
	100	1.43E+03	1.62E+03	2.14E+03	1.04
	200	1.32E+03	1.48E+03	1.90E+03	1.04
	300	1.25E+03	1.41E+03	1.80E+03	1.04
	1E+03	1.01E+03	1.13E+03	1.44E+03	1.04
	1E+04	4.30E+02	4.86E+02	6.33E+02	1.04
	1E+05	6.83E+01	8.62E+01	1.46E+02	1.03
	1E+06	2.17E+01	2.51E+01	3.65E+01	1.03
	1E+07	1.11E+01	1.11E+01	1.11E+01	1.05
290 MWh/kgU	30	3.94E+03	5.11E+03	9.34E+03	1.05
	100	2.13E+03	2.42E+03	3.22E+03	1.05
	200	1.88E+03	2.09E+03	2.61E+03	1.04
	300	1.78E+03	1.98E+03	2.47E+03	1.04
	1E+03	1.43E+03	1.59E+03	1.97E+03	1.04
	1E+04	6.20E+02	6.98E+02	9.06E+02	1.04
	1E+05	1.26E+02	1.60E+02	2.69E+02	1.03
	1E+06	3.28E+01	3.92E+01	5.98E+01	1.03
	1E+07	1.11E+01	1.11E+01	1.11E+01	1.05
General note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements. Note 1: values shown are the ratio of neutron/s in a 28-element fuel bundle to that that of the reference fuel bundle (see Table 5).					

Table C-9: Neutron Source Average Energy (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Average MeV of Emitted Neutron Particles		
		Inner Ring	Intermediate Ring	Outer Ring
220 MWh/kgU	30	2.12	2.12	2.12
	100	2.12	2.12	2.12
	200	2.10	2.10	2.11
	300	2.09	2.09	2.10
	1E+03	2.04	2.04	2.04
	1E+04	2.02	2.02	2.01
	1E+05	1.97	1.97	1.97
	1E+06	1.90	1.91	1.94
	1E+07	1.80	1.80	1.80
290 MWh/kgU	30	2.12	2.12	2.13
	100	2.12	2.12	2.12
	200	2.10	2.10	2.11
	300	2.09	2.09	2.09
	1E+03	2.04	2.04	2.04
	1E+04	2.01	2.01	2.00
	1E+05	1.97	1.97	1.98
	1E+06	1.93	1.94	1.96
	1E+07	1.80	1.80	1.80
Note: see Figure C-3 for the locations of inner ring, intermediate ring, and outer ring fuel elements.				

C.4 SINGLE FUEL BUNDLES SUBMERGED IN WATER

This scenario is comparable to the one discussed in Section 3.

In this scenario, the used fuel bundle is submerged in water. The fuel sheath integrity is breached, allowing water ingress into the space or gap between the fuel elements and the fuel sheath. The water density is set to 0.9655 g/cm^3, which is the value at 93°C and 5 MPa. The 93°C temperature is based on the maximum container surface temperature associated with an example sedimentary rock placement room configuration (Guo 2018); the actual temperature would vary depending on the fuel burnup and age.

The maximum dose rates at the fuel-water interface are presented.

C.4.1 Alpha Dose Rates

Only alpha particles that are generated within the 20 µm from the fuel-water interface contribute to the dose rates in water. For the alpha particles which reach the fuel-water interface, essentially all alpha particles escaping the fuel element is deposited within the 50 µm water-filled gap between the fuel element and the fuel sheath. Insignificant alpha particle energy is deposited in the fuel sheath and no energy deposition occurs beyond the fuel sheath.

Maximum dose rates in water in contact with the fuel surface at different decay times are listed in Table C-10. The alpha dose rate increases during the 30 to 100 years decay time due to the buildup of Am-241 (half-life = 432 years) from Pu-241 (half-life = 14 years) decay. Am-241 is the principal alpha source in the used fuel element during that decay period.

Although the total alpha sources from the 28-element fuel bundle is approximately 5% higher than that of the reference fuel bundle (see Table C-2), the resulting maximum alpha dose rates at the fuel-water interface associated with the 28-element fuel bundle are slightly lower than that of the reference fuel bundle (see Table C-10).

Table C-10: Maximum Alpha Dose Rates at Fuel-Water Interface (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Gy/h	Ratio to Reference Fuel Bundle (see Table 10)
220 MWh/kgU	30	1.60E+02	0.97
	100	1.66E+02	0.97
	200	1.45E+02	0.97
	300	1.29E+02	0.97
	1E+03	7.11E+01	0.97
	1E+04	2.41E+01	0.99
	1E+05	1.28E+00	0.99
	1E+06	4.23E-01	0.99
	1E+07	2.66E-01	1.00
290 MWh/kgU	30	2.17E+02	0.97
	100	2.20E+02	0.97
	200	1.90E+02	0.97
	300	1.67E+02	0.97
	1E+03	8.92E+01	0.98
	1E+04	2.78E+01	0.99
	1E+05	1.37E+00	0.99
	1E+06	4.81E-01	0.99
	1E+07	2.67E-01	1.00

C.4.2 Beta Dose Rates

The maximum dose rates occur in water that fills the gap between the fuel element and the fuel sheath. The maximum beta dose rates in water are listed in Table C-11. The maximum beta dose rates at the fuel-water interface associated with the 28-element fuel bundle are slightly higher than that of the reference fuel bundle.

Table C-11: Maximum Beta Dose Rates at Fuel Water Interface (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Absorbed Dose Rate, Gy/h			Ratio to Reference Fuel Bundle, see Table 11
		Electron	Secondary Photons	Total	
220 MWh/kgU	30	1.74E+02	1.21E+00	1.75E+02	1.010
	100	3.24E+01	2.25E-01	3.26E+01	1.005
	200	3.00E+00	2.03E-02	3.02E+00	1.005
	300	2.86E-01	1.90E-03	2.88E-01	0.996
	1E+03	9.00E-03	3.54E-05	9.03E-03	1.028
	1E+04	1.18E-02	3.69E-05	1.19E-02	1.029
	1E+05	1.20E-02	5.01E-05	1.20E-02	1.035
	1E+06	1.09E-02	6.12E-05	1.09E-02	1.036
	1E+07	9.75E-03	7.16E-05	9.82E-03	1.023
290 MWh/kgU	30	2.14E+02	1.47E+00	2.16E+02	1.011
	100	3.98E+01	2.74E-01	4.01E+01	1.007
	200	3.69E+00	2.48E-02	3.71E+00	1.006
	300	3.55E-01	2.28E-03	3.57E-01	1.004
	1E+03	1.23E-02	3.75E-05	1.24E-02	1.032
	1E+04	1.49E-02	3.98E-05	1.49E-02	1.014

Bundle Burnup	Decay Time (Years)	Absorbed Dose Rate, Gy/h			Ratio to Reference Fuel Bundle, see Table 11
		Electron	Secondary Photons	Total	
	1E+05	1.33E-02	5.31E-05	1.34E-02	1.041
	1E+06	1.14E-02	6.24E-05	1.15E-02	1.017
	1E+07	9.82E-03	7.40E-05	9.89E-03	1.013

C.4.3 Gamma Dose Rates

The maximum dose rates occur in the interstitial water within the fuel bundle envelope. The maximum beta dose rates in water are listed in Table C-12. The maximum gamma dose rates at the fuel-water interface associated with the 28-element fuel bundle are slightly higher than that of the reference fuel bundle.

Table C-12: Maximum Gamma Dose Rates at Fuel Water Interface (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Gy/h	Ratio to Reference Fuel Bundle, see Table 12
220 MWh/kgU	30	4.44E+01	1.03
	100	8.70E+00	1.03
	200	8.79E-01	1.03
	300	9.37E-02	1.02
	500	-	-
	1E+03	1.91E-03	1.04
	1E+04	1.33E-03	1.07
	1E+05	3.06E-03	1.00
	1E+06	4.59E-03	1.00
	1E+07	4.31E-03	0.99
290 MWh/kgU	30	5.86E+01	1.03
	100	1.15E+01	1.03
	200	1.16E+00	1.03
	300	1.23E-01	1.02
	500	-	-
	1E+03	2.80E-03	1.03
	1E+04	1.79E-03	1.06
	1E+05	3.38E-03	1.00
	1E+06	4.68E-03	1.00
	1E+07	4.30E-03	0.99

C.4.4 Neutron Dose Rates

The maximum dose rates in water inside the used fuel bundle envelope are listed in Table 13. Contributions from neutrons and neutron-capture gammas are accounted for. Absorbed dose rates in water from neutron sources are small compared to the dose rates from alpha, beta, and gamma sources described in Sections C.4.1, C.4.2, and C.4.3, respectively.

Table C-13: Maximum Neutron Dose Rate at Fuel-Water Interface (28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Gy/h	Ratio to Reference Bundle, see Table 13
220 MWh/kgU	30	1.56E-05	1.020
	100	1.02E-05	1.002
	200	9.20E-06	0.998
	300	8.72E-06	0.998
	1E+03	6.90E-06	0.996
	1E+04	2.96E-06	0.995
	1E+05	5.60E-07	1.005
	1E+06	1.55E-07	0.999
	1E+07	6.11E-08	0.983
290 MWh/kgU	30	3.57E-05	1.034
	100	1.53E-05	1.003
	200	1.29E-05	0.997
	300	1.22E-05	0.997
	1E+03	9.61E-06	0.996
	1E+04	4.24E-06	0.995
	1E+05	1.04E-06	1.005
	1E+06	2.45E-07	1.001
	1E+07	6.11E-08	0.983

C.4.5 Total Dose Rates

The maximum total dose rate in water from alpha, beta, gamma, and neutron for the single fuel bundle configuration occurs in water that fills the gap between the outer-ring fuel element and the fuel sheath. Alpha and beta sources contribute almost equally at 30 years decay time. The alpha dose rates become dominant at decay times greater than 100 years since beta and gamma emitting fission products in the fuel have decayed significantly.

The maximum dose rates in water inside the used fuel bundle envelope are listed in Table C-14. Except for the longest decay time (10 million years), the total dose rates associated with the reference fuel bundle are more limiting versus the 28-element fuel bundle.

Table C-14: Maximum Dose Rate at Fuel-Water Interface (Single Fuel Bundle, 28-Element Fuel Bundle)

Bundle Burnup	Decay Time (Years)	Total Gy/h	Percent Contribution				Ratio to Reference Fuel Bundle, see Table 14
			Alpha	Beta	Gamma	Neutron	
220 MWh/kgU	30	3.74E+02	42.9%	46.8%	10.4%	0.0%	0.995
	100	2.06E+02	80.5%	15.8%	3.7%	0.0%	0.975
	200	1.49E+02	97.5%	2.0%	0.5%	0.0%	0.969
	300	1.29E+02	99.7%	0.2%	0.1%	0.0%	0.968
	1E+03	7.12E+01	100.0%	0.0%	0.0%	0.0%	0.975
	1E+04	2.41E+01	99.9%	0.0%	0.0%	0.0%	0.988
	1E+05	1.29E+00	98.9%	0.9%	0.2%	0.0%	0.993
	1E+06	4.37E-01	96.7%	2.5%	0.8%	0.0%	0.996
	1E+07	2.79E-01	95.3%	3.5%	1.1%	0.0%	1.003
	30	4.84E+02	44.9%	44.6%	10.5%	0.0%	0.996

Bundle Burnup	Decay Time (Years)	Total Gy/h	Percent Contribution				Ratio to Reference Fuel Bundle, see Table 14
			Alpha	Beta	Gamma	Neutron	
290 MWh/kgU	100	2.70E+02	81.4%	14.9%	3.7%	0.0%	0.980
	200	1.95E+02	97.6%	1.9%	0.5%	0.0%	0.975
	300	1.68E+02	99.7%	0.2%	0.1%	0.0%	0.974
	1E+03	8.92E+01	100.0%	0.0%	0.0%	0.0%	0.980
	1E+04	2.78E+01	99.9%	0.1%	0.0%	0.0%	0.991
	1E+05	1.39E+00	98.8%	1.0%	0.2%	0.0%	0.995
	1E+06	4.96E-01	97.0%	2.3%	0.7%	0.0%	0.994
	1E+07	2.80E-01	95.3%	3.5%	1.1%	0.0%	1.001

C.5 SINGLE INTACT UFC SUBMERGED IN WATER

This scenario is comparable to the one discussed in Section 8.3.1.

An intact, air-filled, UFC is submerged in water. The dose rates (including gamma and neutron contributions) outside of the UFC are presented in Table C-15. The maximum dose rates in axial and radial directions are similar. In general, the dose rates outside of the UFC containing 28-element fuel bundles are similar to that of a UFC containing reference fuel bundles, with differences within two percent. At 1,000- and 10,000-years decay times, the differences are larger, around ten percent. At these decay times, the average energy of emitted gammas is at the lowest value (around 40 keV, see Table C-7). Thus, the differences in dose rates are approaching the observed differences in gamma emission intensity of the outer ring (see Table C-6).

Table C-15: Maximum Dose Rates at External UFC-Water Interface (Intact UFC, 28-Element Fuel Bundle)

Bundle Burnup	Decay Time (years)	Maximum Gy/h at External UFC-Water Interface		Ratio to Reference Fuel Bundle, see Table 41	
		Radial (Cylindrical Body)	Axial (Hemispherical Head)	Radial (Cylindrical Body)	Axial (Hemispherical Head)
220 MWh/kgU	30	1.22E+00	1.14E+00	1.00	1.00
	100	2.37E-01	2.20E-01	1.00	0.99
	200	2.42E-02	2.24E-02	1.00	0.99
	300	2.70E-03	2.44E-03	1.00	1.01
	1E+03	3.56E-05	3.07E-05	1.09	1.11
	1E+04	4.33E-05	3.73E-05	1.05	1.10
	1E+05	1.39E-04	1.04E-04	1.00	1.01
	1E+06	2.29E-04	1.66E-04	1.01	1.02
	1E+07	2.18E-04	1.58E-04	1.01	1.02
290 MWh/kgU	30	1.60E+00	1.50E+00	1.00	1.00
	100	3.10E-01	2.89E-01	1.00	0.99
	200	3.16E-02	2.94E-02	1.00	0.99
	300	3.48E-03	3.14E-03	1.00	0.98
	1E+03	4.95E-05	4.38E-05	1.08	1.12
	1E+04	5.48E-05	4.81E-05	1.06	1.12
	1E+05	1.51E-04	1.14E-04	1.01	1.01
	1E+06	2.32E-04	1.69E-04	1.01	1.02
	1E+07	2.18E-04	1.57E-04	1.01	1.02